DISPOSAL OF RADIOACTIVE WASTE IN THE KONRAD IRON ORE MINE

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ABSTRACT

A brief survey is given of the situation of radioactive waste disposal in the Federal Republic of Germany (FRG). For the "Konrad system" which comprises the overall geological situation, the mine and the waste forms and packagings, marginal conditions and fundamental requirements are stated as input data for the site-specific safety analysis. The procedure of analyzing the normal operation, the incidents, the heat generation of the waste and the long-term behaviour of the total system is given, emphasis being laid on the operational phase. Starting from the protection objectives dose limitation and prevention of prejudice to the barrier functions of the host rock and the underlying and overlying formations by heat generation, in the analysis requirements to be met by waste packages, especially the limitation of radioactivity, have been derived. For some important radionuclides, the function of the system of activity limitation is demonstrated.

SITUATION OF THE DISPOSAL OF RADIOACTIVE WASTE IN THE FEDERAL REPUBLIC OF GERMANY

Since 1965, procedures and techniques for the safe disposal of radioactive waste have been developed and tested in the abandoned Asse salt mine near Braunschweig. Within the scope of a test programme implemented from 1965 to 1978, about 125 000 drums with low-level waste and 1300 drums with medium-level waste were disposed of[1]. Since 1979, the work has been focussed on in-situ investigations, chiefly in connection with the disposal of heat-generating waste in salt formations. In the course of 1987 a decision will presumably be taken as to whether the Asse mine will additionally serve as a repository. For this purpose the cavities still existing in the mine would make enough space available to take up some 100 000 drums with radioactive waste.

Since 1979 investigations have been carried out by the PTB into the Gorleben salt dome's suitability for the final disposal of all kinds of solid waste, especially heat generating waste. At Gorleben which is situated about 150 km north-east of Braunschweig, preparations are at present being made for the underground exploration after the exploration from above ground has been completed[2]. If the underground investigations yields positive results and if the plans have been approved probably in 1995, the construction of the repository mine and the driving of storage chambers could then be started. First emplacements of radioactive waste will be possible presumably by the end of the nineties.

From 1978 to 1981, a site-independent concept was developed for the Gorleben project. The PTB is now starting on the preliminary planning work for the establishment of a site-specific concept. Within the scope of this work it is intended, among other things, to define the emplacement methods for all kinds of waste, particularly for vitrified high-level waste from reprocessing and for conditioned non-reprocessed fuel elements. Simultaneously, parameter investigations will be carried out on the same subject within the scope of a research and development project[3].

Investigations into the suitability of the Konrad iron ore mine near Braunschweig have been in progress since 1975. It is planned to dispose of that waste which has a negligible thermal effect upon the surrounding rock. The iron ore opened up in its southern part by two shafts, was deposited about 150 million years ago during the Upper Jurassic (Malm) and was discovered only at the beginning of the thirties in connection with oil explorations (Fig. 1).

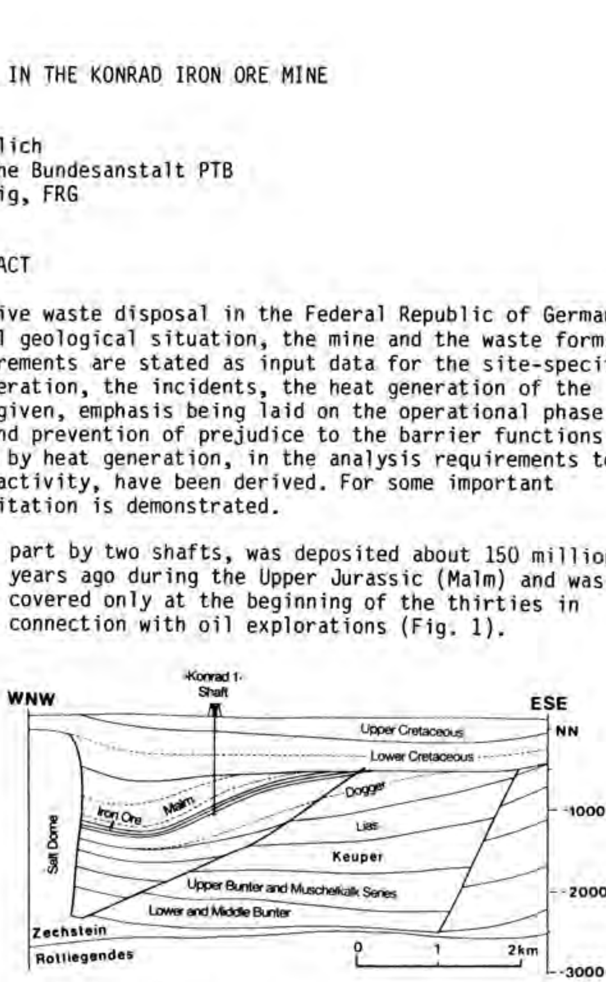

Fig. 1. Geological Cross Section of the Konrad Site

The Konrad mine is exceptionally dry for an iron ore mine. The position of the iron ore body at a great depth (800m - 1300 m) and the minor permeability to groundwater due to the predominantly argillaceous composition of the overlying strata form the base for the safe deposition of radioactive waste. On August 31, 1982, the PTB filed an application for the initiation of a plan-approval procedure for construction and operation of a repository. At the same time, underground work and supplementary work above-ground within the scope of the site investigation programme was carried out, the results of which are expected to demonstrate the suitability of the mine in the plan-approval procedure.

