

LOCALIZATION AND EVOLUTION SCENARIO ANALYSIS FOR RW DEEP DISPOSAL FACILITY AT THE ENISEISKIY SITE (KRASNOYARSK REGION)

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The paper provides a feasibility study of certain design solutions with relevant possible corrections concerning the design of the disposal facility and engineered barrier materials aiming to enhance the safety of the final disposal facility and to reduce the cost of its construction and operation.

Keywords: *isolation of radioactive waste, geological structure, excavation, underground water, engineered barrier, degradation of materials, migration of radionuclides, radiation safety.*

In 2016 FSUE NO RAO got a positive expert statement from the Federal Agency on Subsoil Uses saying that Eniseiskiy site can be considered as a potentially suitable site for the construction of a final isolation facility for immobilized radioactive waste (RW). It was also indicated there that the distance between the emplacement horizons shall account for 75 m. The Government of the Russian Federation subsequently issued a permit on the use of site subsoils for RW disposal purposes [1]. Construction of an underground research facility (URL) was specified as the initial stage of facility construction to be started in 2019. Main permits concerning these activities have been already issued to FSUE NO RAO in 2016: Design Documentation for URL Construction was approved by Federal independent expert agency Glavgosekspertiza. Relevant license on siting and construction of the URL was issued by Rostekhnadzor [1]. Of course, there's a long way to go from a URL to a deep disposal facility (DDF) and many design solutions should be elaborated and altered on this way. However, it is supposed that the main concept discussing DDF layout will remain unchanged as no principle changes were introduced to it during pre-design investigations [2–4],

but only if no significant reasons for this are detected. Some alternative design options differing by the scope of changes to be introduced but aiming to enhance the designs as regards DDF safety anyway are discussed below. Expert analysis method was used to enable the feasibility study of design solutions proposed and to introduce relevant changes. This analysis covered the disposal facility layout, the bedrocks and engineered barrier materials considered in the existing designs. The study was based on the knowledge gained to date in this field.

Facility localization

Current designs suggest [4, 5] that the DDF will be constructed according to the concept presented in figures 1a and 1b. According to these designs, horizontal excavations will be used for the emplacement of RW class 2 involving long-lived intermediate-level waste (ILW) and high-level waste with low heat output. This waste should be immobilized in a cement matrix or vitrified and packaged into metal NKM containers or some containers of similar designs. These excavations are located at two horizons: +5 and –70 m (Baltic height system). This

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Disposal of RW

level is located at least 120 m lower than regional drainage basis — the Enisei river bed. The excavations are located in parallels with a small relative displacement. Vertical boreholes of large diameter (1.3 m) will be used to accommodate vitrified heat-generating HLW (RW of class 1) drilled at constant intervals between upper and lower level boreholes. To accommodate the excavations in the bedrocks enabling to emplace the amount of RW that was specified in the designs (4,500 m³ of RW class 1 and 155,000 m³ of RW class 2), a homogeneous and tectonically undisturbed unit composed of low-permeable rocks is required having a minimum size of 700×350×100 m.

Studies of the Eniseisky site geological structure have been actively and deliberately performed since 2008 when it was finally selected for further investigations performed under DDF development project [2]. Preliminary investigation results based on which this decision had been made were presented

in [6]. To date integrated surface-based geophysical investigations were performed. Geological and hydrogeological studies covering a depth of up to 700 m were also performed — 20 investigation boreholes were engaged in these studies enabling to perform comprehensive core sampling, geophysical investigations, ground water inflow and laboratory tests [1]. It's important to note that the knowledge gained during the last 60 years involving the construction of way bigger underground excavations at FSUE MCC (up to 100×20×60 m) may be applied during the Eniseiskiy site characterization. These excavations are located in a quite similar bedrock unit of 1500×1,000×250 m located 4 km away to the north of the Eniseiskiy site.

Currently available data [5–7] suggests that the site has a quite complex geological structure (figure 1c). It's considered hardly possible to indicate a single unit of the required size enabling to accommodate all excavations and boreholes for RW

