# FORMATION OF REQUIREMENTS FOR THE SYSTEM OF DISPOSAL OF RADIOACTIVE WASTE IN THE CONTEXT OF PLANS FOR REPROCESSING OF SPENT NUCLEAR FUEL

A.N. DOROFEEV
State Atomic Energy Corporation “Rosatom”
Moscow, Russian Federation

A.V. KHAPERSKAYA
State Atomic Energy Corporation “Rosatom”
Moscow, Russian Federation

D.V. BIRIUKOV
Nuclear Safety Institute of the Russian Academy of Sciences
Moscow, Russian Federation
Email: biryukov@ibrae.ac.ru

P.A. BLOKHIN
Nuclear Safety Institute of the Russian Academy of Sciences
Moscow, Russian Federation

N.I. DROBYSHEVSKY
Nuclear Safety Institute of the Russian Academy of Sciences
Moscow, Russian Federation

A.A. SAMOYLOV
Nuclear Safety Institute of the Russian Academy of Sciences
Moscow, Russian Federation

S.S. UTKIN
Nuclear Safety Institute of the Russian Academy of Sciences
Moscow, Russian Federation

**Abstract**

The paper describes the process of forming requirements for the disposal system for radioactive waste on the example of the thermal regime of deep radioactive waste disposal facility.

## 1. INTRODUCTION

In accordance with the Russian legislation in the field of the atomic energy use, radioactive waste disposal in final isolation facilities is required, and every new nuclear fuel cycle facility must have its own infrastructure to avoid placing an undue burden on future generations.

At present, the Russian Federation is designing a deep radioactive waste disposal facility (DRWDF) in the Nizhnekansky rock massif (hereinafter referred to as NKM), where it is planned to dispose high-level waste, which is generated from the spent nuclear fuel (SNF) reprocessing from the operated and newly constructed facilities.

The work on the development of the DRWDF is carried out in accordance with the approved strategy [1] and stipulates that an underground research laboratory (URL) will operate at least until 2030 (Fig. 1), then a decision of the final isolation facility construction will be made.

The creation of the URL precedes the stage of the facility operation as a deep disposal facility for radioactive waste. The concept of DRWDF provides for the disposal of class 1 and 2 radioactive waste, as well as secondary RW generated during its operation, namely:

— RW class 1 - heat-generating HLW (heat generation not more than 1.5 kW / m3). This waste is planned to be placed in the wells connecting the underground horizons of the object;
— RW class 2 - long-lived ILW and short-lived (heat generation not more than 100 W / $m^3$), which are planned to be placed in horizontal mine workings.

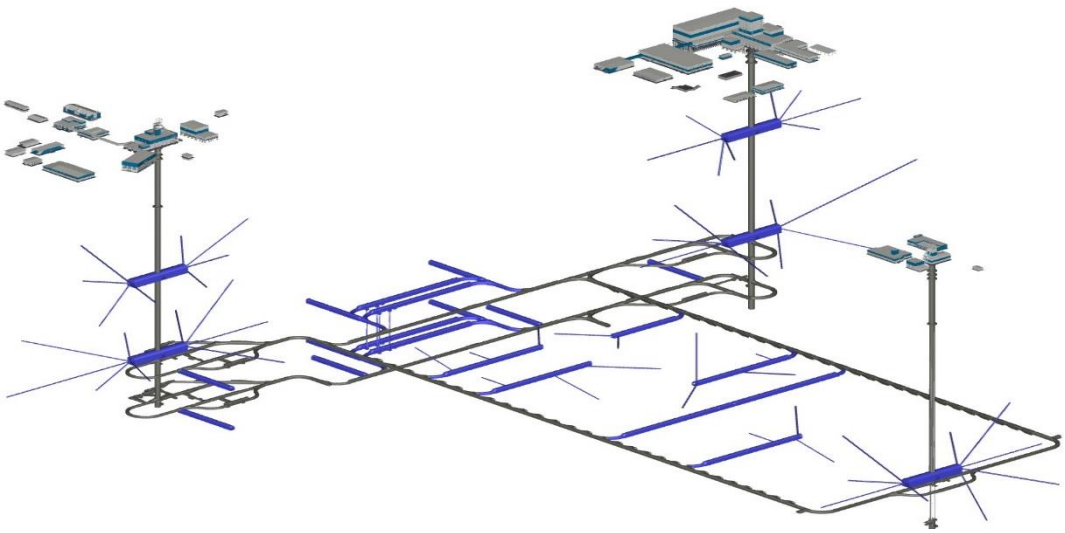


*FIG. 1. The design of the URL (blue marked projected research formulation).*

Nowadays, in view of the next iteration, related to the specification of data on the characteristics of disposed wastes, work has begun to clarify the requirements for the radioactive waste disposal system, which should create conditions for a project development that will dispose of radioactive waste in compliance with safety requirements.

Such a scheme is fully consistent with the recommendations of the IAEA (SSR-5, SSG-14), according to which the safety case is a multi-iteration process that can be simplified as follows (Fig. 2).