In the planned repository the shaft Konrad 1 will be used for the transport of the iron ore extracted during the construction of the emplacement rooms and for the transport of material and manriding. The upcast ventilating shaft Konrad 2 will be used for the transport of the radioactive waste in the mine. Emplacement operations and the excavation of cavities underground are to be kept separate in space and also in time (Fig. 2). For emplacement operations, storage galleries (emplacement rooms or chambers) with a cross section of about 40 m^2 must be driven. In these galleries the radioactive waste will be stacked. The remaining cavities will be backfilled with rock material from the excavation of the galleries.

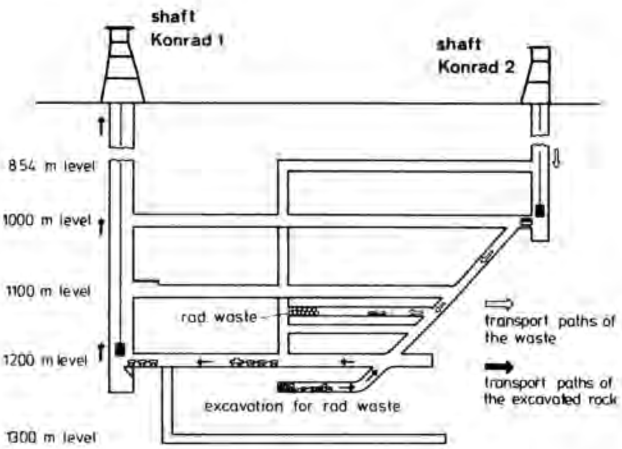

Fig. 2. Principle of the Planned Excavation and Emplacement Operations.

The underground mine area at present under consideration allows emplacement rooms of about 650 000 m^3 in volume to be driven. On the basis of the waste package volume to be disposed of annually (about 20 000 m^3 per shift), a period of operation of more than twenty years is calculated. This would mean that during its period of operation, the Konrad mine could be used for the disposal of about 95 % of the volume of all the radioactive waste produced in the Federal Republic.

It is planned to stack the waste in cylindrical concrete containers (lost concrete shieldings), in cylindrical cast iron containers or in rectangular concrete, steel or cast iron containers in chambers up to 1000 m in length with the aid of shielded vehicles, to backfill the remaining cavities with rock material from the excavations and to seal each filled chamber with a closing-off structure acting as a barrier against air-borne, volatile radionuclides released from the waste. The accesses to the abandoned, filled emplacement fields and working galleries and areas no longer used are also backfilled with rock material. At the end of operation, both shafts will be provided with a locking system acting as a barrier against the penetration by solutions from the repository into overlying formations and vice versa against the penetration by formation waters from the overlying rock into the repository.

Providing a plan-approval, the emplacement of radioactive waste could possibly be started by the end of the eighties.

SITE-SPECIFIC SAFETY ANALYSIS FOR THE KONRAD MINE

General

Pursuant to the "Safety criteria for the disposal of radioactive waste in a mine"[4], the safety of a repository must be proved within the scope of a site-specific safety analysis. A distinction has to be made between the operational and the post-operational phase. In the operational phase, the radiological effects in normal operation and in the case of an incident are to be analysed and evaluated with respect to the objective "limitation of individual doses" of the Radiation Protection Ordinance. The influences of the heat generation of the deposited waste on the host rock and the overlying rock are to be determined both for the operational and the post-operational phase and evaluated with respect to the objective "intactness of the geological barriers". For the post-operational phase the long-term radiological effects in the biosphere are to be calculated and, for a reasonable period of time in the future, are to be evaluated with respect to the objective "limitation of individual doses" noted in the "Safety criteria"[4]. From the four partial analyses independent requirements for the design and construction of the repository[5] and for the waste packages (radionuclide inventory, waste form and packaging; this presentation) are derived. In each case the most restrictive requirement must be complied with.

In the following, chiefly the operational phase will be dealt with as the analysis of the post-operational phase is not yet completed.

Operational phase, normal operation

The radiological effects concern staff and environment of the repository. They are due to external radiation and the release of radioactive substances from the waste and their controlled discharge into the environment. The effects of external radiation are kept small by limitation of the dose rate of the waste packages and by the shieldings in the repository as well as by administrative measures[5]. The effects of the release of radioactive substances into the environment are limited by the barriers of the waste form, the packaging, the closing-off structure of filled emplacement rooms, and by dilutions in the case of propagation in air and water. For conservative reasons, a sedimentation of activity in the mine workings is not assumed. In addition, the contamination adhering loosely to the surface of the waste packages is limited.

