THERMAL INFLUENCE OF THE DECAY HEAT AND RESULTING REQUIREMENTS ON THE DISPOSAL STRATEGY FOR WASTE PACKAGES

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ABSTRACT

Within the scope of a comprehensive safety assessment for a repository, the resulting effects of the decay heat generated by the emplaced radioactive waste must also be analyzed. Based on the planning for the German Konrad repository intended to dispose of radioactive waste with negligible heat generation, calculations of the thermal influence upon the host rock and resulting requirements on the emplacement of the waste packages are carried out and described.

INTRODUCTION

The objective of radioactive waste disposal in repositories is to ensure that waste is handled and stored in such a way that the protection of man and the environment from harm caused by the ionizing radiation of the waste is guaranteed. The basic aspects which must be taken into account to achieve this objective are compiled in the "Safety Criteria for the Disposal of Radioactive Waste in a Mine" (1) which state, among other things, that the required safety of a repository constructed in a geological formation must be demonstrated by a site-specific safety assessment which includes the respective overall geological situation, the technical concept of the repository and the waste packages intended to be disposed of.

The safety criteria specify the measures to be taken in order to achieve the objective of disposal and define the principles by which it must be demonstrated that this objective has been reached. Consequently, within the scope of a site-specific safety assessment, the Bundesamt für Strahlenschutz (BfS) being responsible for the disposal of radioactive waste in the Federal Republic of Germany investigated the following effects:

- In the operational phase of a repository, the radiation exposure of the personnel and of individuals in the environment of the facility to direct or scattered radiation as well as to radionuclides released via the air path or by liquid effluents must be judged.
- In incident scenarios in the operational phase, mechanical and/or thermal impacts on the waste packages must be considered.
- Additionally, the criticality safety and the resulting effects of the decay heat generation per waste package upon the host rock must be analyzed.
- In the post-operational phase of a repository, the radiation exposure of individuals in the biosphere due to radionuclides released form waste packages and transported via the water path is to be evaluated.

The heat generated during the decay of radionuclides within the radioactive waste which is disposed of in a repository leads to a temperature increase in the emplacement room or in the borehole and in the surrounding rock. Therefore, the thermal influence on the host rock is one part of the safety assessment.

Due to the degree of heat production of the radionuclides and due to the knowledge on the material law of the geological formation, the safety assessment concerning the thermal influence upon the host rock has to be performed in a comprehensive manner. Depending on kind and extent of the heat propagation short-term or long-term instabilities of the rock may arise leading to possible stability problems of the excavated emplacement rooms and transport galleries during the operational phase of the repository. Moreover, the thermal influence may lead to long-term changes of the geological barrier, for example the arising of new pathways where the radionuclides may be transported from the emplacement level to the biosphere.

The basic assumptions and procedures of deriving requirements from the results of such an assessment are described using as an example the German Konrad repository.

SAFETY ASSESSMENT AND RESULTING REQUIREMENTS ON WASTE PACKAGES

In the abandoned Konrad iron ore mine in the southeast of Lower Saxony at Salzgitter, it is intended to dispose of such radioactive waste which has a negligible thermal influence upon the host rock. This German classification of waste type is based on repository-specific conditions. Waste with negligible heat generation contains mainly low-level waste and a part of medium-level waste including radionuclides with long half-lives but no high-level waste and no spent fuel elements.

The reason for this decision is the fact that the theromechanical behavior 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 easily quantified due to the inhomogeneous character of the geological formations.

According to the experience gained in the operation of the Konrad iron ore mine, no effects which are significant from the point of view of safety engineering and which are due to the heating-up of the host rock in connection with the heat dissipated from the radioactive waste are to be expected if the temperature variations in the rock of the emplacement rooms are small compared with those caused by mine ventilation. These temperature variations are up to about 20 K.

Hence, the above mentioned marginal condition is considered to be met with a temperature variation of 3 K inside the rock. Moreover, this value corresponds approximately to the temperature difference at a difference in depth of 100 m in the natural temperature field. It should be noticed that the waste disposal is planned in depths of 800 m up to 1300 m.

