Note:
This is a translation of the ESK discussion paper entitled
“Diskussionspapier zur Endlagerung von Wärme entwickelnden radioaktiven Abfällen, abgereichertem Uran aus der Urananreicherung, aus der Schachtanlage Asse II rückzuholenden Abfällen und sonstigen Abfällen, die nicht in das Endlager Konrad eingelagert werden können, an einem Endlagerstandort”.
In case of discrepancies between the English translation and the German original, the original shall prevail.

E S K

DISCUSSION PAPER of the Nuclear Waste Management Commission (ESK)

Discussion paper on the disposal of heat-generating radioactive waste, depleted uranium from uranium enrichment, waste to be retrieved from the Asse II mine and other waste that cannot be emplaced in the Konrad repository at one site

CONTENTS

1 Introduction

According to the procedure set out in the Site Selection Act (StandAG) of 23.07.2013 [1], a site is to be selected in Germany for a disposal facility for in particular heat-generating radioactive waste by 2031. After completion, the Konrad repository will be available for a large part of the radioactive waste generated in Germany with negligible heat generation (303,000m^3), as laid down in the plan approval decision. The other part of the radioactive waste with negligible heat generation, i.e.

- uranium tails from uranium enrichment,
- waste to be retrieved from the Asse II mine, and
- other waste that cannot be emplaced in the Konrad repository[1],

is to be considered in the site selection process according to the StandAG as laid down in the "Programme for the responsible and safe management of spent fuel and radioactive waste" (National Programme, NaPro [2]) of the Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (BMUB).

Depending on the conditions at the site and depending on the disposal concept (in particular the layout of the respective emplacement areas), potential interactions between the heat-generating radioactive waste and the radioactive waste with negligible heat generation is to be taken into account when emplaced at one single site. In principle, cross-influences of the waste types by physical and chemical interactions (e.g. heat input, gas generation, formation of chemical gradients) of different waste components are conceivable. Heat-generating radioactive waste is characterised by a large radionuclide inventory (total activity about 10^{20} Bq [3]) and mainly consists of uranium oxide or borosilicate glass. The radioactive waste to be considered with negligible heat generation has a significantly lower radioactivity inventory (see Table 1). It is partially conditioned in a cement matrix, has a much larger volume and contains a partly complex mixture of different substances (salts, organic materials, decontamination agents, etc.) varying from waste stream to waste stream. This applies, in particular, to the radioactive waste with negligible heat generation that is to be retrieved from the Asse II mine and for which the type of conditioning is still unknown. Of particular relevance for safety considerations are such interaction processes that may lead to increased radionuclide release from heat-generating radioactive waste.

As already stated in the final report of the former Committee on a Site Selection Procedure for Repository Sites (AkEnd) [4], a disposal facility must equally fulfil all requirements resulting from different types of waste. This implies that waste types and quantities must be known for a successful site selection procedure. In addition, the AkEnd emphasises that it is therefore to be expected that the number of sites that are potentially suitable for all types of waste is lower than the number of sites potentially suitable for parts of the waste. Furthermore, the AkEnd calls for the spatial separation of radioactive waste with negligible heat generation from the heat-generating radioactive waste as being indispensable under aspects relating to long-term safety and the safety case.

[1] According to [2], this refers to radioactive waste that owing to its nuclide inventory and/or its chemical composition or the time of its generation is not suitable for emplacement in the Konrad repository.

When developing a concept for a disposal facility in accordance with the StandAG, measures must be taken to exclude or minimise interactions between heat-generating radioactive waste and radioactive waste with negligible heat generation resulting from disposal at one site that are relevant in terms of long-term safety and the safety case. This applies, in particular, to interactions that could lead to increased radionuclide release. This discussion paper presents potential interactions and outlines their relevance for disposal sites in different host rocks.

2 Consultations

At its 49th meeting on 03.09.2015, the ESK started the discussion on the various aspects to be considered for the disposal of the above-mentioned waste at one disposal site. Already at this meeting, the ESK expressed its view that the spectrum of waste to be emplaced and the emplacement concept (joint or separate emplacement) will have an impact on the site selection procedure. The ESK continued its consultations at the 50th ESK meeting on 29.10.2015 and requested an ad hoc working group of the ESK Committee on FINAL DISPOSAL (EL), which had already dealt with a compilation of waste stream characteristics, expected volumes and resulting requirements for a disposal facility, to further develop their consultation results according to the ESK discussion. This ad hoc working group on “other wastes” (SONSTIGE ABFÄLLE) presented the accordingly updated document at the 51st meeting of the ESK on 10.12.2015 as a draft discussion paper. At its 52nd meeting on 11.02.2016, the ESK continued its consultations. Subsequently, an enlarged ad hoc working group SONSTIGE ABFÄLLE revised the draft text again and finalised it at its meeting on 05.04.2016 and by way of circulation. At its 54th meeting on 12.05.2016, the ESK adopted this discussion paper.