The activity limitations in the mine and in the individual waste packages are derived in the following three steps, the starting dose being defined by the dose limitation for the environment specified by the Radiation Protection Ordinance:

1. Determination of the release rates applied for (air) for the volatile radionuclides and the radionuclide groups H-3, C-14, I-129, Rn-222, β/γ-aerosols, and α-aerosols by

   - nuclide (group)-specific propagation and radioecological calculations to determine the environmental dose per activity released,
   - weighting of the radionuclides and radionuclide groups according to the frequency of their occurrence in the waste and their volatility,
   - normalization of the mixture composition obtained to activity values so that the doses are clearly inferior to the limiting dose values in accordance with the minimization prescription of the Radiation Protection Ordinance (20 % of the prescribed standard value of 0,3 mSv/a effective dose),
   - taking into account of the secondary condition ≤ 0,5 mSv/a effective inhalation dose for the staff (avoidable exposure) and clear inferiority to the limiting dose values via the path of liquid effluents (mine water), as well as
   - establishing of the annual release rates applied for (air and water), see Table I.

The release rates applied for for H-3, C-14 and Rn-222 are essentially determined by the limitation of the inhalation dose of the staff.

2. Derivation of the quantitative relationship between radionuclide inventory and release, with reference to the specific radionuclides and in dependence upon their physical and chemical form, by

   - introduction of two groups of waste forms (1st barrier), of specifiable leaktightnesses of packagings (2nd barrier) as well as of marginal conditions for the properties of the closing-

TABLE I

Release Rates of the Repository Applied For.

Radionuclide / Group of Radionuclides	Release Rates Bq / year	
H - 3	$1{,}48 \cdot 10^{13}$	
C - 14	$3{,}7 \cdot 10^{11}$	
I - 129	$7{,}4 \cdot 10^{6}$	via air
Rn - 222	$1{,}85 \cdot 10^{12}$	
aerosoles ($T_{1/2}$> 10d)		
$\beta\gamma$-emitters	$7{,}4 \cdot 10^{7}$	
α-emitters	$3{,}7 \cdot 10^{6}$	
H - 3	$7{,}4 \cdot 10^{12}$	via water
other radionuclides	$7{,}4 \cdot 10^{8}$	

off structure of emplacement rooms (3rd barrier), and

- determination of the annual portions released for both groups of waste forms without and with leaktight packaging as well as the hold-back properties of the closing-off structures.

In a simplified form Fig. 3 shows the annual portions released from the waste form (1st barrier). The models for the derivation of these release rates which are described in detail in[6] use, among other things, the following mechanisms: diffusion through solids, formation of HT by radiolysis, evaporation either directly or subsequently to leaching by residual water, transition to the gas phase in the case of evaporation of water, solubility, radioactive decay. In addition, release rates defined by way of experiment were used as far as such rates were available. The values of the portions released determined in this way are upper-limit estimates.

For the packagings (2nd barrier), a distinction is made between unspecified leaktightness (no hold-back capacity) and specified leaktightness with permeabilities of 10^{-2}/a, 10^{-3}/a, 10^{-4}/a and better. A permeability of 10^{-2}/a, for example, is in any case attained by a type B packaging (transport regulations) for which a leakage rate of $\leq 10^{-5}$ Pa$\cdot$m^3/s is prescribed.

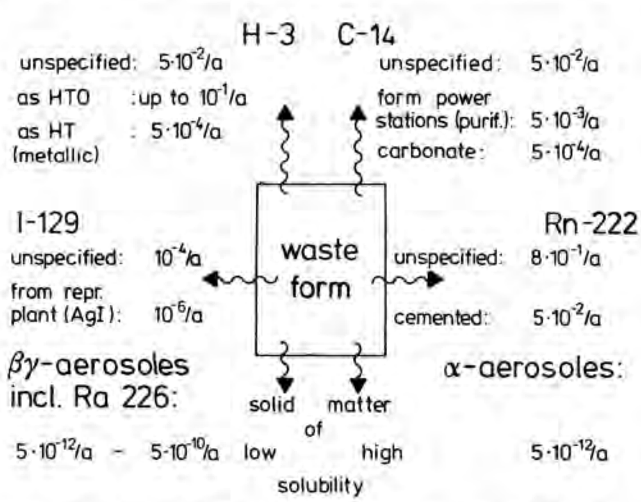

Fig. 3. Annual Portions of Radioactivity Released from the Waste Form (Simplified).

The closing-off structures (3rd barrier) of filled emplacement rooms are designed in such a way that release from a chamber due to air pressure variations does not occur. Responsible for release from a chamber into ventilated mine workings are rather the diffusion (light gases) through the closing-off structures and loosened surrounding rock, and above all the forced decrease in the remaining cavity volume in the chamber due to rock convergency. Details are described elsewhere[6,9]. Especially for HTO it was recognized that the release is limited by the evaporation equilibrium of the water and HTO system in the sealed emplacement rooms. An essential aspect is that a specified leaktightness of packagings can no longer be taken for granted after a chamber has been closed off, as quantifiable statements on the mechanical intactness of the containers and their influences on the tightness cannot be made. Fig. 4 is a simplified representation of a filled emplacement room provided with a closing-off structure as well as of the annual permeability portions of the radionuclides through this structure.