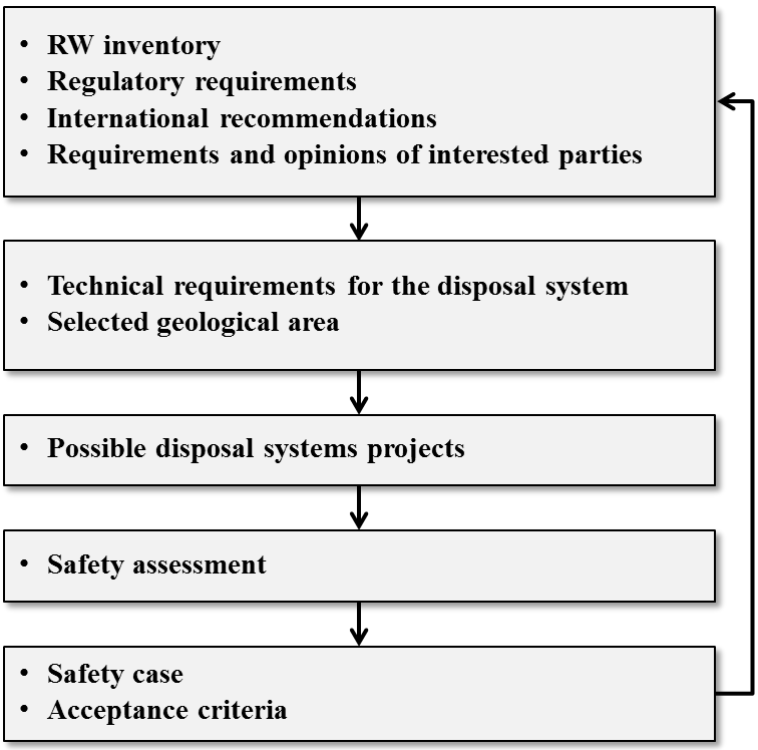



*FIG. 2. Simplified safety justification scheme.*

Figure 3 shows the formation of the system of requirements for the disposal system for radioactive waste. The geological environment properties, the volume and characteristics of accumulated waste, as well as decisions on the system of engineering safety barriers influence the decisions on the DRWDF characteristics. The concept of disposal is developed taking into account the characteristics of the waste, such as radionuclide composition, heat generation, leaching rate, etc.

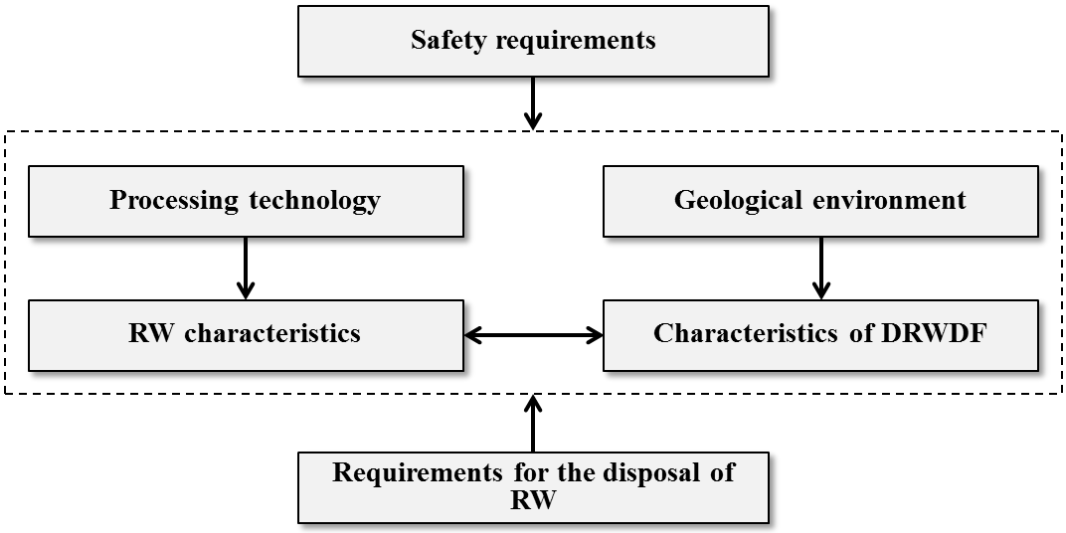



*FIG. 3. Scheme of formation of the system requirements for the disposal of radioactive waste.*

## 2. SOURCES OF RADIOACTIVE WASTE

Currently, the only one plant in the Russian Federation for the reprocessing of spent nuclear fuel from power, research, industrial and transport ship installations is the radiochemical plant RT-1 of the FSUE PA «Mayak» with a design capacity of 400 tons of SNF per year [2]. Plant RT-1 was commissioned in 1977.

The main task of the RT-1 plant is the reception, temporary storage and processing of various types of SNF from power and research reactors. At the present time over 6,000 tons of SNF have been reprocessed. The plant is processing irradiated nuclear fuel from power-generating reactors (WWER-440, BN-600, RBMK-1000, WWER-1000), transport power ship installations, research reactors.

