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ON REACTOR GRAPHITE DISPOSAL

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The paper considers the current state of activities and studies on the management of irradiated reactor graphite. It suggests a number of disposal cost optimization options, including the development and application of decontamination methods, reduction of graphite RW activity, reduction of RW conditioning costs, and establishment of a specialized disposal facility. The paper also discusses future prospects for arranging activities and studies on graphite disposal.

Keywords: *radioactive waste, radiocarbon, reactor graphite, uranium-graphite reactor, RW disposal, RW disposal facility.*

Graphite properties acting as a neutron moderator prompted its wide use in nuclear reactor construction, including uranium-graphite water- and gas-cooled nuclear plants: plutonium production, research and power nuclear reactors, as well as dual-purpose facilities. To date, operation of the graphite reactor fleet has been largely completed, although there are some ideas on building new reactors of this type.

Before addressing the topic of graphite RW (hereinafter, graphite) disposal in Russia, two issues should be considered in brief: relevant radiological aspect and international experience in managing such waste.

The radiological aspect is associated with a significant content of long-lived radionuclides with high migration ability within natural systems. These radionuclides include ^{14}C (with a half-life of 5,730 years) — its contribution to the activity of irradiated graphite exceeds 95 %, and ^{36}Cl with a lower activity level but much longer half-life of 301,000 years. A positive aspect under these conditions is that on a global scale the mechanisms associated with the accumulation of these radionuclides in particular components of biological environment, human and animal organs are practically absent in comparison, for example, with similar characteristics of iodine and strontium. In general, after being released into the atmosphere, gradual decrease in ^{14}C concentrations in air can be observed — up to background values in accordance with known patterns [1].

Availability of man-made ^{14}C in the biosphere is due to nuclear weapons tests. Since 1950s relevant issues have been considered with caution [2], mainly due to the possible growth of malignant neoplasms. Despite certain political incentives on conducting the first assessments [3], these concerns remain. Data on global exposure levels from ^{14}C are regularly summarized in the reports of the

UN Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). Thus, according to [4], exposure of population resulting from ^{14}C and associated with nuclear tests performed attained its peak 55 years ago (0.1 mSv/year), and currently does not exceed $5 \cdot 10^{-3}$ mSv/year. To put this in perspective, the global average annual dose for representative population groups residing in the vicinity of nuclear power plants is 50 times smaller than this value and tends to decrease. Another comparison based on the materials of this report is that the exposure doses of the population obtained as the result of medical treatment in 1997—2008 have increased by more than 70 %. The data presented indicate a positive trend, but they should not be considered as the basis for making simplified decisions on graphite management, involving radiocarbon release into the atmosphere [5]. Moreover, new data appear suggesting that radiocarbon has a specific devastating effect when it is incorporated into living cells [6].

The following comparison seems quite of interest in the context of the radiological assessment of accumulated graphite management challenges: an atmospheric nuclear test with a yield of 1 Mt generates approximately $8.6 \cdot 10^{14}$ Bq of radiocarbon [3]. Thus, as a result of ten most powerful nuclear tests with a total yield of about 200 Mt, the amount of radiocarbon released into the atmosphere equaled the total amount of it contained in all reactor graphite accumulated globally.

Global approaches to solving the challenge of irradiated graphite management

The largest amount of irradiated graphite was accumulated in the UK (86,000 tons), Russia (60,000 tons), the USA (55,0000 tons), France (23,000 tons), and a bit less in Ukraine (5,700 tons), Lithuania (3,800 tons), Spain (3,700 tons), North Korea (3,500 tons), Italy (3,000 tons), Japan (3,000 tons), Belgium (2,500 tons), Germany (2,000 tons). The overwhelming majority of "graphite" nuclear facilities are being considered within relevant decommissioning programs, and in many cases [7] a deferred decision was made suggesting their differed dismantlement with controlled maintenance for 80 or more years. The main reason is the lack of sound approaches for managing the reactor graphite and the need of reducing the activity of other isotopes, including ^{60}Co. In this case, both disposal and graphite processing options are being considered.