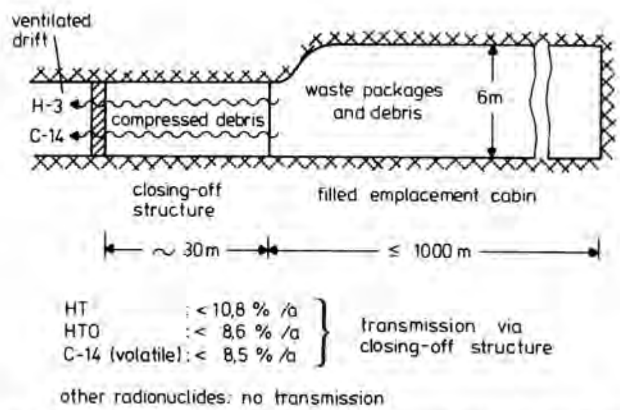

Fig. 4. Annual Portions of Radioactivity Permeating through the Closing-off Structure of Filled Emplacement Rooms.

3. Determination of the maximum emplaceable activity with reference to the specific radionuclides and in dependence upon their physical and chemical form by backward calculation on the basis of the annual release rates applied for, with regard to the design and operation concept for the repository.

The following two relations must be fulfilled independently of each other for the respective waste form group and for the given leaktightness of the packagings:

$$a_i \cdot n_o \cdot f_{io} + a_i \cdot n_a \cdot f_{ia} < G_i \quad (1)$$

$$\sum_i \frac{a_i\, n_o\, f_{io} + a_i\, n_a\, f_{ia}}{ALI_i} \cdot t_A \cdot \frac{AR}{LR} < 0{,}01 \quad (2)$$

where:

a_i maximum activity of the i-th radionuclide or radionuclide group of Table I, which can be emplaced in the Konrad mine per year

f_{io}, f_{ia} relative release rate of the i-th radionuclide or radionuclide group per year from open (o) or closed-off (a) emplacement rooms

n_o, n_a number of the open (o) or closed-off (a) emplacement rooms filled with waste

G_i annual release rate applied for for the discharge of the i-th radionuclide or radionuclide group with exhaust air
ALI_i limiting value of the activity intake per year for persons of category A according to the Radiation Protection Ordinance, exposed to radiation
AR annual rate of inhalation during the working hours in the mine
t_A portion of the hours worked in exhaust air from emplacement rooms
LR minimum ventilating current in exhaust air from emplacement rooms.

Operational phase, incidents

The radiological effects concern the staff and the environment of the repository and are chiefly due to the release of radioactive substances from the waste packages. They are limited by precautionary measures, by the barriers of the waste form, the packaging, the mine workings, and by dilution when propagating in the atmosphere. The derivation of the activity limits per waste package is carried out in the following five steps and is based on the standard dose values in the environment prescribed by the Radiation Protection Ordinance for the planning.

1. Systematic analysis of the planned operating sequences in the repository with incident situations as well as potential site-dependent and site-independent influences upon the repository such as external fire or earthquakes in accordance with the Incident Guidelines[7].

2. Determination of few incidents of this kind which cover the great number of potential incidents according to 1. with respect to the release of radionuclides and the radiological effects; quantification of the appurtenant external impacts upon a waste package.

The analysis reveals the following three events which are representative from the radiological viewpoint:

- drop of a waste package during handling from 3 m height, onto the floor of the hall (above ground),
- drop of a waste package during emplacement in the chamber from 5 m height and
- collision of a vehicle resulting in a fire during waste transport in a transport gallery. For the fire a fire temperature of 800 °C for 1 hour is assumed.

3. Establishment of the quantitative relationships between radionuclide inventory and release from the waste package, with reference to the specific nuclides, by

   - introduction of six groups af waste forms according to the different release behaviour (requirement to be met by the waste forms) in the incidents according to 2.,
   - introduction of two waste classes according to the different release behaviour of the packagings in the incidents according to 2. (requirement to be met by the packaging), and
   - determination of the portions released in the form of the grain-size spectrum of the released radioactive substances for the six waste form groups and waste classes I and II with reference to the specific nuclides for the incidents according to 2. Consideration of holdback properties of the mine workings[8] and the surface installations.

The definition of the waste form groups, waste classes as well as the derivation of the portions released which is based partly on experiments on the fragmentation behaviour of some waste forms, is described in detail elsewhere[9].

4. Determination of the source term prior to the discharge into the atmosphere with reference to the specific nuclides of the six waste form groups and waste classes I and II separately for the incidents according to 2., normalized to the respective unit of activity in the waste package. Atmospheric propagation and radioecological calculations with these source terms to determine the exposure to radiation in the environment.

The calculations are carried out in accordance with the Federal Incident Calculation Bases[10] using parameters for particles with a size of more than 10 µm in the Federal Technical Directives for the Prevention of Air Pollutions[11].