The technological scheme for the reprocessing of spent nuclear fuel at the RT-1 plant is based on water-extraction technologies that are close to the classical version of the PUREX-process scheme. Highly active wastes are cured to produce aluminophosphate glass. For curing HLW in operation is the installation of EP-500/5. Currently vitrified high level waste is stored in a specialized storage facility. The technology of SNF reprocessing is described in details in [3, 4]. At the same time, during the processing of SNF, vitrified HLW with a heat generation less than 5 kW/m$^3$ and a specific formation of vitrified HLW is about 0.6-0.8 m$^3$ per ton of SNF [4, 5] is formed.

At present, a pilot demonstration center is being built for the development of promising technologies and equipment and the processing of SNF from thermal reactors at FSUE “MCC” with a design capacity of up to 250 tons of SNF per year (hereinafter referred to as PDC MCC). The PDC MCC provides for the use of innovative technologies for the reprocessing of SNF with minimal waste generation and the complete absence of discharges into the environment. All highly active liquid RW are sent to a vitrification plant with the production of borosilicate glass, and further to temporary storage.

The PDC MCC project provides that during the processing of WWER-1000 SNF, vitrified HLW with a maximum permissible heat generation less than 30 kW/m$^3$ RW with a specific formation of 0.1-0.14 m$^3$ per tons of SNF will be generated [5].

Russia operates two fast reactors with sodium coolant and there is a program for their development, at the same time work is underway to create an alternative type of fast reactors - with lead coolant and to build a pilot demonstration reactor BREST-OD-300. As part of this program, technologies for closed nuclear fuel cycle of the RBN are being developed, including technologies for the production of mixed uranium-plutonium fuel in the RBN (oxide and nitride) and technologies for the processing of FBR SNF reactors, including combined (pyrometallurgy + hydrometallurgy) technology for processing MNUP and MOX of SNF FBR (PH-process), as well as HLW fractionation technology. This program provides for the construction of a module for the reprocessing of spent nuclear fuel in the framework of the construction of Pilot Demonstrator Energy Complex (PDEC) on the basis of a closed nuclear fuel cycle with a capacity of up to 10-25 tons/year of high burnup fuel [6].

Thus, it is possible to reprocess SNF from fast reactors (MNUP and MOX) BREST-300, BN-600, BN-800 at several sites for the reprocessing of SNF. Reprocessing of SNF BN-1200, transport and research reactors is potentially possible.

Thus, there are three sources of disposable HLW and their characteristics can be distinguished:

(a) RW from SNF reprocessing at PA “Mayak” (the technology of SNF reprocessing and the characteristics of RW were determined);
(b) RW PDC MCC (technology for reprocessing spent nuclear fuel has been determined, the characteristics of RW can be partially managed);
(c) RW PDEC (SNF reprocessing technology is at the R&D stage). In this paper will not be considered.

## 3. SAFETY REQUIREMENTS

The task of forming a system of requirements is an integrated one; in the framework of this work, we will consider only one of its aspects - the thermal effect of the disposed radioactive waste on the state of the DRWDF, which is one of the key factors affecting the concept of disposal.

The temperature regime is a limiting factor in determining the allowable density of HLW during disposal and, therefore, one of the fundamental criteria for vitrified HLW will be the limitation on energy release. This limitation is due to two factors - the limitation of the temperature and the associated stress-strain state of the mountain massif and the limitations caused by the best modes of operation of safety barriers. The first restriction, in addition to the heat generation requirements, can be resolved by sparse placement of packages in the mountain massif, and the second, by restriction on the materials used in engineering barriers and their design.

At present, the current requirements have established that the heat dissipation of the packaging of RW class 1 should not exceed 2 kW/m$^3$. Given the estimated size of the package, the heat dissipation restrictions on the primary canister with vitrified HLW will be at the level of 600 W, which is consistent with the criteria established in the world.

It should be noted that in the future the requirements for heat generation can be updated taking into account the characteristics of radioactive waste and layout solutions of the DRWDF.

As for the accumulated and generated RW at PA «Mayak», their characteristics preliminarily satisfy the acceptance criteria (which will be shown below).

For prospective wastes, it is possible to form a vitrified HLW with the characteristics of heat generation required for disposal by optimizing technological processing processes, namely, targeted selection of accumulated SFAs of various extracts for co-processing.

Figure 4 presents data on heat generation during combined processing of SFAs with different exposure and burnup.

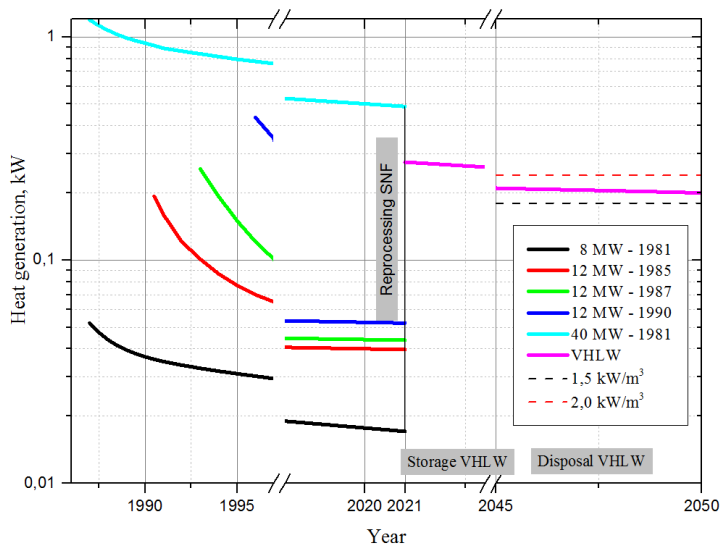



*FIG. 4. Calculation of the heat generation of waste from the processing of accumulated SFA with different exposure and burnup.*