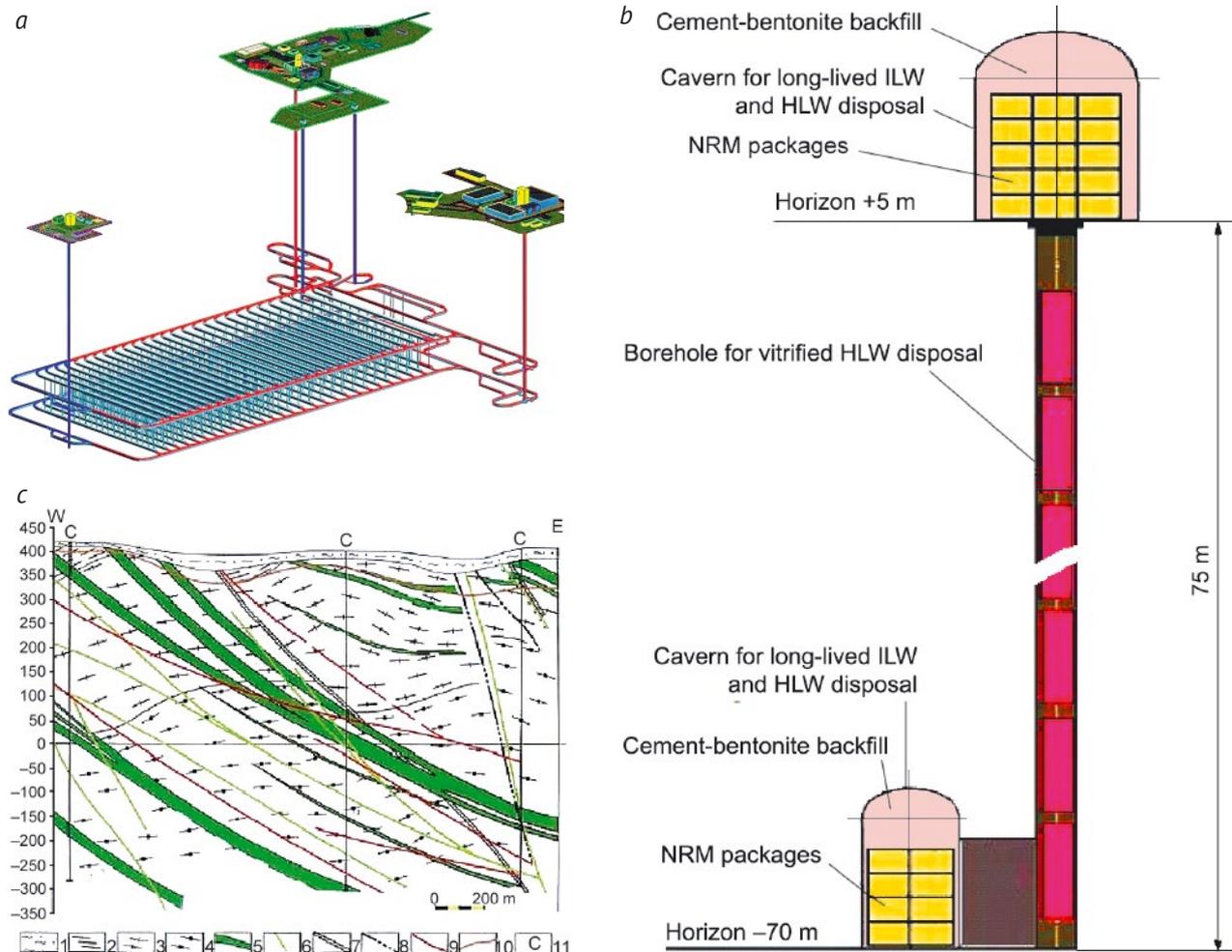


Figure 1. DDF layout according to [4, 5]

a — underground excavations, *b* — disposal caverns, casks and packages with ILW and HLW, *c* — sub-lateral geological cross section of the Eniseiskiy site [7] (simplified): 1 — quaternary loose sediments, 2 — biotite and mica gneisses and crystalline schists, 3 — biotite plagiogneisses, 4 — biotite-cordierite gneisses, 5 — dikes of metabasites of the first stage of intrusion, 6 — dikes of dolerite and diabase porphyrites of the second stage of intrusion, 7 — area early cemented breccias, 8 — faults with open fracturing, 9 — zones of tectonic breccias with quartz-microcline cement, 10 — lower limit of exogenous fracturing, 11 — exploration wells

emplacement being not disturbed by post-metamorphical dykes of stage 2, shear zones, with no mylonitization and open fracturing, moreover strictly complying with absolute height requirements. This is hardly possible even if Archean gneiss and metabasis associated with the first stage of intrusion are considered. Due to these concerns, DDF layout attached to monolith unit with sparsely distributed boreholes as it is at the moment may not pass the review with closer investigation borehole allocation. For this reason, the first DDF siting proposal suggests that relevant opportunities should be provided to divide one big unit into smaller sections that should be matched with monolith rock mass elements precisely enough characterized during investigations even though these might be of a smaller size. The number and the size of such sections may be definitely identified following detailed geological investigations. To increase the number of potentially suitable sections it would be worth to enlarge the range of their allocation in terms of the absolute height that had been reduced unreasonably only due to the concept selected and to increase the number of horizons.