Therefore, a prerequisite for the emplacement of radioactive waste in the planned repository is that the average temperature rise at the side wall and at the face of the emplacement room (7 m in width and about 6 m in height) does not exceed 3 K.

Based on these marginal conditions, the main part of the safety assessment is to calculate the maximal admissible heat source of the emplaced radioactive waste with the help of the solution of the heat conduction equation taking into account the waste packages, the backfill material and the host rock formation (the heat conductivity of the host rock has been determined experimentally).

The calculations of the thermal influence and resulting requirements on the emplacement of the waste packages can be carried out in different steps which are described in more detail in the following.

The first step of this procedure is the determination of a limited heat production. If a homogeneous distribution of the heat sources along an emplacement room is assumed, a limited heat production can be determined for an individual radionuclide such that the determined temperature rise of 3 K at the side wall of the emplacement room is not exceeded during a certain period of time which is assumed to be 100,000 years. For this purpose, the limited heat production is defined as a maximum permissible heat output of a radionuclide per unit length of the emplacement room.

The temperature variations within the waste packages, in the emplacement rooms and in the host rock result - due to the linearity of the heat conduction equation - as a sum of the temperature increases of each of the radionuclides contained in the waste packages. Therefore, the temperature increase caused by the different radionuclides can be calculated independent from each other. The total maximal temperature increase cannot exceed the sum of the maximal temperature increases of all the radionuclides because the temperature maxima occur at different periods of time.

In case of radionuclides with halflives up to a few years, the thermal influence of emplacement rooms located near to each other can be neglected. Otherwise, 9 parallel emplacement rooms with a length of 1000 m are taken into account in the model calculations.

The temperature field is calculated with the help of an analytical solution of the heat conduction equation using a quasi-stationary approach in the near field of the emplacement rooms. More details of the performed calculations are given in (2, 3).

The length-related initial limiting heat outputs in an emplacement room which result from the calculations described above vary between the values $3.8 \cdot 10^2$ W/m and $7.7 \cdot 10^{-3}$ W/m according to whether the halflives are a few days or $>> 10^5$ years. Typical examples are 67.1 W/m for Co-60, 2.36 W/m for Cs-137, 0.37 W/m for Ra-226 and 0.087 W/m for Th-232.

The next step is the conversion of these limiting heat outputs into limits of activity per unit length in the emplacement room taking into account the energy release per decay.

The length-related activity limits for the most relevant radionuclides (key nuclides) are shown in Table I. These key nuclides stated in Table I are only the 13 first alpha-emitters, the 13 first beta/gamma-emitters and the frequently occurring 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. The activity limits of further alpha- and beta/gamma-emitters are then defined by the limits for the most restrictive of the other alpha-(U-236) and beta/gamma-emitters (Sr-90).

TABLE I

Length-related Limits of Activity for Key Radionuclides and Non-specific further Alpha- and Beta/gamma-Emitters Resulting from the Safety Assessment of the Thermal Influence upon the Host Rock. Data in Bq/m.