Preliminary calculations show that co-processing of SFAs with different exposure and burnup allows us to withstand the requirements for heat generation. Moreover, from the point of view of compliance with the acceptance criteria for HLW obtained, the introduction of HLW fractionation technology with the release of heat-generating nuclides (cesium and strontium) is promising. With regard to the characteristics of the DRWDF, there are currently prepared several options for building layout decisions, for which the computational models have been prepared and preliminary estimates of temperature regimes have been made.

In the framework of this work, we will consider the thermal state of an object using the example of a basic layout of a DRWDF [7] with regard to stored waste.

## 4. THERMAL STATE OF DRWDF IN NKM WHEN LOADING WASTE PA “MAYAK”

In the framework of this work, all calculations were carried out in relation to the basic layout of the DRWDF [7]. According to the basic layout, the facility will be divided into two sections, with accumulated waste with an initial heat generation of about 1 kW/m$^3$ will be placed on the first site, and waste with an initial heat generation of 1.5 kW/m$^3$. Each year one row of wells will be successively filled.

To simulate the thermal state of the DRWDF, the calculation code is FENIA (Finite Element Nonliner Incremental Analysis) [8]. As a result of the calculation, the following values of the temperature regime of the DRWDF were obtained. Figure 5 shows the change in the spatial temperature profiles with time at the stage of heating and cooling of the DRWDF.

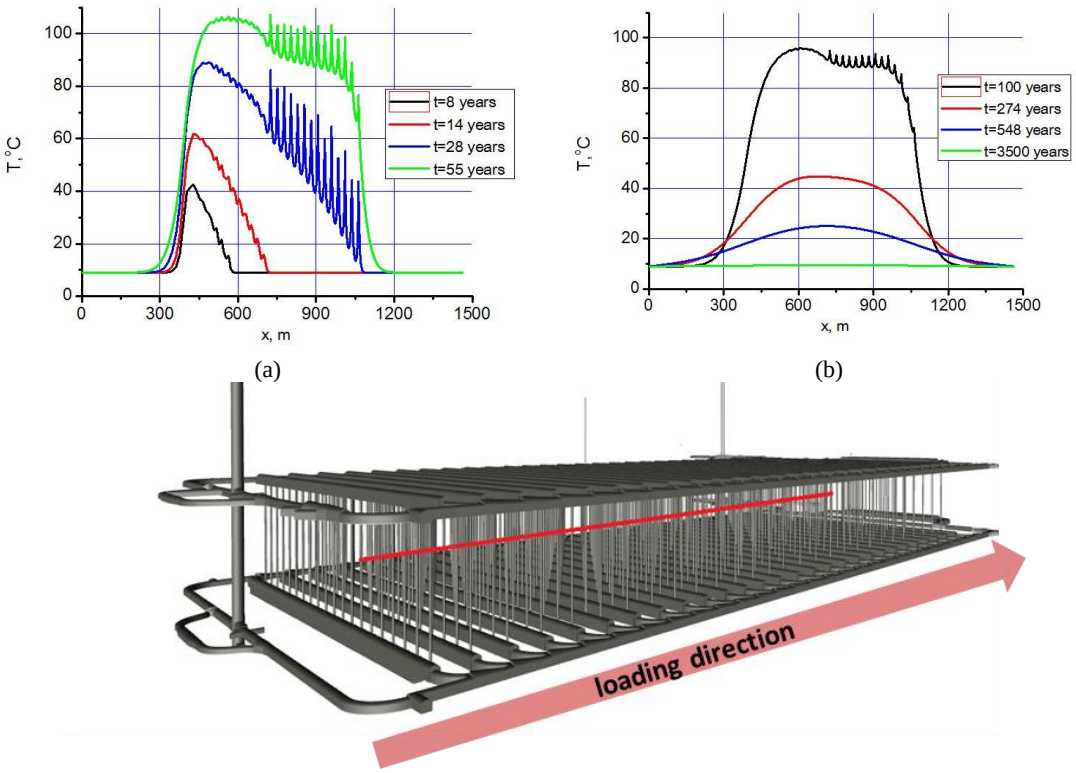



*FIG. 5. Spatial temperature profiles along the horizontal axis of symmetry at the stage of heating (a) and stage of cooling (b).*

The time t = 28 years corresponds to the full load of DRWDF, at the time point t = 55 years, temperatures close to the maximum are observed. After 100 years from the beginning of filling, the temperature in the volume of DRWDF is equalized and is 90-95 ° C, and after 3500 years, as noted above, it almost returns to its original values.