Obviously the excavation of contiguous boreholes extending along long parallel profiles according to the so-called matrix disposal layout is associated with reduced cost as compared to an option when the same disposal volume is excavated for scattered boreholes suggesting longer transportation tunnels having no functional purpose at all. But in which way the economic feasibility of the project can be affected in case if part of already excavated boreholes turns out to be unsuitable for waste disposal? It's likely that evaluation of advance drilling results would be enough to reveal this with no need for full excavation of such boreholes. It's considered feasible to allow for some prompt decisions to be made on halting design development activities at an intermediate stage if adequate reasons for this are found. This will enable effective use of available resources. Another thing to do is to provide for the possibility of introducing some alterations into designs and if necessary the implementation of an alternative option. This approach will obviously result in lower costs compared to an option when already excavated and unsuitable for waste disposal boreholes are to be backfilled and sealed. Moreover this option also requires further searches and excavation of new boreholes to enable the disposal capacity specified in the design.

Even if a unit being adequate in its size and not disturbed by tectonic processes occurred to date enabling to accommodate all boreholes specified in the designs is found, excavation of such a big number of long, closely located boreholes of large diameter with a dense grid of parallel profiles will result in significant reduction of the unit's strength. If stresses occur in the rocks, even local ones, associated, for example, with the thermal field of the facility, these profiles may act as perforation

channels which may result in fracturing and rock displacement along profile surface.

In this case not only high inflow areas may be generated, but with high probability mechanical impacts produced directly on engineered barriers and RW packages may occur resulting both in deformation and destruction of these structures.

The second question as regards facility siting is associated with its orientation in terms of geological structures considered. Current designs suggest its meridional orientation with transportation tunnels running from north to south and waste emplacement (operational) — from west to east. Main geological structures at the site (complex of metamorphic rocks, involving metabasite, dykes of the second intrusion stage, shear zones) excluding some newly formed fractures have sub meridional strike and are declining to the east. For this reason, sub-lateral subsection (figure 1c) maximum dipping angles are shown being however lower than 45°. Considering the abovementioned orientation of geological structures, the probability of their crossing by excavations is minimal if these are excavated horizontally in meridional direction. Subsequently, this direction is viewed as an optimal one for operational excavations. This statement is also true for the decision making regarding the orientation of excavations (boreholes) designed for HLW heat-generating waste. Bearing this in mind, their horizontal orientation is viewed as a preferred design solution as compared to the vertical one. Layouts of DDF with horizontal emplacement boreholes for spent nuclear fuel (SNF) and RW disposal have been considered abroad not only in the context of crystalline rocks, but also in case of sedimentary ones [8]. This option is viewed as an important alternative for the existing DDF concept, however, its safety and economic feasibility haven't been evaluated to date which can not be considered as a graded approach. Below, this alternative is discussed in more detail, namely, in terms of RW package robustness during the DDF evolution.

The last remark concerning space orientation of the DDF: if two matrixes are considered with same open porosity and pore geometry, the one with a bigger number of one-side open pores will have lower permeability. For this reason, the layout suggesting that all excavations are joined up at both ends (are open) enables higher ground water flow as compared to tree-type structure when boreholes spur off from one trunk and have a dead end. This system is very similar to a river system with all its rises coming from nowhere and running into one mainstream.

This river bed may be co-oriented with river estuary in a safest way or even provide its deepening while operational caverns may be uplifted to higher horizons enabling reduced costs of excavation and operation. The main transportation tunnel may even have a shape of a ramp as it is provided for in many foreign disposal projects [8]. Furthermore, after DDF is closed such layout will ensure that the

main flow of ground water passing through the excavations will overpass the RW disposal excavations thus reducing advective radionuclide flow and enhance the radiation safety level.

Repository evolution

Different repository evolution scenarios associated with different processes that may occur and take place due to some global or regional climatic, tectonic, cosmic or other events (features). Basically, probabilistic in nature these scenarios have as a starting point a scenario expected if all existing at this time internal impacts remain unchanged. It is that this scenario describing natural evolution of internal processes in the repository that is presented below.

Repository life cycle ideally involves two stages: short-term (decades) stage of construction and operation — RW emplacement and gradual closure (isolation) of the facility, and long-term (hundreds of thousand and even millions of years) stage of natural-engineered system evolution after its closure with no human intrusion. Repository safety should be ensured for the whole time period while the RW emplaced therein pose threat to the biosphere [9]. To meet this requirement careful siting is performed so that host rock mass performs its natural protective functions to the fullest extent possible. Engineered safety barriers are also installed — they may not fully exclude, but at least retard and slowdown radionuclide migration from the repository.