5. Determination of the limits of activity per waste package with reference to the specific nuclides of the six waste from groups and waste classes I an II, separately for the three incidents according to 2. by way of backward calculations starting from the prescribed standard dose values in the environment. Determination of the most restrictive of the three activity limits for each case and taking into account of the simultaneous existence of several radionuclides in a waste package by a summation criterion for the nuclide-specific activities.

$$S_s(p,k) = V \cdot \sum_i \frac{A(i)}{G_s(i,p,k)} < 1 \qquad (3)$$

where:

S_s summation value (s = index for incident)
$A(i)$ activity of the radionuclide i in the waste package
$G_s(i,p,k)$ limit of activity of the radionuclide or radionuclide group i determined from incident calculations for the waste form group p and waste class k (k = I or k = II)
V packaging factor (= number of waste packages handled on a pool pallet; V = 1, 2 or 3).

If the summation values S_s reaches the value 1, the limiting dose for the environment (effective dose or dose for the critical organ) is just reached in one of the three incidents.

Operational phase, post-operational phase: generation of heat by the waste

The behaviour of the host rock (Minnette-type iron ore) and of the surrounding rock under the influence of considerable quantities of heat is not known and cannot be quantified. For this reason, it is intended to emplace only radioactive waste with neglectable thermal effect upon the host rock. This requirement can be considered to be met when the temperature increase at the chamber wall caused by the decay heat of the radionuclides contained in the waste does not exceed 3 K. This value corresponds approximately to the temperature difference at a difference in depth of 100 m in the natural temperature field, and it is small as compared with the temperature variations to which the rock is subjected when cavities are driven, ventilated and closed again. These variations are up to about 20 K.

On the basis of the temperature limitation mentioned above, activity limits for the waste packages are derived. The analysis is based on the following assumptions and carried out in the following steps:

1. Homogeneous distribution of the acitivity and, thus, of the heat output in an emplacement room. Consideration of a period of time of up to 100 000 years.

2. Determination of a limiting heat output $W^*(i)$ per unit of length in the emplacement room for each radionuclide i so that in the absence of other heat sources (radionuclides) the maximum temperature increase at the chamber wall is 3 K.

3. Conversion of the length-related, nuclide-specific limiting heat outputs $W^*(i)$ into nuclide-specific, package-specific (B) limits of activity $G_w(i,B)$ (w = index for heat).

The conversion is made by means of package factors which are a function of the number of the waste packages in the cross-section of an emplacement room and of the length sections required for stacking the waste packages. Furthermore, the energy release per decay enters into the conversion.

4. Transition to a summation criterion for the nuclide-specific activities A(i) in a waste package for mixtures of radionuclides.

$$S_w(B) = \sum_i \frac{A(i)}{G_w(i,B)} < 1 \qquad (4)$$

5. Taking into account of mixtures of different radionuclide composition in the waste in the cross-section and along the axis of a chamber. Formulation of summation criteria.

In the derivation of the requirements to be met by waste packages from the effects heat exerts upon the host rock, the heat output is homogeneously distributed by calculation over the emplacement rooms. It is, however, also possible to emplace a mixture of waste packages with different nuclide mixtures in one chamber when waste packages with summation values of more than 1 are emplaced together with waste packages whose summation values are accordingly low. In doing so, the volume-weighted mean of the summation values is to be taken on the basis of a maximum of three stacking sections.

Post-operational phase, long-term radiological effects

For the post-operational phase of a repository, all considerations must start from the problem of potential inflows of solutions to the product emplaced. If the inflow of water into the remaining cavities of a repository cannot be precluded, the release of radioactive substances from the waste and their propagation with the transport medium in the repository as well as through the repository formation and the overlying and adjoining rock into the biosphere must be used to prepare a model for determining the potential exposure to radiation.

For the post-operational phase of the Konrad repository, influx of solution up to the product emplaced is assumed because slight inflows of solutions from drip and swallow holes in the host rock were observed so that they cannot be precluded. As source term of the activity for calculating the propagation in the repository and the subsequent transport with the medium water through the host formation (Oxford) as well as through the overlying and adjoining rock into the biosphere, a volume source is used. Its activity density is essentially determined by the limits of the solubility of the individual radioactive substances in water. The transport of the radioactive solution initially takes place almost exclusively along the host rock, as the impermeability of the neighbouring formations is relatively high as compared with the host rock formations. For the rough model at present under preparation, the following permeability values are started from: 10^{-14} m^2 for the Oxford formation, 10^{-15} m^2, 10^{-12} m^2 and 10^{-17} m^2 for the three upper superposed formations, 10^{-13} m^2 and 10^{-17} m^2 for the two lower superposed formations. A strip 14 km in width and 48,5 km in length in the north-north-eastern direction was defined as region of significance for propagation. At this distance from the repository, the host formations is in contact with an aquifer.

ACTIVITY LIMITATIONS

The requirements on design and construction of the repository resulting from the safety analysis will be presented in a contribution to an IAEA-Symposium in Hannover[5]. The following sections deal with the results concerning the requirements on the waste, especially the acitivity limitations.