Radionuclide/ Radionuclide Group	Length-Related Limit of Activity	Radionuclide/ Radionuclide Group	Length- Related Limit of Activity
Th 232	$1.3 \cdot 10^{11}$	Ra 228	$3.7 \cdot 10^{12}$
U 235	$1.5 \cdot 10^{11}$	Nb 94	$5.0 \cdot 10^{12}$
U 233	$1.8 \cdot 10^{11}$	Pu 238	$8.9 \cdot 10^{12}$
Th 230	$1.9 \cdot 10^{11}$	Pb 210	$1.5 \cdot 10^{13}$
Pa 231	$2.0 \cdot 10^{11}$	Ca 41	$1.7 \cdot 10^{13}$
U 234	$2.6 \cdot 10^{11}$	Ag 108m	$2.6 \cdot 10^{13}$
Cm 248	$3.0 \cdot 10^{11}$	Cl 36	$2.6 \cdot 10^{13}$
Np 237	$3.3 \cdot 10^{11}$	Be 10	$2.6 \cdot 10^{13}$
Cm 247	$3.5 \cdot 10^{11}$	Sn 126	$3.3 \cdot 10^{13}$
Pu 244	$4.7 \cdot 10^{11}$	Rb 87	$3.7 \cdot 10^{13}$
Ra 226	$4.7 \cdot 10^{11}$	Co 60	$5.2 \cdot 10^{13}$
U 238	$5.2 \cdot 10^{11}$	Ar 39	$5.3 \cdot 10^{13}$
Cm 245	$9.0 \cdot 10^{11}$	Cs 137	$8.8 \cdot 10^{13}$
Ac 227	$2.5 \cdot 10^{12}$	Ni 63	$7.5 \cdot 10^{14}$
Am 242m	$3.6 \cdot 10^{12}$	Fe 55	$2.9 \cdot 10^{16}$
further alpha - emitters	$1.2 \cdot 10^{12}$	further beta/gamma - emitters	$6.6 \cdot 10^{13}$

The conversion into activity limits is necessary and reasonable in order to formulate the requirements to be met by the waste packages in a uniform way. Otherwise, there would be requirements on admissible activities (normal operation, incidents), heat output and masses (criticality).

Consequently, activity values per waste package are determined. These values only depend on the geometric dimensions of the waste packages (cf. (4)). The dimensions of the packages determine the maximum number of waste packages which can be stacked in the cross section of an emplacement room and thus the activity introduced per unit length of the emplacement room.

Examples of such package-specific activity values are given in Table II. For example, as it can be seen from Table II, the activity values for waste container type V and type VI differ by a factor of 2. This is due to the fact that only 5 waste container type V can be emplaced in the cross section of an emplacement room but 10 waste container of type VI.

A further step is to derive requirements for radionuclide mixtures in waste packages. These requirements lead to a criterion which is stated in the form of a sum rule:

$$S(B) = \sum_i \frac{A(i)}{G(i,B)} < 1$$

where:

- S — Summation value
- A(i) — Activity of the radionuclide i/the radionuclide group i in a waste package
- G(i, B) — Activity values of the radionuclide i/the radionuclide group i, determined from calculations of heat dissipation for the packaging B.

Examples of these G values are given in Table II.

The summation criterion can be applied in two different ways:

- Only the activity values for the key nuclides and for non-specified alpha- and beta/gamma-emitters are used, or
- the key nuclide activity values and activity values for further 76 specified radionuclides and for non-specific alpha- and beta/gamma-emitters are used.

For this the following rules apply:

- If the activity of a key nuclide or the activity of non-specified further alpha- and beta/gamma-emitters exceeds 1% of the relevant activity value, this activity must be specified and allowed for the application of the summation criterion.
- If the value of 1% is not exceeded, the activity of the key nuclide concerned or the activity of non-specified further alpha- and beta/gamma-emitters must not be specified nor taken into account when the summation criterion is applied.

The general claim of introducing key radionuclides and non-specified further alpha- and beta/gamma-activity is to determine a practicable scope of necessary information on the composition and activity of radionuclides in a waste package.

If $S > 1$, 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 activity values of these radionuclides which are generally less restrictive.

Moreover, it would be reasonable to admit not only a mixture of radionuclides in a waste package, but also the mixture of waste packages in the cross-section of an emplacement room.

As it can be expected that the temperature inside the emplacement rooms will largely equalize within periods of a few years, the heat output of the individual waste packages is assumed to be homogeneously distributed over the cross section of an emplacement room for the respective calculations. It is, therefore, feasible to jointly emplace waste packages with different heat outputs.