Figure 6 shows the temperature dependence for the central part of the internal wells of sections 1 and 2.

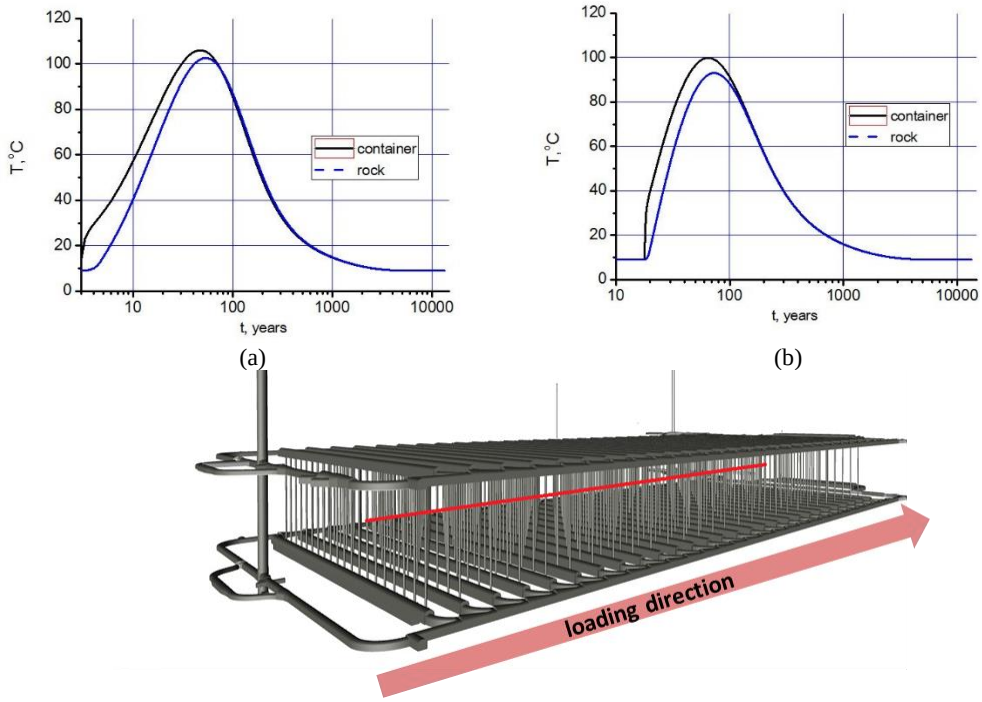



*FIG. 6. Time dependences of temperatures for the center of an internal well for section 1 (a) and 2 (b).*

Calculations show that the maximum temperatures will be reached in 50-70 years from the beginning of the DRWDF filling and only after 8-9 thousand years the temperature in the entire computational domain returns to the initial values.

## 5. THERMAL STATE OF DRWDF NKM WHEN LOADING ACCUMULATED AND PROMISING RW

### 5.1. Analysis of heat generation of promising waste

Preliminary calculations for the heat emission of promising RW, performed using the TRACT calculation code [9], indicate that the energy release of the generated class 1 waste at the initial time is ~ 25 kW / $m^3$ (during the reprocessing of fresh, not aged SNF) (Fig. 7) that does not meet the preliminary criteria for the acceptability of RW for DRWDF in NKM (heat generation is not more than 1.5 kW / $m^3$). The approximate time to achieve the requirements for specific heat generation in the implementation of long-term exposure using standard canisters and burial cells is ~ 100 years.

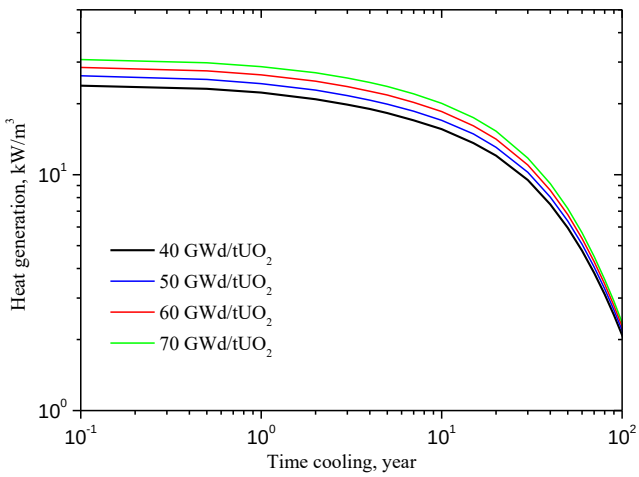



*FIG. 7. Residual heat generation in the RW for $UO_2$ fuel of various burnup.*

### 5.2. Evaluation of the thermal state of the pilot ground protection plant under joint loading of waste from the PA «Mayak» and PDC MCC

As in the previous calculation for PA «Mayak», the DRWDF will be divided into two sections, while the first section will contain waste with an initial heat generation of about 1 kW / $m^3$ (waste from the PA "Mayak"), and the second section will contain the waste from the PA "Mayak" and PDC MCC.