The number of foreign publications on the topic of processing (decontamination) is quite large. According to the SCOPUS bibliographic database — about ten publications are issued each year, according to Web of Science — a little less, and based on eLibrary data — some 2—4 per year. A wide range of technologies is being considered, including heat treatment (up to 1,400 °C [8]), high-temperature oxidation, acid treatment, graphite washing with detergents, molten salts, the use of spark erosion, plasmification, electrochemical methods, biological methods, as well as deactivated graphite recycling, etc. [9]. An important thing to note, is that until present time no progress giving reason for some optimism regarding the emergence of some technologies ensuring efficient graphite treatment was attained, namely in terms of avoiding ^{14}C releases into the atmosphere, reducing the amount of secondary solid radioactive waste, and generating waste in a form suitable for disposal.

International cooperation on reactor graphite is quite active — its activities resulted in such projects as: CARBOWASTE [9], GRAPA [10] and a number of others projects coordinated by the IAEA [11]. IAEA paper [12] states that a unified global strategy has not been adopted, but most countries intend to dispose graphite of in geological formations rather than deactivate it. To some extent, this decision is due to very slow progress in the development of an effective technology. Questions are still raised regarding the model of activity release from the irradiated graphite [13].

Incineration of irradiated graphite is associated with unacceptable consequences of ^{14}C release into the atmosphere. Separation of ^{14}C from combustion products is considered to be quite difficult and economically infeasible thing to do [14]. If all carbon dioxide (^{12}C and ^{14}C) is captured, this only leads to an increase in the waste amount. Thus, incineration of irradiated graphite with the entire volume of carbon dioxide being captured, for example, by slaked lime, would provide more than eight-fold increase in the mass of generated secondary radioactive waste.

A more optimistic outlook exists in terms of direct graphite disposal. It is postulated that graphite cannot be disposed of in near-surface storage facilities due to the long half-life of ^{14}C and ^{36}Cl. Siting efforts aimed at finding a suitable site even in France, being considered as a most experienced country in terms of RW disposal and having a significant reactor graphite inventory, have not been completed yet [15]. The developed disposal concept assumes the final isolation of such radioactive waste in clay formations with simultaneous disposal of radium-containing waste (at a depth of 15 m in case if exposed clay layers are available) and graphite radioactive waste (at a depth of 100—200 m with a clay layer thickness of at least 50 m) [16].

German experts are studying the prospects for graphite-containing waste disposal in KONRAD deep repository (CarboDISP project). Estimates show that the maximum total activity of ^{14}C contained in the waste that can be safely disposed of in KONRAD facility amounts to some $4 \cdot 10^{14}$ Bq. Taking into account the total useful capacity of the facility of 303,000 m^3, average specific activity of the waste disposed of there would account for $1.32 \cdot 10^9$ Bq/m^3 [17].

In the UK, it is proposed to excavate the reactor graphite disposal facility at an average depth (not less than 30 m from the ground surface). The concept being developed [18] includes a concrete shaft lined with concrete with a wall and base plate of reinforced concrete acting as elements of multi-barrier safety system. Cemented waste packed into containers will be emplaced into the shaft with the remaining gaps backfilled with cement mortar until a single monolith is produced. The overhead chamber is to be backfilled, thus, ensuring that the waste is separated by tens of meters from the ground surface level. A mound will be built on the surface reducing the amount of water infiltration into the repository. It is interesting to note that the siting efforts are to be focused on the coastal zone with groundwater discharge into the marine environment. A preliminary safety assessment of this concept has already been completed with relevant research activities still underway.

Characteristics of reactor graphite accumulated in Russia

Accumulation of irradiated reactor graphite in Russia is mainly associated with the operation of RBMK and PUGR reactors (Figure 1). Significantly smaller amounts of graphite RW were generated from the operation of the world's first nuclear power plant (JSC SSC RF-IPPE) and research facilities. The total amount of accumulated graphite accounts for some 57,000 tons [19], and the ratio of graphite stack material by mass is about three times greater than the one of sleeve-type graphite. The inventory data will be refined after the shutdown of all uranium-graphite reactors.