3 Safety requirements and safety concept

The “Safety Requirements Governing the Final Disposal of Heat-Generating Radioactive Waste” [5] apply to the disposal facility for in particular heat-generating radioactive waste to be constructed according to the StandAG. Accordingly, the permanent protection of man and the environment must be achieved taking the following safety principles into account:

- The radioactive and other contaminants in the waste must be concentrated and enclosed in the containment-providing rock zone, and thus kept away from the biosphere for as long as possible.

- Disposal must ensure that in the long term, any release of radioactive substances from the disposal facility increases the risks associated with natural radiation exposure only to a very limited extent.

The Safety Requirements [5] explicitly only apply to the disposal of heat-generating radioactive waste. With a view to joint disposal with radioactive waste with negligible heat generation, they stipulate the following: “If, on the basis of alternative considerations, radioactive waste with negligible heat generation is also to be emplaced in this final repository, observation of these Safety Requirements shall be extended to include such waste, with the exception of the requirements applicable to waste containers pursuant to section 8.6.” (Section 8.6 contains requirements as regards the retrievability and handleability of the waste containers in case of recovery). With the exception of this generic statement, no further details are given regarding the emplacement of waste with negligible heat generation. The ESK holds the view that appropriate safety requirements may need to be re-developed.

According to [5], the safe disposal of heat-generating radioactive waste is to be carried out in deep geological formations with a high containment capacity. “The safety of the final repository after decommissioning must therefore be ensured by means of a robust, graduated barrier system that fulfils its functions in a passive, maintenance-free manner and which continues to ensure adequate functionality even if individual barriers fail to develop their full effect.” To implement this safety requirement, safety concepts are developed which take into account the site conditions, in particular the host rock properties, as well as the properties of the radioactive waste to be disposed of (Chapter 4).

4 Current arisings of radioactive waste in Germany destined for disposal

The current situation has changed compared to the conditions under which the final report of the AkEnd [4] was drawn up. According to a decision by the Federal Government, the Konrad repository will be available for about 303,000 m^3 of radioactive waste with negligible heat generation that has been and will be generated in Germany after the shutdown and dismantling of nuclear power plants as well as from research and other sources [2]. About 10,500 t HM of spent fuel and 7,979 canisters with waste from reprocessing are to be emplaced in the disposal facility according to the StandAG [2]. As stated in the NaPro [2], there may also be about 100,000 m^3 of uranium tails from uranium enrichment according to current estimates. This amount corresponds to approximately 30 years of operation of the Urenco plant; it may increase in case of an extended period of operation. Furthermore, there may be an additional 175,000 to 220,000 m^3 from the Asse II mine (radioactive waste with negligible heat generation as well as contaminated materials). In addition, there are waste types which, from today's perspective, cannot be emplaced in the Konrad repository. Their amount is difficult to quantify and there are currently only estimates for upper limits [6] (Table 1).

The composition of the different waste types that are to be taken into account, as stated in the NaPro [2], for the disposal facility according to the StandAG varies considerably (Table 1). The level of knowledge with regard to the characterisation of the three waste streams is also very different. While the composition of the heat-generating radioactive waste and the uranium tails from uranium enrichment is very well known, the physicochemical characterisation of the waste to be retrieved from the Asse II mine can only be done after retrieval and will also depend on the conditioning concepts still to be defined. The existing knowledge of the composition is based on the emplacement documentation. The other waste types also to be considered here that cannot be emplaced in the Konrad repository originate from various sources and are very different in their chemical composition and have not been further characterised yet.

Table 1: Amount, components and characteristics of different radioactive waste types intended for emplacement at the site of the disposal facility according to the StandAG [2, 6, 7-10].