Interaction of engineered barrier system components, including RW matrix, with ground water is viewed as the key external cause resulting in radionuclide migration contributing to EBS degradation and radionuclide transfer first beyond repository boundaries, and then through the rock mass to the surface. It's considered to be impossible to find an unsaturated crystalline rock mass somewhere except for continuous permafrost zones with water present in the form of ice. And even in the latter case the depth is still not adequate. Thus, rocks of the massif and the rock mass itself should be at least of low permeability characterized with low permeability coefficients and rates.

Indeed, according to hydrogeological classification of soils, Eniseiskiy site rocks can be mostly attributed to low-permeability rocks with filtration factors varying from 0.001 to 0.0001 m per day. However, groundwater inflow into repository excavations (section of the repository constructed at the first stage of the project) obtained based on this data was estimated to reach 346 m³ per day [7]. This figure not being considered significant under operational conditions (to calculate water pumping amounts), can cause some problems at post-closure stage in terms of ground water interaction with EBS.

Ground water filtration velocity is harder to estimate as it is inversely related to opening rate (rate

between the surface area of open fractures to the cross-sectional area being transverse to the flow) and is in direct ratio to hydraulic gradient. As for the latter one, at the operational and post-closure stage (under atmospheric air) pressure (head) gradient of ground water in the drainage area of underground excavations — at a very small distance from excavation walls (meters—first tens of meters) may reach 5 MPa in accordance with the repository siting depth (some 500 m from the surface). This corresponds to the hydraulic gradient of 50—500 units being thousand and even tens of thousand times higher than relevant hydraulic gradient values estimates for undisturbed rock mass (up to 0.05) that will gradually decrease to normal values as hydraulic pressure (head) restores in the drainage zone of the excavations.

Opening rate depends on fracture aperture. For this reason, it usually decreases with the depth. However, this will only increase the filtration rate which is partially made up with the decrease of filtration coefficients with depth. Thus, under expected conditions, rock excavation saturation, or to be more precise — saturation of excavation damaged zone surrounding them and the gap before backfilling (first EBS component) will start already at the operational stage with excavations being gradually closed.

After ground water saturates the area around rock excavations EBS will ensure certain hydro isolation. Moreover, ground water head (pressure) values will remain the same — 5 MPa. This is due to the fact that EBS are closed under atmospheric air pressure, and the temperature especially in boreholes with heat-generating HLW may rise significantly as compared to natural background, especially during the first hundred of years [5].

EBS structure and materials — is one of most unelaborated issues to be addressed in further field and laboratory tests and URL experiments. In broad strokes, according to [3, 5] current EBS designs involve the components described below. For Class 2 RW the use of one filtration and sorption barrier is suggested. This barrier will be most probably made of clayey or cement-clayey material (figure 1b). RW will be placed into metal or reinforced-concrete container (1.5 m³). The waste is to be either incorporated into cement or glass matrix. Detail designs of waste matrix and waste emplacement itself have not been developed yet. Only design waste acceptance criteria containing relevant characteristics derived based on regulatory requirements are readily available to date [10].

Characteristics of RW Class 1 have been more thoroughly elaborated — vitrified HLW accumulated at RT-1 plant site (PA Mayak). More complex system of engineered barriers is proposed to ensure their safe disposal (table 1). Isolating containers (IC) made of steel and concrete having a wall thickness of some 60 mm in some way similar to the supercontainers used abroad will ensure the

Table 1. EBS structure for RW class 1 disposal

Barrier	Thickness of the barrier (package diameter), mm	Material porosity (void ratio), % vol.	
		Initial	Post-degradation
RW glass matrix	600	0–20	50
Steel cladding of the glass matrix	6+6+3	0	50
Basket	700	30–40	70
Compacted bentonite	150–200	40–45	45–50
IC internal steel cladding	3	0	50
IC bentonite cladding	60	10–15	20–30
IC outer steel cladding	5	0	50
IC	1200	30–35	60
Slip	50	70	70
Borehole	1,300	33–38	61

straight of RW package [8]. The gap between the IC and borehole walls, as well as the gaps between the containers will be filled with viscous bentonite thixotropic aqueous suspension – slip.