As a result of these considerations and the respective calculations, waste packages with summation values above 1 can be emplaced, provided that other waste packages show correspondingly lower summation values. In this case, a volume-weighted averaging of the summation values must be

TABLE II

Examples of Nuclid-specific Activity Values in Bq per Waste Package

Waste package	Th 232	U 235	Co 60
Cylindrical concrete packaging			
type I	$6.8 \cdot 10^{9}$	$7.4 \cdot 10^{9}$	$2.6 \cdot 10^{12}$
type II	$7.4 \cdot 10^{9}$	$8.1 \cdot 10^{9}$	$2.9 \cdot 10^{12}$
Cylindrical cast iron packaging			
type I	$4.3 \cdot 10^{9}$	$4.7 \cdot 10^{9}$	$1.7 \cdot 10^{12}$
type II	$7.4 \cdot 10^{9}$	$8.1 \cdot 10^{9}$	$2.9 \cdot 10^{12}$
type III	$5.8 \cdot 10^{9}$	$6.3 \cdot 10^{9}$	$2.2 \cdot 10^{12}$
Container			
type I	$2.0 \cdot 10^{10}$	$2.2 \cdot 10^{10}$	$7.8 \cdot 10^{12}$
type II	$2.2 \cdot 10^{10}$	$2.4 \cdot 10^{10}$	$8.5 \cdot 10^{12}$
type III	$4.8 \cdot 10^{10}$	$5.3 \cdot 10^{10}$	$1.9 \cdot 10^{13}$
type IV	$4.0 \cdot 10^{10}$	$4.4 \cdot 10^{10}$	$1.6 \cdot 10^{13}$
type V	$5.6 \cdot 10^{10}$	$6.2 \cdot 10^{10}$	$2.2 \cdot 10^{13}$
type VI	$2.8 \cdot 10^{10}$	$3.1 \cdot 10^{10}$	$1.1 \cdot 10^{13}$

carried out. Emplacement is possible if the summation value thus determined is less than one. A further "thermal dilution" in axial direction of the emplacement room is allowed for up to three container lengths. A volume-weighted averaging of the summation values must be carried out in this case, too, and it must be guaranteed that the summation value thus determined is less than one.

Moreover, maximum summation values are determined depending on the type of packaging. As an example, the maximum summation values for the mixed emplacement in a cross-section are given:

- 4 in the case of container type III, IV and V,
- 8 in the case of container type I, II and VI,
- 20 in the case of cylindrical packaging.

In any case, the delivery of waste packages to the repository exceeding the summation value 1 needs the preceding approval of the BfS.

Such a mixed emplacement leads to more flexibility, because in each of these stacking sections different types of packaging could be emplaced. The application of the flexibility provided by the use of such a mixed emplacement is correlated with an appropriate waste packages control program and a well organized emplacement strategy in order to exhaust the radionuclide content being allowed under thermal aspects.

DISPOSAL OF HEAT GENERATING WASTE

In case of the planned Gorleben repository intended to dispose of all kinds of solid and solidified radioactive waste including especially heat generating waste, a very comprehensive safety assessment is necessary.

Thermal and thermomechanical constraints in the near- and the far-field of the repository served as design basis. For spent fuel/heat generating waste such as vitrified fission product solutions and cemented intermediate level waste, maximum temperatures of 200°C and 100°C, respectively, in the waste package-salt interface for the heat generating waste and the spent fuel or in the waste form itself in case of intermediate level waste are not to be exceeded (5). These limitations resulted from various aspects in the case of glass and from the cement properties, respectively.

In the far-field, tensile stresses should be avoided at the top of the salt dome to prevent a possible formation of pathways for water intrusion into the repository.

The knowledge on the material data and the material laws necessary for the thermal calculations and proposals for an appropriate disposal strategy are under progress within a comprehensive R & D program for waste disposal in salt (6).

The results of the thermal calculations combined with the expected waste arisings will serve as the basis for a detailed engineering layout of the repository concept, including a conceptual design of the mine, the machinery, the surface installations, etc.

In accordance to the actual time schedule the safety assessment including the thermal aspects should be completed at the end of this century depending on the results of the current underground exploration of the Gorleben salt dome.

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