Normal operation

The limits of activity per year a_i are shown in Table II for waste forms in packagings without any leaktightness and with leaktightness, resulting in an annual portion of release of 10^{-2}.

Conservative upper limits of activity $a^*_i = 10^{-4} . a_i$ per waste package result from the assumption that up to 10^4 packages are emplaced in a chamber. The values a_i or a^*_i can, at the same time, be taken to the limit specified for the individual radionuclides and radionuclide groups. In this connection attention must be given to the fact that, if a radionuclide is contained in a waste package in various specifications - e.g. C-14 unspecified and as carbonate - the most restrictive limiting value of the activity must be taken as a basis. The activities emplaced in the Konrad repository are balanced nuclide by nuclide. If the balancing of a certain radionuclide or radionuclide group shows for the running year of operation that the limits of activity a_i emplaceable in the Konrad repository per year are not reached, waste packages which exceed the limits of activity a^*_i per waste package can also be emplaced if the limiting values a_i are complied with.

Incidents

In addition to the limiting conditions of normal operation of the plant maximum permissible activities (limits of activity) per waste package are obtained from the incident analysis. The selected individual nuclides cover all the radionuclides with half-lives of more than 10 days which occur in the radioactive waste intendend to be disposed of in the Konrad mine. The limits of activity per waste package are shown in Table III for waste class I for 14 so called key radionuclides and for other α- and β/γ-emitters. The limits of activity for the waste class II are higher by a factor of about 80 to 800 except for the waste form bitumen and plastic products for which factors of about 200 - 1600 are valid.

Among the key nuclides are the 14 radionuclides which are of special radiological significance for incidents. For the other radionuclides which range among the other α- and β/γ-emitters and which, as compared with the key nuclides, are radiologically less important, the limits of activity of the other α- and β/γ-emitters can be used when the sum rule (Eq. (3)) is applied. If $S_S > 1$ compliance with the summation criterion can possibly be proved by indicating the activities of some of the other individual nuclides and by appropriately applying the limits of activity of these radionuclides which are generally less restrictive.

TABLE II

Limits of Activity for Radionuclides and Groups of Radionuclides Disposable per Year, Resulting from Safety Analysis - Numbers Are Given in Bq/Year

Radionuclide / Group of Radionuclides	Non Metallic Waste form except for HT	
	Packaging without Specified Tightness	Packaging with Specif. Tightness Annual Portion of Release ≤ 0,01
H - 3		
- unspecified	$2,3 \cdot 10^{13}$	$7,2 \cdot 10^{13}$
- as HTO * a)	$6,9 \cdot 10^{13}$	$6,9 \cdot 10^{15}$
b)	$6,6 \cdot 10^{13}$	$2,4 \cdot 10^{15}$
c)	$2,3 \cdot 10^{13}$	$7,2 \cdot 10^{13}$
- as HT **	$4,8 \cdot 10^{15}$	$9,6 \cdot 10^{15}$
C - 14		
- unspecified	$1,0 \cdot 10^{12}$	$1,9 \cdot 10^{12}$
- from clarific. of primary coolant in nucl. power stations	$1,0 \cdot 10^{13}$	$1,9 \cdot 10^{13}$
- as carbonate	$1,0 \cdot 10^{14}$	$1,9 \cdot 10^{14}$
I - 129		
- unspecified	$2,1 \cdot 10^{10}$	$2,1 \cdot 10^{12}$
- as AgI on filters from off gas purification in repr. plants	$2,1 \cdot 10^{12}$	
βγ-emitters d)	$4,2 \cdot 10^{18}$	$4,2 \cdot 10^{20}$
e)	$4,2 \cdot 10^{16}$	$4,2 \cdot 10^{18}$
α-emitters	$2,1 \cdot 10^{17}$	$2,1 \cdot 10^{19}$
Ra - 226		
- unspecified	$1,0 \cdot 10^{10}$	$5,2 \cdot 10^{13}$
- cemented	$1,6 \cdot 10^{11}$	

* Concentration of H-3 as HTO in the water resp. the residual moisture of the waste form $\leq 1,1 \cdot 10^{13}$ Bq/m³. Whole activity in the waste form without H-3 activity: a) $< 10^{10}$ Bq; b) $\geq 10^{10}$ Bq, $< 10^{12}$ Bq; c) $\geq 10^{12}$ Bq. Percentage (mass) of H_2O in the waste form d) < 1 %, e) ≥ 1 %.

** In metallic solid matter.

For a potential rest of unspecified α- or β/γ-emitters, the limit of activity of the other α- or β/γ-emitters must again be used when the sum rule is applied.

Heat generation by the waste

The length-related initial limiting heat outputs in an emplacement room vary between the values $3,35 \cdot 10^{2}$ W/m and $6,83 \cdot 10^{-3}$ W/m according to whether the halflives are a few days or $\gg 10^{5}$ years. Typical examples of this are 19,4 W/m for Co-60, 2,32 W/m for *Cs-137*, 0,33 W/m for Ra-226 and 0,077 W/m for Th-232. For the limiting values of the activity of the individual nuclides, the range is larger and the order modified as here the decay energy is to be allowed for in addition to the half-life. For reasons of simplicity, *Table IV* does not give the package-related limits of activity $G_W(i,B)$ but the length-related limits of activity G^*_W. According to the container size, the $G_W(i,B)$ amount to different fractions of G^*_W: $G_W(i,B)/G^*_W$ = 5 % to 12 % for cylindrical concrete containers (lost concrete shieldings), = 3 % to 5 % for cylindrical cast iron containers, and = 15 % to 40 % for rectangular containers.