A cocoon with a thickness of no less than 15 mm made of bentonite blocks is lined inside the IC. This cocoon shall isolate a three-layered package with glass involving a basket, cask with three canisters placed on the top of each other. The total thickness of the packing layers is 15 mm, but, considering the assumption suggesting that the canisters are initially leaking, as the casks were exposed to prolonged radiation exposure during storage in the storage facility at PA Mayak site, barrier function can be only performed by the outer steel basket with wall thickness of 6 mm. Each canister can hold about 0.2 m³ of glass weighing 500 kg. In keeping with the standards established it can withstand specific heat generation rate of up to 2 kW/m³ resulting in package heating under limited heat transfer [5].

A specific challenge is attributed to the fact that RW package cannot be completely filled with barrier materials (up to 100%). Especially when it has multi layered structure. Gaps initially filled with air under atmospheric pressure may occur both inside materials being characterized by certain porosity or fracturing, between the layers of materials or can be produced due to material shrinking or may be associated with so called technological gaps. In the end, all these gaps are to be filled with ground water resulting in enhanced degradation of barrier materials and mobility of radionuclides.

Due to degradation of barriers themselves their porosity and, subsequently, void ratio of packages will increase. This will occur due to the dissolution of phases and phase transformations inside barrier materials. Most drastic increase in porosity values will be associated with aluminum phosphorous glass matrix and steel claddings – from 0 to 50% vol., to the smallest extent – the one of bentonite (no more than 10% vol.). Intermediate values associated with porosity increase (10–20% vol.)

are expected for concrete. Relevant values for glass matrix were obtained from experiments presented in [11]. As for the other materials – they were evaluated based on thermodynamic calculations.

Some issues associated with excessive void occurrence inside waste packages can be avoided already at the pre-disposal stage. For example, at PA Mayak casks with vitrified HLW will be additionally placed into additional baskets. Thus, these casks suffering from radiation degradation may be put away. So that the glass containing canisters will be placed into more robust containers instead of the casks and gaps filled with some inert material or, which is even better, with some useful barrier material. At the same time, sampling or non-destructive study of glass could be arranged for which is required for more accurate certification of disposed waste.

Arguments against such procedures based on cost assumptions can not be considered adequate. If no actual control of waste being disposed of is implemented, it can later result in much bigger costs required to ensure the safety. Thus, to ensure the required safety level foreign countries have already faced the need to redispense waste due to the misestimation of their actual state at the time of disposal [8] requiring huge resources to be invested. However, benevolent intentions can not drive this process. So, the enactment of federal norms and rules containing relevant requirement on the void volume of packages introduced as part of RW acceptance criteria [10] for disposal seems to provide a solution enabling to increase the repository safety. Summarizing the structure of repository EBS presented above, it should be noted that the list of main materials used for their construction is quite short: cement or glass matrix, rough (carbon) steel, bentonite, concrete, slip.

Properties of the above-mentioned materials are quite well studied. The fact that these materials will be used in different proportions and combinations for EBS construction does not change the essence of the matter. It allows some qualitative, and sometimes quantitative, predictions on the behavior under repository conditions that are needed to substantiate its possible evolution scenario.

Key functional characteristics of EBS materials have been evaluated using expert judgement approach with a 5-point scale. Relevant changes occurring due to material degradation and the rate of degradation were also evaluated. Table 2 summarizes the results obtained. These estimations were based on the evaluation of known EBS properties, some of which were discussed in [5, 12]. The expert values provided are directly proportional to the numerical values describing relevant material properties. For this reason, the scale “bad-good” presenting the functional usefulness of the materials different properties will be reflected differently. As for strength and sorption the higher value will account for “5”, as for filtration, diffusion and degradation rate – “0”. Evaluations for EBS material

Table 2. Qualitative estimates for the main functional characteristics of EBS materials

Property	Material				
	Slip	Steel	Concrete	Bentonite	Glass
Initial					
Strength	0	5	4	2/1	4-1(3)
Filtration	5	0	3	1	1-4 (3)
Diffusion	5	0	3	1	1-4 (3)
Sorption	2	1	3	5	1-4 (3)
Degradation rate	-	5/3	4	1	2-4 (3)
Post-degradation					
Strength	0	1	1	1	1
Filtration	5	5	4	2	4
Diffusion	5	5	4	2	4
Sorption	1	3	1	4	4

degradation rate has been also presented. A change of this characteristic by one unit corresponds to a change in the duration in years relevant for the material degradation process until the complete loses its useful properties by one decimal order from 10^6 years for “1” to 100 years for “5”.

The functional purpose of the slip is still unclear, since it has practically none of the useful barrier properties, apart from its relatively small sorption capacity, due to its “looseness” and small thickness of the layer. However, not much simplification can be expected, if it is assumed that this barrier is completely absent.