<table>
<tr><th></th><th>Waste type</th><th>Composition
(main components)</th><th>Potential impacts on the environment of the emplacement area to be considered</th></tr>
<tr><td rowspan="2">Heat-generating waste
(>99% of the total radionuclide inventory , approx. 10^{20} Bq [12])</td><td>Spent fuel [2]
(material composition: mainly UO_2, MOX material, iron containers)</td><td>- 10,500 t HM from power reactors; ceramic UO_2/MOX material
- 10-12 t HM from research reactors; graphite pebbles, USi, Al …
- steel/cast iron containers or containers made of other metals</td><td>- heat input
- hydrogen production (from container corrosion)</td></tr>
<tr><td>Waste from reprocessing [2]
(material composition: borosilicate glass, Zircaloy cladding tubes, other metals, iron containers)</td><td>- borosilicate glass (3,875 canisters)
- compacted metal-containing, technological waste (4,104 canisters)
- steel/cast iron containers or containers made of other metals</td><td>- heat input
- hydrogen production from container corrosion</td></tr>
<tr><td rowspan="3">Waste with negligible heat generation
(< 1% of the radionuclide inventory)</td><td>Waste from the Asse II mine [7-10]
(complex material composition)</td><td>A total of approx. 175,000 to 220,000 m³, including:
- approx. 12,000 t cement stone and 45,000 t concrete
- approx. 19,000 t iron/steel
- approx. 9,500 t organic compounds [10] (including 40 t complexing agents EDTA, citrate, oxalate, …)
- approx. 900 t nitrate
- approx. 50,000 m^3 contaminated salt [10]
- 102 t U, 87 t Th, 28.9 kg Pu [10];</td><td>- CO_2 production (from microbiological decomposition of organic components)
- high-pH plume from cement corrosion
- introduction of soluble complexing agents
- introduction of soluble salts
- hydrogen production from metal corrosion</td></tr>
<tr><td>Waste from uranium enrichment
(material composition: mainly U_3O_8)</td><td>- approx. 100,000 m^3 uranium (in the form of U_3O_8) [2]
- approx. 2,240 t UO_2F_2 [11, 12]
- unknown amount of hydrofluoric acid (HF) [11, 12]</td><td>- introduction of fluorides and hydrofluoric acid
- hydrogen production from container corrosion</td></tr>
<tr><td>Other waste that cannot be emplaced in the Konrad repository [6, 11, 12]
(complex material composition)</td><td>- graphite-containing waste net max. 500 m^3 [6]: (H-3, C-14 …); approx. 1,540 t coal stone; approx. 660 t graphite [11, 12]
- other “mixed” waste: net waste volume max. 5,000 m^3 [6]
– C-14-containing radioactive waste,
– H-3-containing waste,
– H-3 in beryllium-containing materials,
– thorium and paraffinic waste</td><td>- high-pH plume from cement corrosion
- any entry or production of other substances not known until now
- hydrogen production from container corrosion</td></tr>
</table>

5 Specifics of different host rocks

In the following, central safety functions of typical concepts for deep geological disposal of high-level radioactive (heat-generating) waste in salt, clay and crystalline formations are presented whose effectiveness and robustness would have to be checked in the case of joint disposal with other waste types.

Emplacement in *rock salt* [13] is primarily aimed at preventing the inflow of solutions, i.e. potential transport media for radio- and chemotoxic substances, to the emplaced waste. This is to be achieved by means of low-permeable barriers: the emplacement cavities are to be excavated in undisturbed, negligibly permeable rock salt parts. Potential pathways via the excavated access drifts and shafts are to be sealed by geotechnical barriers. In the first phase after emplacement (decades to a few centuries), these are the drift and shaft seals. It is to be ensured that their permeability (including contact and excavation-damaged zones) remains sufficiently low to prevent solution inflow. The function of preventing the inflow of solutions is taken over by the crushed salt backfill in the cavities as soon as it has sufficiently low permeabilities due to the convergence resulting from overburden pressure.

Should there be a release of contaminants from the waste packages and subsequent migration – possible carrier media would be solutions entered despite the above-mentioned measures, liquids introduced with the waste or gases formed – efforts must be made that as few contaminants as possible are dissolved and their migration is chemically retarded so that migration in shafts and access drifts only takes place to a negligible extent.

In view of the above, it is to be examined whether a pressure build-up due to gases additionally formed by the radioactive waste with negligible heat generation can jeopardise barrier integrity (i.e. the maintenance of low permeabilities) or influence the fluid movement or whether these gases could act as a carrier medium for contaminants. Furthermore, it is to be examined whether negative chemical conditions can occur that increase solubility or promote complex formation and thus promote the release from the packages and whether there may be negative effects with regard to the durability of the container materials and geotechnical barriers or with regard to retention mechanisms. High contents of salt, as can be expected in the waste to be retrieved from the Asse mine, will be compatible with the host rock in case of emplacement in a disposal facility in rock salt. Negative effects on geotechnical or geological barriers are not to be expected here, while this cannot be ruled out for other host rocks and therefore has to be examined.