Each insulating container in the row contains 2 canisters with RW PDC MCC with an initial heat generation of about 25 kW/$m^3$ and 3 canisters with RW PA «Mayak» with an initial heat generation of 1.5 kW / $m^3$. In this case, it is considered that for the waste of the PDC MCC, the heat generation rate decreases in time according to the exponential law.

As a result of the calculation, the temperature regime values were obtained for a period of up to 10,000 years. As an example, Fig. 8 shows the temperature distribution 50 years after loading. A graph of temperature distribution along the line (Ox axis), passing in the middle of the burial perpendicular to the rows of wells, crossing the central well of each row of section 2, is shown in Fig. 9.

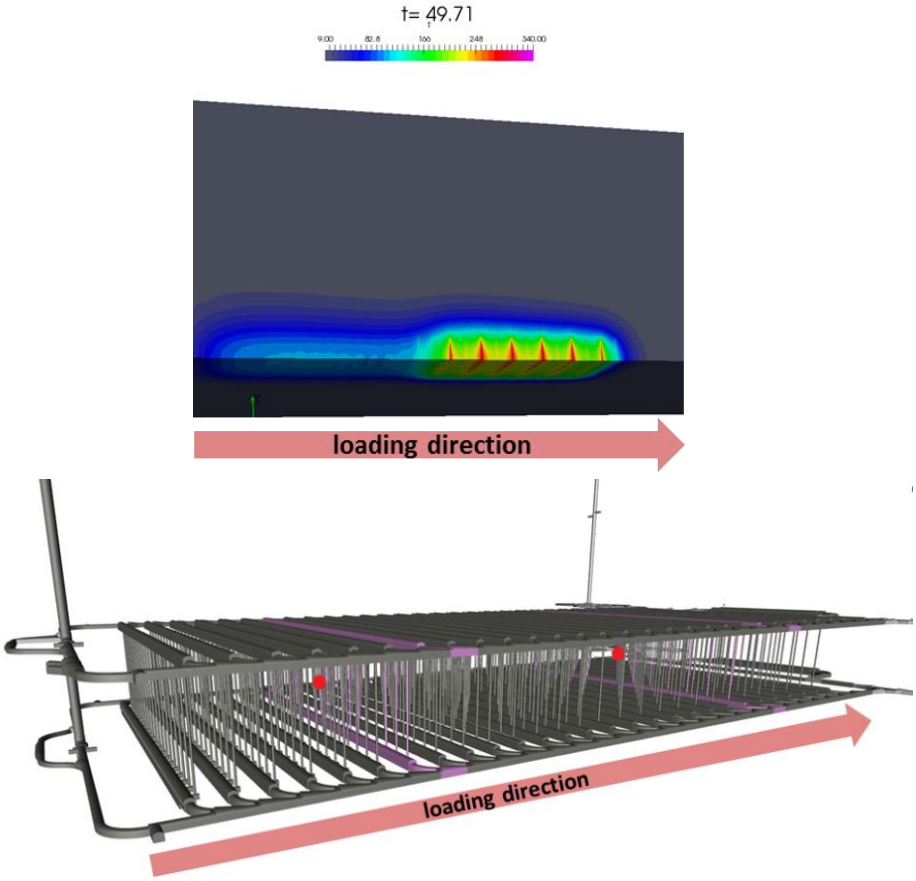



*FIG. 8. Temperature distribution 50 years after loading.*

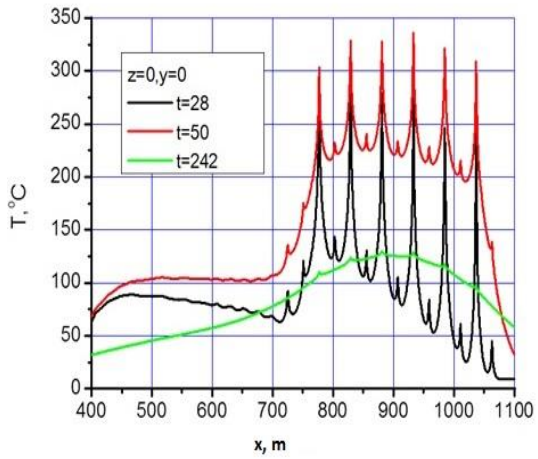



*FIG. 9. Temperature distribution along the Ox axis at different points in time.*

The maximum temperature in the matrix of RW PDC reaches 340 ˚C.

### 5.3. Assessment of the thermal condition of the DRWDF at the joint disposal of waste from PA «Mayak» and RW PDC MCC after preliminary exposure

The main difference from the previous calculation is in the disposal of RW PDC MCC after 50 years of exposure. As a result of the calculation, the temperature regime values were obtained for a period of up to 10,000 years. Figure 10 shows the temperature distribution after a 50-year exposure before loading.

The dependence of temperature on time in different materials at points in the area of the penultimate well (the last with RW PDC) of section 2 is shown in Fig. 11. Solid lines show temperature versus time at the disposal of RW PDC without exposure, dashed at burial with an exposure of 50 years and 50 years after loading.