Steel has the highest strength of all materials considered and prevents 100% filtration and diffusion, until it is corroded. The process of steel corrosion is very uneven. The most susceptible to corrosion are welds and areas deformed by flexing and stamping, as well as the areas exposed to radiation. In addition, the rate of intergranular corrosion is always higher than the one of volumetric corrosion. Consequently, for steel element the decrease in the waterproof properties occurs much faster than of strength ones. Relevant times associated with the decrease of these functional characteristics to the minimum values of 10^2 and 10^4 years, respectively, can be calculated for the given thicknesses of steel claddings being part of EBS structure. Corrosion process will result in formation of new fine phases with high porosity, specific surface and sorption capacity.

Concrete strength is lower than the one of steel. Due to its high porosity, concrete also provides quite low sealing protection. Its sorption capacity is quite average and is associated mainly with the retention of components that are sensitive to high pH values of the solution (these values result from concrete interaction with water). However, this effect will gradually disappear due to Ca leaching being considered as the main component of the cementitious phase binder. At the same time, concrete strength properties will deteriorate. According to our estimate, in about 103 years, concrete

will turn into a low-strength, well-filtering mass with low sorption capacity.

Bentonite has the best filtration and sorption characteristics that can be preserved for much longer period of time as compared to other barrier materials. This is obvious, since bentonite is the only natural material, and moreover a product of weathering or hydrothermal rock alteration. Therefore, it “feels good” in a natural environment, and according to equilibrium-kinetic thermodynamic modeling if it is not disturbed, for example, by aggressive concrete leaching products, its lifetime will account for 106 years till the time of its complete degradation. Bentonite degradation results from the speciation process affecting its main component, montmorillonite. This component may speciate into other clayey minerals at different rates depending on physical and chemical conditions: chlorite, illite, kaolinite, having lower filtration and sorption characteristics as compared to montmorillonite [12].

The only “fault” that bentonite is characterized with is its low strength in the dry state and moreover when it gets saturated. However, a positive aspect regarding this “fault” is associated with its plastic flow properties under pressure enabling it to fill the voids not yet filled with other solid matters, including fractures and pores, increasing the sealing integrity.

It seems that glass matrix is considered a most questionable material of all EBS elements. Huge uncertainties associated with its characteristics result from the lack of knowledge on the current state of vitrified waste at PA Mayak. If this RW complies with waste acceptance criteria set for Class 1 RW then this material should be homogeneous in terms of its phase composition, element and radionuclide content and composition. On the other hand, it should be a monolith, weakly fractured, nonporous and durable material. It should not infiltrate water solutions. It should be characterized by low leaching rate and actual glass matrix blocks should retain their properties for up to 105 years.

However, if during storage thermal and radiation impacts results in glass recrystallization, its structure may be altered up to a loose unconsolidated state. In this case no durability of such material may be guaranteed due to high porosity and specific area. Due to ground water interactions the latter one will increase both filtration and leaching rate, as well as radionuclide transfer (diffusion, advection) in liquid phase. The only positive effect in this situation is associated with increased sorption due to glass degradation products.

The range of derived estimates for glass characteristics accounting for these critical cases is presented in table 2. However, possible and event more probable is an intermediate state of this material the characteristics of which indicated in brackets are to be considered during the evaluation of repository evolution scenarios.

Further are discussed the stages and the time frames for key evolution processes that are to occur at repository post-closure stage (table 3). These were derived based on the layout of EBS for RW Class 1 (table 1) and main characteristics of EBS materials (table 2). EBS evolution involves three stages differing in terms of the driving forces engaged and duration.

The first stage covering processes from 1 to 7, takes place under high local pressure gradient of ground water flow resulting in a flow through fractured rocks and porous barrier materials that is temporarily retarded only by impervious steel cladding of EBS elements. As the sealing capacity of the claddings gradually decreases due to local corrosion intensified by high heat output, the flow resumes and ground water fills pores and fractures inside EBS materials and engineering void spaces. Duration of this stage is estimated to be up to 1,500 years. Furthermore, these processes occur one by one as depicted at figure 2 representing maximum duration of relevant processes.