Regarding emplacement in *clay*, containment of the heat-generating radioactive waste in the containers is assumed for several thousand years. This requirement is therefore the main determining factor for the design of the containers. Later, when no credit can be taken from enclosure through the containers, liquids and gases can be considered as carrier media for released contaminants. The movement of liquids should be hindered such that the transport by advection is negligible compared to diffusive transport. This is to be achieved by the low permeability of the host rock and the geotechnical barriers (seals, swelling bentonite backfill – taking into account contact and excavation-damaged zones). Here, too, efforts are to be made to ensure that as few contaminants as possible are dissolved. Chemical retention mechanisms in the host rock and the geotechnical barriers are to retard migration by diffusion. Small pore sizes will contribute to filtering out any contaminant-bearing colloids.

Accordingly it is also necessary for emplacement in clay to examine whether the gases additionally formed by radioactive waste with negligible heat generation can contribute to a pressure build-up and thus to mechanical barrier damage or whether they can act as a driving force for fluid movement or as a carrier medium for contaminant migration. The effects of changes in the chemical composition of solutions on the materials used in the geotechnical barriers as well as the behaviour in terms of solubility, complex formation and sorption are also to be examined.

Regarding emplacement in the usually fractured *crystalline rock*, the Swedish-Finnish KBS-3 concept, for instance, is based on the assumption that the heat-generating radioactive waste is contained in canisters encapsulated in copper for several hundred thousand years. The canisters are surrounded by buffers of swelling bentonite that are to ensure mechanical stability and to prevent attacks by corrosive groundwater. The selection of a favourable host rock and an appropriate design of the geotechnical sealing system are to prevent or hinder hydraulically and hydrogeochemically induced damage to the buffer (bentonite erosion) and the transport of corrosive substances (e.g. sulphides) to the canister-buffer system. In case of container damage, efforts are to be made, as in the above-mentioned concepts, that as little contaminants as possible are dissolved. In this case, the buffers are intended to largely prevent advective transport of contaminants and to counteract diffusive transport by chemical retardation as well as to filter out colloids. Further retardation can occur in the host rock by sorption in the fracture fillings and by matrix diffusion.

For emplacement in crystalline rock, the impacts of gases formed by radioactive waste with negligible heat generation on the flow field are to be examined particularly. Corrosion of the copper-encapsulated canisters provided for the KBS-3 concept and the related production of large hydrogen volumes can at least be regarded as negligible for the probable development of the disposal system. In addition, the question arises as to the impacts of changed hydrogeochemical conditions on the erosion and swelling behaviour of bentonite, on copper corrosion, on radionuclide mobilisation, and on the retention properties of bentonite and the host rock.

Emplacement of different waste streams in different e.g. superimposed host rock formations at the same site is also conceivable. Such concepts could possibly be applied if the host rock suitable for heat-generating radioactive waste is limited in its extent. Making use of different host rock properties for different waste streams is also conceivable. Thus, for example, pore volumes could be used to absorb the gases formed in the case of waste generating larger amounts of gases, and waste generating less gas could be emplaced in denser host rock formations.

Due to its limited radionuclide inventory, other host rock formations may be considered for radioactive waste with negligible heat generation than for the disposal of heat-generating waste. Under certain circumstances, better separation of waste types from one another can be achieved this way. For example, only the Opalinus Clay was proposed in several possible site regions in the Swiss Sectoral Plan for Deep Geological Repositories for the disposal of high-level radioactive waste (HLW), while for low- and intermediate-level waste (LILW), other host rock formations were recommended in addition to Opalinus Clay, in some cases in the same site regions [14].

6 Analysis of influencing parameters and potential interactions

Safety considerations for a disposal concept which provides for the emplacement of both heat-generating waste and waste with negligible heat generation require first of all the analysis of influencing parameters and processes that may lead to interactions of different waste components and, in particular, to additional radionuclide release, and may thus impair long-term safety. In addition, influences on the complexity of the overall system have to be considered which may affect the quality of safety analyses. The safety consideration for such a disposal concept must, in particular, take into account the influences of radioactive waste with negligible heat generation on heat-generating radioactive waste. Furthermore, the type of waste conditioning has an influence on the chemical reactivity of the waste. Table 2 shows that the potential mutual influences of individual types of waste can be very different. It also shows that potential interactions, mainly between heat-generating waste and waste components of “other waste with negligible heat generation” (see footnote in Table 2) are to be expected. In particular, such processes need to be quantified or excluded.