TABLE III

Limits of Activity for Key Radionuclides and Other Unspecified α- and β/γ - Emitters for the Different Waste Form Groups (Waste Class I), Resulting from Incident Analysis.
- Numbers Are Given in Bq/Waste Package -

Radionuclide / Group of Radionuclides	Cemented / Concrete Waste; Concentrates	Radionuclide / Group of Radionuclides	Cemented / Concrete Waste; Concentrates
Cl - 36	$7,3 \cdot 10^{8}$	Ac - 227	$9,6 \cdot 10^{12}$
I - 129	$2,7 \cdot 10^{9}$	Tc - 99	$1,1 \cdot 10^{13}$
Ra - 226	$4,9 \cdot 10^{11}$	Am - 242M	$1,5 \cdot 10^{13}$
Sr - 90	$1,9 \cdot 10^{12}$	Cs - 137	$4,0 \cdot 10^{13}$
Pb - 210	$3,1 \cdot 10^{12}$	Co - 60	$4,9 \cdot 10^{13}$
Pa - 231	$3,3 \cdot 10^{12}$		
Th - 232	$4,4 \cdot 10^{12}$	other α-emitters	$6,7 \cdot 10^{12}$
Ra - 228	$4,9 \cdot 10^{12}$		
Pu - 239	$6,7 \cdot 10^{12}$	other βγ-emitters	$4,9 \cdot 10^{12}$

The limits of activity for the other waste form groups referred to the values given in the table are: solid matter: 5 %, metallic solid matter: 12 %, compacted waste: 30 %, bitumen and plastic products: 1 ‰.
For Cl-36 and I-129 the values are equal for all six waste form groups.

The radionuclides stated in Table IV are only the 13 first α-emitters, the 14 first β/γ-emitters and the frequently ocurring radionuclides Pu-238, Cs-137, Ni-63 und Fe-55 from the list of all radionuclides contained in the waste, listed in the order of decreasing significance for the influence exerted by heat. They are the key nuclides for determining the limitation of heat in the waste. The limits of activity for

TABLE IV

Limits of Activity per Meter Length of an Emplacement Room for the Key Radionuclides, Resulting from Limitation of Heat Production

Radionuclide	Limit of Activity in Bq/m	Radionuclide	Limit of Activity in Bq/m
Th - 232	$1,2 \cdot 10^{11}$	Ni - 59	$5,9 \cdot 10^{12}$
U - 235	$1,3 \cdot 10^{11}$	Pu - 238	$7,9 \cdot 10^{12}$
U - 233	$1,6 \cdot 10^{11}$	Pb - 210	$1,3 \cdot 10^{13}$
Th - 230	$1,7 \cdot 10^{11}$	Ca - 41	$1,5 \cdot 10^{13}$
Pa - 231	$1,8 \cdot 10^{11}$	Ag - 108M	$2,3 \cdot 10^{13}$
U - 234	$2,3 \cdot 10^{11}$	Cl - 36	$2,3 \cdot 10^{13}$
Cm - 248	$2,7 \cdot 10^{11}$	Be - 10	$2,7 \cdot 10^{13}$
Np - 237	$2,9 \cdot 10^{11}$	Sn - 126	$2,9 \cdot 10^{13}$
Cm - 247	$3,1 \cdot 10^{11}$	Rb - 87	$3,8 \cdot 10^{13}$
Pu - 244	$4,1 \cdot 10^{11}$	Co - 60	$4,7 \cdot 10^{13}$
Ra - 226	$4,2 \cdot 10^{11}$	Ar - 39	$4,7 \cdot 10^{13}$
U - 238	$4,6 \cdot 10^{11}$	Cs - 137	$7,8 \cdot 10^{13}$
Cm - 245	$8,0 \cdot 10^{11}$	Ni - 63	$6,6 \cdot 10^{14}$
Ac - 227	$2,2 \cdot 10^{12}$	Fe - 55	$6,7 \cdot 10^{14}$
Am - 242M	$3,2 \cdot 10^{12}$		
Ra - 228	$3,3 \cdot 10^{12}$	other α-emitters	$1,1 \cdot 10^{12}$
Nb - 94	$4,4 \cdot 10^{12}$	other β-emitters	$5,9 \cdot 10^{13}$

unspecified α- and β/γ-emitters are then defined by the limits for the most restrictive of the other α-(U-236) and β/γ-emitters (Sr-90).