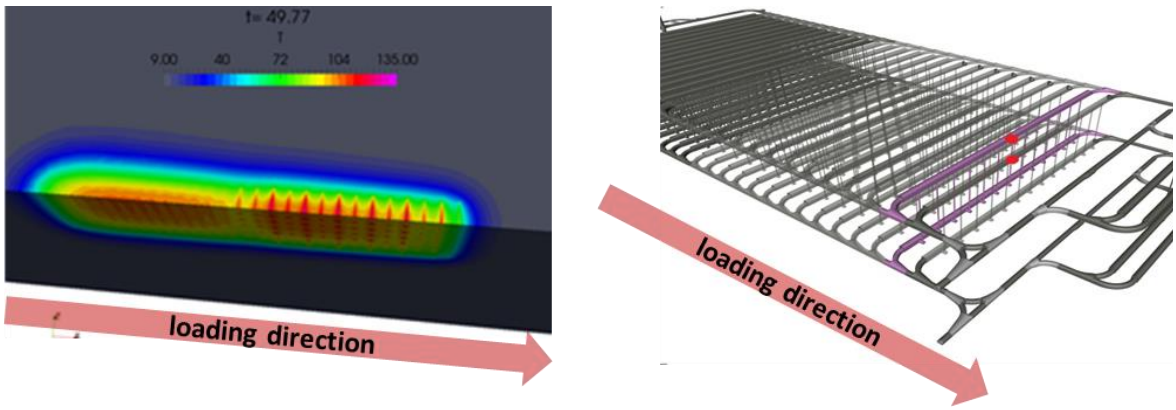



*FIG. 10. Temperature distribution after 50 years of exposure and 50 years loading.*

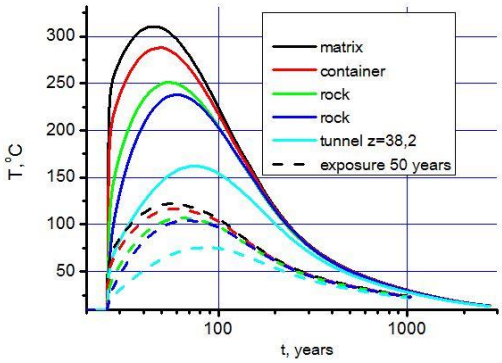



*FIG. 11. Temperature distribution over time.*

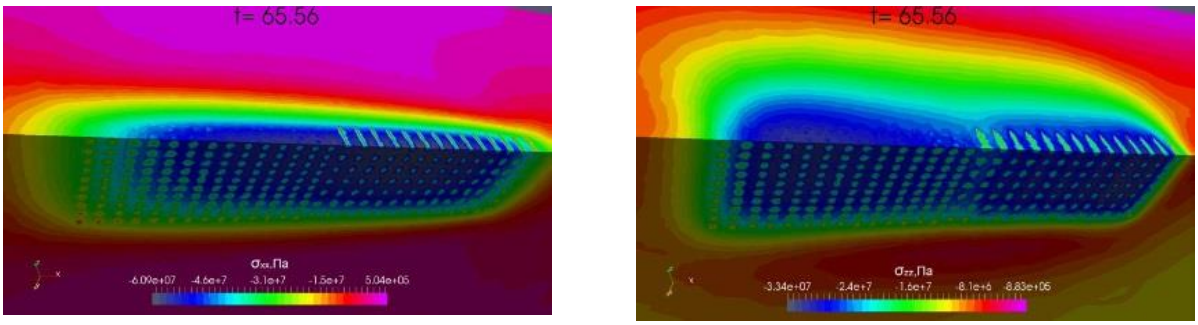
The use of 50 years for disposal of radioactive waste has led to a decrease in maximum temperatures to 130 ° C.

## 6. INVESTIGATION OF THE STRESS-STRAIN STATE

In the process of loading and operation, the structural elements of the underground burial and the surrounding rock are subjected to various stresses, in particular mechanical and thermal ones. The main mechanical loads are associated with the effect of the weight of the overlying rock and tectonic stresses caused by the interaction of rock blocks. In the calculations, only the weight of the overlying rock is used as the external mechanical load, and the tectonic stresses of the rock blocks are not taken into account.

When calculating the stress-strain state, the results of the calculation of the temperature regime of the PDC MCC after 50 years of exposure were used (the results of the calculation of the thermal regime of this option were considered above).

Figure 12 for the time point t = 66 years shows the distribution of horizontal and vertical stresses. The moment of time t = 66 years is chosen, since it is precisely in the period of 50-70 years after the start of burial that the maximum values of temperature are observed and, accordingly, the maximum values of stresses.




(a) (b)

*FIG. 12. Stress distribution at time t = 66 years. (a) - horizontal stresses (b) - vertical stresses*

Calculations show that such increased stress values can lead to destruction of the host rock in the space between the wells or in the area bordering the disposal site. In order to assess the possibility of such failure, the strength criteria were used, which allow judging by the stress values of the destruction or the preservation by the material of its mechanical strength.