Some of the influencing parameters and interactions to be analysed are outlined below. In general, the existing data situation is limited for many processes described and must be newly developed to a considerable extent.

Table 2: Potential influences to be investigated regarding the emplacement of waste types on other waste streams

Potential impacts on / *when emplacing*	**heat-generating radioactive waste**	**uranium tails**	**other waste with negligible heat generation***
heat-generating radioactive waste		- physical/chemical effects of increased temperatures - 2-phase flow induced by H_2 gas generation (container corrosion) and possible accelerated transport of contaminants	physical/chemical effects of increased temperatures on: - microbial degradation of organic waste components (formation of complexing agents, CO_2, CH_4) leading to increased radionuclide solubility/mobility - metal corrosion (H_2 generation) - 2-phase flow induced by H_2 gas generation (container corrosion) and possible accelerated transport of contaminants
uranium tails	- introduction of fluorides and hydrofluoric acid influencing radionuclide solubilities/mobilities - retention of radionuclides at uranium tails [15]		- retention of radionuclides at uranium tails [15]
other waste with negligible heat generation*	- entry of saline solutions: influence on bentonite barriers in a disposal facility in clay or crystalline rock - introduction of organic complexing agents, CO_2: influence on radionuclide solubility/mobility - high-pH plume: influence on geological/geotechnical barriers and on radionuclide solubilities/mobilities	- entry of saline solutions: influence on bentonite barriers in a disposal facility in clay or crystalline - introduction of organic complexing agents, CO_2: influence on radionuclide solubility/mobility - high-pH plume: influence on geological/geotechnical barriers and on radionuclide solubilities/mobilities	

* “other waste with negligible heat generation” refers to waste described in Table 1 from the Asse II mine and other waste that cannot be emplaced in the Konrad repository.

Temperature

The heat input of a disposal facility for heat-generating radioactive waste into the environment will lead to a temperature increase in neighbouring emplacement areas for radioactive waste with negligible heat generation. The extent to which temperatures will rise depends on the host rock, on the emplacement concept, and on the planned distances between the emplacement areas. Previous safety analyses for disposal facilities for radioactive waste with negligible heat generation do not assume temperatures significantly higher than the rock temperature. Reactions, such as the decomposition of organic components under possibly microbial influence and in the presence of oxidants (nitrate), corrosion of waste matrices and thus release of radionuclides, can be considerably accelerated at elevated temperatures. The data available which allow drawing reliable conclusions on the effects of elevated temperatures on the processes occurring in a disposal facility with heterogeneous waste composition are limited.

Gas generation

Regarding gas generation, the following three factors are to be considered, which can be of different relevance for the operating and the post-closure phase:

In connection with the disposal of heat-generating radioactive waste, metal corrosion in the case of solution inflow into the near field in the post-closure phase is regarded as an essential source for the generation of hydrogen. Regarding the disposal of radioactive waste with negligible heat generation, corrosion processes can take place due to residual water content in the waste also without solution inflow. Increased hydrogen partial pressures lead to reducing conditions and can lead to a significant reduction in fuel corrosion [16]. The increased gas pressure can also affect the functionality of geotechnical barriers.

In the case of a large inventory of organic material in a disposal facility for radioactive waste with negligible heat generation (e.g. resins, oil, plastic films), scenarios are to be analysed that involve the generation of considerable amounts of gases in the form of carbon dioxide (CO_2), methane (CH_4) and hydrogen (H_2). The effects on the function of geotechnical barriers are also to be examined for these cases. In addition, the introduction of CO_2 in a disposal facility with a high radionuclide inventory can lead to a considerable increase in the solubility and thus the mobilisation of actinides by carbonate complexation [17]. The relevance of such scenarios is to be assessed.

Depending on the composition of the waste, these may contain relevant amounts of radioactive, volatile or gas-bound radionuclides tritium (H-3), carbon (C-14), iodine (I-129) and radon. These are to be considered for the operating phase and may result in special requirements for the construction, ventilation and exhaust air ducting as well as for the organisation of the operational processes in order to ensure operational safety and radiation protection.