As in the case of incidents, the summation criterion Eq.(4) for heat limitation can be applied in two ways: Either the limits of activity for 31 key nuclides and for unspecified α- and β/γ-emitters (cf. Table IV) or - if necessary - those for the 31 key nuclides, other individual nuclides and the unspecified α- and β/γ-emitters from Table IV are used. Furthermore, it is possible to have recourse to mixed emplacement. In isolated cases, waste packages B with a summation value $S_W(B)$ of up to 60 can be emplaced provided that enough packages with summation values of less than 0,01 are available.

Long-term radiological effects

Results of the model calculations for determining the radiological effects are not yet available. On the basis of the hydrogeological conditions, the periods of time water takes to cover the distance from the repository to the biosphere of the order of about 10^5 years are to be reckoned with. It is therefore to be expected that restrictive limitations of the acitivity can ensue at most for very long-lived radionuclides or their daughter products for which in the geosphere relatively small K_d-values (constants of equilibrium adsorption distribution) are valid. To these radionuclides Ni-59, Se-79, Tc-99, I-129, Np-237 and U-238 can belong.

DISCUSSION

Owing to the variety of the primary limitations of activity which are related to a year in normal operation, to a waste package in the case of incidents, to the unit of length of the emplacement room for the generation of heat, and to the whole repository in the post-operational phase, it is not possible to uniformly cover the limiting values by a single list of requirements. They apply simultaneously and independently of each other, i.e. in each isolated case, the most restrictive requirement must be met. Only with the requirements derived from incidents and from the limitation of heat are simplifications possible which will be made in the tables of the final emplacement conditions.

In conclusion, some essential activity limitations from the four ranges will be discussed.

H-3

The emplaceable activity practically depends only on normal operation. Standard HTO-containing waste produced by nuclear power plants such as filter and evaporator bottoms or resins of clarification of primary coolant can be emplaced without additional balancing, as the conservative upper limits of activity per package $a^*_{H-3} = 10^{-4} \cdot a_{H-3}$ are not reached. Higher H-3 activities as HTO from a reprocessing plant (400 t/a) with about $3,7 \cdot 10^{15}$ Bq/a in a cemented form can be emplaced in the Konrad mine only after balancing and only in packagings of specified leaktightness. At present the disposal of this HTO waste from reprocessing by sinking into deep-lying geological strata is given preference. Under the present conditions of the maximum quantities of H-3 waste of different specification which can be emplaced per year, about 50 % to 80 % will be actually emplaced.

C-14

As with H-3 waste, the emplaceable activity is practically determined only by normal operation. The waste produced in the clarification of primary coolant in nuclear power plants does not reach the upper limit of activity: $A(C\text{-}14) < a^*_{C-14} = 1 \cdot 10^9$ Bq/package, so it can presumably be emplaced without additional balancing. Furthermore, C-14 is contained in industrial waste and waste produced by research laboratories; this waste can partly be emplaced only after balancing. Under the present conditions of the maximum quantities of C-14 waste of different specification which can be emplaced per year, about 15 % to 30 % will be actually emplaced; further waste can be reckoned with.

I-129

Here two kinds of waste are concerned: iodic waste from nuclear power plants and from the offgas clarification in reprocessing plants. The waste mentioned first is supplied in containers without specified leaktightness and can presumably be emplaced without separate balancing. The second kind of waste contains I-129 in the form of AgI on filters which are cemented in and are to be supplied in lost concrete shieldings with $A(I\text{-}129) = 10^{11}$ Bq. Emplacement of this waste is possible only after separate I-129 balancing. In addition, to allow for potentail incidents, the packaging must comply with the requirements to be met by class II waste. It would have to be proved that the lost concrete shielding meets these requirements. At present, the question as to whether additional activity limitations result for I-129 from the analysis of the post-operational phase cannot yet be answered. With this reservation, the limiting value of the quantities of I-129 which can be emplaced per year is exploited to about 45 %.

Ra-226/Rn-222

The most different forms of Ra-226 with about 10^{13} Bq, essentially from medical applications, are at present stored in the Federal Republic. In some cases, this waste was conditioned in containers of specified leaktightness; in addition, research and development programmes are being carried out with a view to finding other suitable forms of conditioning waste. In most cases emplacement will be possible only on the basis of separate balancing with a view to normal operation.

Co-60, Cs-137

The possibilities of emplacing waste which contains these radionuclides are determined exclusively by the limiting values of the activity per waste package from the point of view of incidents and heat generation. In addition, due to the limitation of the dose rate of the packages und limited shielding possibilities, limitations of the activity can result.

Cumulated activity

On the condition that the activity limitations resulting from the safety analysis of the post-operational phase are not more restrictive than those resulting from the other partial analyses, it is to be expected that at the end of the period of operation of the Konrad mine the cumulated radionuclide inventory will be of the order of about $8 \cdot 10^{17}$ Bq, a half being due to fission products, a quarter due to actinides, and a quarter due to light elements. This inventory approximately corresponds to the waste from nuclear power plants producing 300 to 400 GWa electrical energy, to the waste with negligible heat generation from the reprocessing of about 12 000 t of burnt-up nuclear fuel, to the waste from medical, industrial and research applications of about 30 years, and to the waste to be disposed of from the decommissioning of five small reactor plants.

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