## 7. EVALUATION OF THE STRENGTH OF ELEMENTS DRWDF

Strength calculations were implemented in the FENIA calculation code [8]. After calculating the stress tensor, the main stresses and the field are calculated with the relative norm of the distance from the current values of the stresses to the combination of stresses at which the material will be destroyed (damaged).

The calculation was carried out for the option of sequential placement of the RW PA “Mayak” and PDC MCC. The stresses for this placement were discussed above; here, using the Hoek-Brown criterion, the strength of the host rock was evaluated.

Estimation of material damage using the Hoek-Brown strength criterion is shown in Fig. 13. The most dangerous stress values are located above and below the wells, where the normalized distance to fracture (we will call this distance material damage later, 0 - the material is completely intact, 1 - destroyed) reaches 0.6.

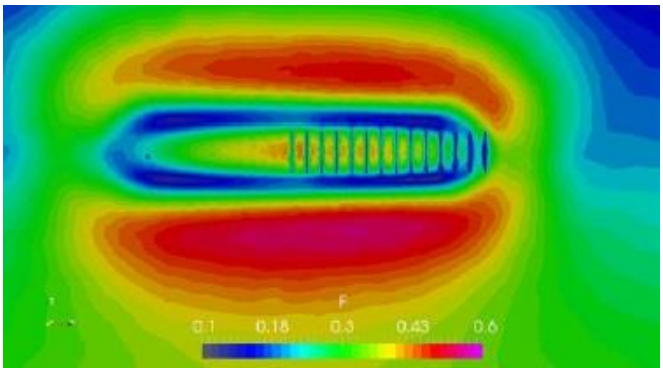



*FIG. 13. Distribution of material damage Hoek-Brown criterion.*

The calculations of the stress-strain state and the strength of the surrounding medium and protective barriers show that when placed in containers of RW PDC MCC after 50 years of exposure, no stresses are observed that lead to the destruction of materials from temperature effects. The results are preliminary and approximate, because the heterogeneities in the distribution of the host rock (different types of rocks, the presence of cracks, etc.) were not taken into account. Specification of the rock properties will be made at the stage of construction and operation of the URL.

## 8. CONCLUSION

In the framework of this paper, the process of forming the requirements for the system for the disposal of radioactive waste is considered on the example of the thermal regime of DRWDF.

It is shown that the characteristics of accumulated and generated waste meet the requirements for heat generation. For prospective wastes, it is possible to form a waste with the required heat generation characteristics for disposal by optimizing the technological processes of SNF reprocessing or their additional exposure before disposal.

These calculations are one of the stages of research and will be adjusted as the characteristics of radioactive waste, design solutions and engineering safety barriers are refined.

## REFERENCES

[1] Strategy for the creation of a deep disposal site for radioactive waste, Radioactive waste. **2** (3) (2018) 114-119, www.ibrae.ac.ru/docs/Radwaste_Journal_2(3)18/114_120_Strategy.pdf

[2] DZEKUN, E., KOLUPAEV, D., LUTOVININ, A., ROVNY, S., Possible solutions for the modernization and reconstruction of the initial stages of reprocessing of spent nuclear fuel at the RT-1 complex, Radiation safety issues 1 (1999) 68-76.

[3] ROVNY, S., SHEVTSEV, P., The current state and ways to improve the radiochemical technology for the isolation and purification of uranium and plutonium // Radiation safety issues **2** (2007) 5-13.

[4] GLAGOLENKO, YU., DZEKUN, E., ROVNY, S., SAZHNOV, V., UFIMTSEV, V., BROSHEVITSKY, V., LAPTEV, G., OSNOVIN, V., ZAZARKIN, B., SMELOV, V., NENAROKOMOV E., NIKIPELOV B., Reprocessing of spent nuclear fuel at the RT-1 complex: history, problems, prospects, Radiation safety issues **2** (1997) 3-12.

[5] GAVRILOV, P., Krasnoyarsk region – territory promising developments of the nuclear industry, www.osatom.ru/mediafiles/u/files/VII_reg_forum_2014/Plenarnoe_zasedanie_Gavrilov_Krasnoyarskij_kraj_territoriya_perspektivnyx_razrabotok.pdf

[6] ADAMOV, E. Closed Fuel Cycle Technologies Based on Fast Reactors as the Corner Stone for Sustainable Development of Nuclear Power / Int. Conf. on Fast Reactors and Related Fuel Cycles: Next Generation Nuclear Systems for Sustainable Development (FR17), IAEA CN-245-561, 26-29 June (2017).

[7] Underground research laboratory in the Nizhnekansky Massif, www.norao.ru/about/underground (available only in Russian)

[8] DROBYSHEVSKY, N., MOISEENKO, E., BUTOV, R., TOKAREV, Y., Three-Dimensional NumericalSimulation of the Thermal State of the Deep Burial Site of Radioactive Waste in the Nizhnekansky Rock Massif, Radioactive Waste **1** (2017) 64-73.

[9] BLOKHIN, A., BLOKHIN, P., SIPACHEV, I., Possibilities of the TRACT calculation code for solving problems of characterization of radionuclide composition of radioactive waste and SNF, Radioactive waste **2** (3) (2018) 95-104.