Organic components

Wastes in the Asse II mine contain relevant amounts of organic material, which partly consists of complexing agents (EDTA, citrate, oxalate) or components, e.g. cellulose, which can be degraded to complexing agents such as isosaccharinic acid (ISA). Furthermore, the generation of considerable amounts of CO_2 (see above) is to be expected and thus also an increase in concentrations of complexing carbonate in solutions by microbial degradation of organics. It is to be examined to what extent complex formation could lead to a considerable increase in the solubility of relevant radiotoxic waste components if such complexing agents enter emplacement areas for heat-generating radioactive waste by, for example, diffusive transport. While the influence of complexing agents on radionuclide mobility in a disposal facility with cemented radioactive waste with negligible heat generation is limited due to retention on cement phases and the resulting corrosion products/secondary phases, this is not necessarily the case in a disposal facility for spent fuel and vitrified high-level radioactive waste.

Influence of salts/acid

High soluble salts, e.g. in waste from the Asse II mine, can spread by diffusion or advective transport. The swelling properties of bentonites, as used in a disposal facility in crystalline and in clay host rocks as geotechnical barriers, decrease with increasing ionic strength. The influence of saline solutions on the potentially selected host rocks crystalline and clay is also to be examined. To date, only few data exist in order to estimate the effects on long-term stability and radionuclide retention under such conditions.

The same applies to the remaining hydrofluoric acid and the amount of fluoride that remain in the uranium tails from uranium enrichment after conditioning. Their transport, their geochemical reactions and their potential influence on the corrosion of fuel and high-level radioactive waste from reprocessing as well as on radionuclide solubilities by complex formation [18] are to be quantified.

High-pH plumes

Due to corrosion of the considerable amounts of cement/concrete contained in waste from the Asse II mine and other radioactive waste with negligible heat generation which cannot be emplaced in the Konrad repository, a high-pH plume can develop in the environment of the emplacement areas that may have an influence on barrier corrosion and radionuclide solubilities in the disposal facility for heat-generating radioactive waste. High pH values in combination with high calcium concentrations may lead to a significant increase in the solubility of, in particular, the tetravalent actinides, which are otherwise considered to be immobile, such as e.g. plutonium [19]. The spatial distribution of pH gradients in a disposal facility in clay will be limited due to extremely slow diffusive transport processes (order of magnitude metres or less), but in fractured rock with advective transport it will be significantly larger. Further effects relate to secondary phase formation causing porosity changes in geotechnical and geological barriers.

7 Potential measures

In the following, possible technical measures are discussed that can contribute to minimising or even preventing the above-mentioned interactions.

Conditioning of the waste

The conditioning (treatment) of the waste can have a significant influence on the generation of gas and the release of radionuclides. Relevant parameters are, in particular, the stability and corrosion resistance of the waste containers, the fixation of the radionuclides within the waste matrix, as well as the water content and the content of organics in the waste. The requirements for conditioning also determine the specific volume of a waste stream, the efforts and the related costs. A detailed concept for the conditioning of the waste can thus only be defined in the context of a disposal concept.

The packaging of the waste with negligible heat generation currently provided, e.g. with steel containers, will not prevent the release of waste components and their possible interaction with heat-generating waste in the long term. When water enters the emplacement area, the barrier function of e.g. thin-walled steel containers over long periods of time is only of limited duration so that no credit is taken from it in terms of long-term safety. Particularly in case of contact with saline solutions, almost complete corrosion can be expected within a few decades and the release of waste components.

However, by separation of individual waste components during conditioning, certain interaction processes can possibly be ruled out. Thus, the separation of salt from the waste retrieved from the Asse II mine is principally conceivable due to its good water solubility. In this case, a further treatment step would have to be carried out in order to also selectively separate highly soluble radionuclides from the saline solutions. For such radionuclides, such as e.g. Cs-137, Sr-90 and I-129, selective ion exchange and solvent extraction methods are described in literature, but their applicability to solutions with high salt concentrations is to be checked. The selective separation of radionuclides such as H-3 or C-14 would be much more complex. In view of the composition of the largely cemented Asse waste, corresponding influences by cement components as well as by other waste components on the respective separation steps would have to be investigated. Established methods for this specific issue are currently not available and would have to be developed. With regard to the estimated volumes of 175,000 to 220,000 m^3 of conditioned waste with approx. 50,000 m^3 of contaminated salt [20], which is to be retrieved from the Asse II mine, one will have to consider appropriately dimensioned conditioning plants which allow the treatment of these amounts of waste and some 100,000 m^3 of brine. The required plants would exceed the capacities of today’s conditioning plants by far.