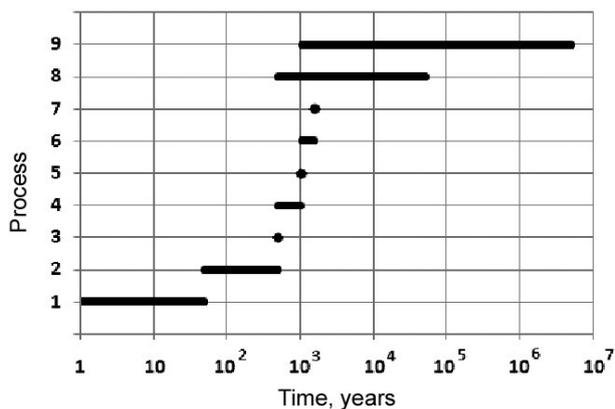


Figure 2. Key processes of EBS material evolution for RW Class 1 at the post-closure stage. Numbers are provided in consistence with the data given in table 3

Such processes as volumetric corrosion of steel cladding and dissolution of cement stone gradually resulting in concrete degradation initiated at the first stage resume further on until the full loss of EBS element strength characteristics occurs. These processes are not depicted at figure 2, but contribute significantly to the next stage of EBS evolution. The second stage of EBS evolution (process 8) is associated with volumetric degradation of barrier materials resulting in the mechanical strength loss. This process occurs due to chemical, electrochemical and radiation impacts produced by water solution on these materials. In the very beginning of this stage high temperature may produce an accelerating effect.

However, the key factor accelerating the degradation of EBS materials may turn out to be associated with their interaction if their chemical properties prove to be incompatible. In this regard, the use of concrete is considered a most questionable issue. Its capacities are not considered to be quite

impressive (table 2), but excessively high pH values of the solution (up to 12 and even higher) generated due to concrete interaction with ground water (alkali plume) produce negative impact on the stability of concrete and particularly glass matrix resulting in accelerated degradation. Moreover, the concrete is the first EBS material to lose its strength. Thus, if another material is used instead concrete inside IC structure, for example, cast iron or cast material (basalt casting) with the latter viewed as a preferred option, RW package life time would be significantly increased.

Strength loss stage may last tens of thousand of years mostly by virtue of steel claddings. However, sooner or later even their sealing capacity will be lost resulting in a situation characterizing general adequacy of RW class 1 packaging layout. It seems very likely that the emplacement scheme proposed for 75 m tall vertical boreholes will induce some unfavorable events each one of them being potentially associated with some negative impact (table 3, p 8a). At the same time, in case of horizontal emplacement boreholes for packages containing RW Class 1 the durability of barrier materials should not be considered a characteristic being critical in terms of repository evolution (table 3, p. 8b).

This RW disposal option is being considered in foreign projects both as the reference one and the alternative one. As an alternative option to KBS-3V, the reference disposal concept, Posiva Oy responsible for disposal of SNF generated by NPPs in Finland together with the Swedish SKB explores the opportunities for horizontal emplacement of SNF canisters (KBS-3H) in the underground research facility ONKALO located in granite gneiss formations of the Olkiluoto municipality [13]. SURAO, Czech national operator for RW management, has selected the KBS-3H concept as the preferred one for the development of deep repository in granites and is currently searching for a suitable site [14]. NRC, U.S. nuclear regulatory authority, has started the implementation of HLW disposal project in 1976 providing for HLW disposal in horizontal excavations drilled in tuff formations (Yacca-Mountain) [15]. Similar examples can be provided for deep disposal facilities and URLs sited not only in crystalline formations, but also in sedimentary rocks [8].

Thus, if negative effects resulting in the loss of strength by RW packages can be avoided by the use of horizontal emplacement concept or some other method, repository evolution will proceed to the last and most long-lasting stage involving continuous leaching and transfer of radionuclides initiated with glass matrix saturation. But in the latter case it will occur along with gradual deterioration of concrete barrier properties. The key goal enabling to defer this time to the maximum possible extent is to ensure that radionuclide transport is as far as possible governed by diffusion. Diffusion mechanisms are being considered as the slowest and most effective way enabling to achieve the full sorption

Table 3. Stages and duration of EBS evolution processes at post-closure stage of deep disposal facility for RW Class 1