Organic components can be eliminated by thermal treatment of the waste. Possibly suitable methods are e.g. calcination, pyrolysis or the application of plasma processes for combustible, bituminised and also cemented waste [21]. During thermal treatment, waste can also be incorporated into matrices suitable for disposal, such as ceramics or borosilicate glass. However, whether and which methods are suitable for the treatment of waste from the Asse II mine as well as other waste that cannot be emplaced in the Konrad repository would first have to be examined.

Separation of emplacement areas (influences from the other mine)

The term disposal “at one site” is not further specified. In principle, all variants are conceivable from the use of common emplacement areas for different types of waste up to the determination of a spacious site above ground from which two completely separated underground facilities are mined. If mutual influences of different waste streams are completely to be excluded, the maximum range would have to be determined for each of the above-described (and possibly other) safety-relevant parameters from which the required distance between the emplacement areas can be derived. If, depending on the results of the interaction analyses, a concept is pursued which provides for only one repository mine for different waste streams, different emplacement areas can be separated from one another, e.g. by barriers and an adapted ventilation design, so as to minimise or even exclude mutual influences. Strict separation of waste types would be the establishment of two neighbouring but completely separated disposal facilities with separate access shafts at one site. Depending on the variant of separation, there would be different requirements for the disposal site and corresponding consequences for the site selection procedure.

8 Influences on the operation of the disposal facility

Regarding the operation of the disposal facility, the following three steps can basically be distinguished: (1) handling of the waste packages above ground in preparation for disposal, (2) transport of the waste packages to underground (e.g. via a shaft) and (3) handling of the waste packages underground for the purpose of emplacement in the intended disposal area. Thus, three different ways of waste management in terms of handling and transportation are conceivable when operating a disposal facility for different waste streams at one site:

1 separation of waste stream management only in the above-ground facility,
2 separation of waste stream management above ground and separated shaft transport,
3 complete separation of waste stream management.

With Options 1 and 2, separate facilities are available for above-ground handling of heat-generating waste and waste with negligible heat generation. While in Option 1, the same shaft hoisting system is provided for the transport to underground for all waste streams, Option 2 provides for two separate shafts.

Due to the different properties and dimensions of the packages, the complexity of emplacement operation increases when using common handling facilities for all waste streams. For Option 1, this applies to the shaft hoisting system, where a frequent change of the stop and locking devices would be necessary for the various types of waste. These retrofit measures are time-consuming, have to be quality-assured and thus place increased demands regarding staff qualification. The required technical equipment for emplacement (transport vehicles, emplacement devices, etc.) becomes considerably more complex in Options 1 and 2 due to the variety of container types.

In case of complete separation of waste management by the construction of two mines with their own above-ground facilities (at one site), separate shaft hoisting systems and separate emplacement areas, which are possibly separated from each other by a sufficiently dimensioned host rock safety pillar (Option 3), both the technical equipment and the work processes can specifically be adapted to the respective waste streams and optimised. Here, frequent retrofitting regarding the handling procedures for waste emplacement would not be necessary. Complete separation also allows independent planning and implementation of decommissioning and closure of the disposal facilities for different waste types at different times. The same applies to possibly necessary retrieval measures, since this option would not lead to any impairment of safety of the respective unaffected mine.

9 Assessment and conclusions

A quantitative assessment of potential interactions and effects on the operational and long-term safety of a disposal facility is only possible within the framework of a safety analysis for a specific site with a corresponding host rock and disposal concept and can therefore at present only be carried out to a limited extent. This is due to the currently still incomplete set of required data as well as the lack of conceptual specifications. Thus, in the following, an attempt is made to make a qualitative assessment of the effects, influencing parameters and measures described in the preceding chapters.

The joint disposal of heat-generating radioactive waste, depleted uranium from uranium enrichment, waste to be retrieved from the Asse II mine and other waste that cannot be emplaced in the Konrad repository in one disposal facility will inevitably increase the complexity of the disposal system. This has an impact on the safety analysis to be performed. In particular, this will increase the uncertainties in the statements, which ultimately leads to a reduction in the robustness of the safety case. Thus, the joint disposal of various waste types in one disposal facility is generally not advantageous in terms of safety and with regard to the proof of safety. These deficits can only be excluded or completely circumvented by a complete isolation of the various types of waste from one another in separate emplacement areas. Here, isolation is primarily understood as an exclusion of chemical and fluid-dynamic interactions between different types of waste. Safety-enhancing synergies resulting from the joint disposal of the various waste types are not to be expected or only in exceptional cases.

Potential chemical interactions that lead to increased mobility of radionuclides, in particular of long-lived α-emitters, are highly relevant for long-term safety and should be excluded or at least largely prevented by technical measures (conditioning, additional barriers). This relates to (organic) complexing agents, carbonate formation by degradation of organic materials and, with limitations, the evolution of high pH plumes, and applies to all three host rocks crystalline, clay and salt when considering solution access scenarios. High pH values can increase radionuclide mobility only under special conditions. In general, they are rather mobility-reducing but can alter the properties of the geological and geotechnical barriers. However, studies on the effects of high-pH plumes in repository systems in clay rock show their comparatively small spatial extension. The potential occurrence of highly saline fluids is also safety-relevant when considering disposal in crystalline and clay host rocks. Here, the possibly negative effect on the effectiveness of the barrier system is of particular significance (direct chemical influence on the mobility of radionuclides is of secondary importance). These potential interactions must also be prevented or excluded through technical measures.

Rather less safety-relevant impacts are to be expected by temperature increase and gas generation, which are additionally caused by the joint disposal of heat-generating waste and waste with negligible heat generation.

With regard to the various types of waste, relevant impacts on long-term safety are primarily to be expected from the disposal of waste to be retrieved from the Asse II mine and other waste that cannot be emplaced in the Konrad repository together with heat-generating waste in one disposal facility. In this context, safety-relevant effects and interactions must be considered, quantified and evaluated. The joint disposal of heat-generating radioactive waste with depleted uranium from uranium enrichment appears to be far less problematic. In this case, only the extent to which the fluoride and hydrofluoric acid contents in the depleted uranium influence the stability of the spent fuel or vitrified waste and the mobility of the radionuclides would have to be assessed. According to the current state of knowledge, major impacts are not to be expected.

Also as regards the operation of a disposal facility for the joint disposal of waste with negligible heat generation and heat-generating waste, an increase in complexity is to be expected. The possibly increased risk of malfunctions, incidents and accidents, which can lead to interruptions of disposal operation, could be minimised by the complete separation of waste stream management via construction of two mines with their own above-ground facilities (at one site), separate shaft hoisting systems and separate emplacement areas. The possibility of independent planning and implementation of closure, decommissioning and possibly retrieval operation represents further advantages of separate disposal facilities for heat-generating waste and waste with negligible heat generation. Overall, this variant would offer significantly more flexibility and safety-related robustness of operation.

In order to avoid the above-mentioned negative effects of components of waste with negligible heat generation on heat-generating waste by specific conditioning, the separation of highly soluble salts as well as organic waste component degradation by thermal treatment could be considered. Both types of conditioning are in principle technically feasible. However, existing procedures would have to be checked for their applicability to the complex material composition of the waste to be retrieved from the Asse and other waste that cannot be emplaced in the Konrad repository. Adjustments, optimisations and new developments may be required. New concepts and the construction of appropriate facilities will particularly be required in case of separation of salt from waste streams and subsequent conditioning of saline solutions.

From all those considerations discussed above, which take into account aspects of long-term safety as well as of operational safety for a repository for waste with negligible heat generation and heat-generating waste, a concept which provides for fluid-dynamically decoupled and separated emplacement areas is deemed to be the most advantageous. This means that two emplacement areas would have to be planned at one site, which may also be located in two different host rock formations. The possibility to build completely separated disposal facilities at one site has also been discussed by the AkEnd and in Switzerland as a so-called combined repository. Type and degree of decoupling can vary according to disposal concept, host rock and technical realisation and it may be necessary to develop new optimised backfilling and closure concepts. Within the framework of a site selection procedure, corresponding concepts and ideas are to be developed for the potential host rocks and disposal concepts, and the effectiveness of decoupling is to be demonstrated in a preliminary safety analysis. To this end, it is also to be assessed how and to which extent the propagation of heat and gases as well as of chemical gradients in the environment of a disposal facility takes place and what impacts are to be expected. This will be a determining issue for the design of an optimised disposal concept. It is evident that such considerations are necessary at an early stage of site selection. The options for a site for the disposal of heat-generating waste are likely to be additionally limited when agreeing upon joint disposal.

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