Process	Duration	Result
1. Restoration of hydraulic pressure in the drainage zone of rock excavations	$n - n \cdot 10$ years	Occurs when IC outer steel cladding interacts with ground water under elevated temperatures and pressure up to 5 MPa
2. Local corrosion of IC outer steel cladding	$n \cdot 10 - n \cdot 10^2$ years	Loss of sealing capacity by IC outer steel cladding, formation of an interface between IC concrete layer and underground water under pressure of up to 5 MPa and elevated temperature
3. Ground water flow under pressure through pores and fractures of IC concrete layer	$n \cdot 10^{-1} - n$ years	Formation of an interface between IC inner steel cladding and ground water under pressure of up to 5 MPa. Start of IC concrete cladding volumetric degradation under elevated temperature
4. Local corrosion of IC inner steel cladding	$n \cdot 10 - n \cdot 10^2$ years	Loss of sealing capacity by IC inner steel cladding, formation of an interface between the compacted bentonite and ground water under pressure head of up to 5 MPa and gradually decreasing temperature.
5. Flow of ground water (with altered composition following its interaction with concrete) through compacted bentonite under pressure	$n - n \cdot 10$ years	Formation of an interface between the steel overpack and ground water with altered composition under pressure head of up to 5MPa. Start of bentonite volumetric degradation under elevated temperature as compared to background values.
6. Local corrosion of steel overpack and cask	$n \cdot 10 - n \cdot 10^2$ years	Loss of sealing capacity by the overpack and cask, formation of an interface between vitrified RW and ground water with altered composition under a pressure head of up to 5 MPa and continuous decrease in temperature.
7. Filling (under pressure) of voids inside the cask, as well as pores and fractures of the glass matrix with water solution resulting from ground water interaction with concrete, bentonite and steel. Equalized hydraulic pressure values within EBS, disappearance of local head and active filtration through EBS materials. Start of glass matrix leaching under continuous decrease of temperature up to background values accompanied by the decrease in volume and increase of the solid phase porosity. Active water radiolysis.	$n - n \cdot 10$ years	Saturation of the water solution with glass matrix elements and radionuclides, start of glass matrix degradation and radionuclide diffusion through bentonite barrier. These processes may start both under increased temperature and the one close to background values. Gas generation may result in local increase of pressure inside the liquid phase and induce hydraulic pressure head directed outwards and inducing filtration of radioactively contaminated solution. Radiolysis also results in formation of oxidizing condition contributing to increased mobility of redox-sensitive radionuclides and steel corrosion, as well as bentonite oxidation. The latter results in the decrease of solution pH values and increased degradation of concrete.
8. Volumetric corrosion of steel claddings, concrete and glass matrix degradation. Loss of strength by barrier structures in the following order: concrete, steel, glass. 8a For the designs considering RW Class 1 disposal in vertical 75-m deep boreholes under gradual reduction of strength characteristics for the whole IC structure possibly resulting in sinking of RW packages undergone strength loss with gradual or uneven increase of the overload pressure acting on the liquid phase near lower waste packages by several MPa. 8b For the designs considering RW Class 1 disposal in horizontal rock excavations, the strength of the structure if no external impacts exist (not being considered in this paper) produce no impact on EBS evolution. A shift to stationary processes is possible under sorption saturation of barrier materials.	$n \cdot 10^4$ years	Option 1 (catastrophic). Failure of bentonite barrier integrity. Instant release of radioactively contaminated materials, mostly, solution outside EBS. Option 2 (evolutionary). Filtration of radioactively contaminated solution outside EBS through bentonite barrier under excessive pressure increasing up to first MPa towards the lower RW packages. A combination of the abovementioned options is possible. In any case this event sequence is considered to be quite a negative one and it can occur in a quite foreseeable timeframe. Radionuclide release out of EBS results from diffusion occurring through barrier materials. Bentonite barrier is considered to be the main retarding element. Additional contribution to the sorption retardation of radionuclides is associated with the products generated due to corrosion of steel claddings, and to lesser extent – the one of concrete.
9. Degradation of bentonite barrier with gradual increase in water permeability and diffusion rate	$n \cdot 10^6$ years	General decrease in radioactive impact may be accompanied by gradual acceleration of diffusion and local advective release of radionuclides out of the engineered barrier system.

capacity of barrier materials contributing to radionuclide retardation. Considering the stability of bentonite in natural environment, even accounting for relevant thermal and radiation impacts we can expect that it will maintain its barrier properties for millions of years. At the same time, particular attention should be given to relevant calculations of bentonite barrier thickness, density and composition, as well as experimental testing aiming to evaluate the adequacy of filtration, compression, diffusion and sorption parameters for this particular case.

Conclusion

Repository localization and evolution scenario analysis performed for the Eniseiskiy site allows to conclude that safety of this facility designed for Class 1 and Class 2 RW disposal can be ensured at least for a time period of some millions of years. But only if relevant corrections into design solutions suggested are introduced. Such corrections, firstly, cover the principles associated with co-location and space orientation of rock excavation groups intended for RW disposal: emplacement in a bigger number of horizons located within small monolith areas identified as the result of detailed investigations. Emplacement excavations should be drilled along geological structures. Secondly, they should address the use of alternative (horizontal) way of Class 1 RW emplacement in excavations. Thirdly, performance assessment associated with the decision making on EBS construction materials (for example, cast iron or stone casting to manufacture IC instead of concrete) should be further elaborated involving relevant calculations to estimate the required thickness, density and composition (bentonite and other clay materials and mixtures).

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Disposal of RW

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