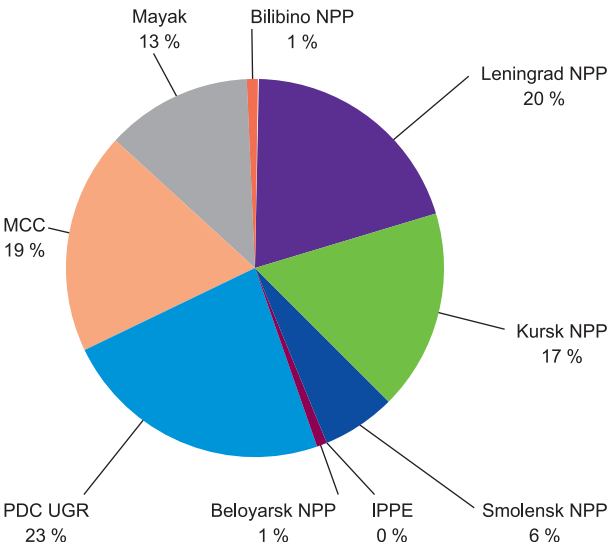

Figure 1. The amount of irradiated graphite accumulated at different sites, in % from the total mass

Since 2008, the Federal Targeted Program "Nuclear and Radiation Safety in 2008—2015" provided for a set of measures on uranium-graphite reactor (UGR) decommissioning. A special purpose organization, Pilot Demonstration Center for Uranium-Graphite Nuclear Reactor Decommissioning (JSC "PDC UGR"), is the linchpin of relevant activities performed. In addition, experts from the Institute of Physical Chemistry and Electrochemistry named after A.N.Frumkin (IEPhE RAS), the Research and Design Institute of Energy Engineering named after N.A.Dollezhal (JSC NIKIET) and a number of other organizations are also actively engaged in these activities. In 2014, Comprehensive Program on Managing Graphite from Uranium-Graphite Reactors at the Enterprises of the State Atomic Energy Corporation Rosatom (hereinafter referred to as, Program) was developed and approved.

Since that time, R&Ds were launched to support the following activities:

1. Certification of graphite;
2. Selection of irradiated graphite management option;
3. Development of requirements and selection of materials for protective safety barriers construction;
4. Development of a technology enabling to manage retrievable RW containing irradiated graphite;
5. Safety demonstration and long-term risk management.

Over the years, certain progress has been achieved in the implementation of the entombment option. In 2015, in-situ disposal of PUGR EI-2 graphite cladding was completed at PDC UGR site.

In keeping with the provisions of the Program, characterization of graphite radioactive waste was performed, studies were implemented to evaluate the properties of graphite and the mechanisms enabling to extract the radionuclides, to select buffer backfilling materials, to assess the stability PUGRs in-situ disposal. It was demonstrated that clay materials applied for backfilling purposes and performing a function of an anti-filtration-sorption-precipitation engineered barrier, can retain their physical and chemical properties for a long time. Thus, during the implementation of the pilot PUGR EI-2 decommissioning project, safety issues were studied

and justified, design solutions, methodological and technological approaches on work implementation were developed. This means that for the PUGR EI-2 the tasks stated under the first and second focus areas of the Program were successfully addressed. An experimental monitoring system [21] has been set being considered adequate only in terms of the first entombment project. It includes inspection channels enabling to monitor the changes in the water content and density of engineered barrier materials and the state of the surface screen [21], as well as a groundwater monitoring system. At present time it seems obvious that this system should be optimized, also in terms of accounting the change in the status of the facility. Two points are considered of importance when discussing similar activities on other PUGRs:

- Reference information on technological solutions and total costs per facility (about 2.5 billion rubles per facility) was obtained;
- Areas for further optimization of solutions suggesting the implementation of the entombment decommissioning option have been identified [22].

The amount of information on radiation characteristics of PUGR graphite cladding has significantly increased (Program area 1). Complex nature of graphite cladding radioactive contamination is mostly due to specific design features of different PUGRs, applied design solutions, different incidents occurred, conditions and duration of operation. Graphite contains some radionuclides produced as a result of activation with ^{14}C and ^{36}Cl being considered as the main ones, as well as fission products (^{137}Cs, ^{134}Cs, ^{60}Co and ^{90}Sr) and actinides (Am, U, Pu, ^{244}Cm, ^{237}Np isotopes) [23—26].

It should be noted that activation of surrounding massive metal structures results in a significant content of gamma-emitting isotopes (mainly ^{60}Co, ^{94}Nb, ^{152}Eu, etc.). Thus, at the time of reactor facility final shutdown, high dose rates of ionizing radiation (up to 10 Sv/h and higher) are observed making it necessary to apply more complex technologies for dismantlement and graphite extraction. Based on the studies involving graphite samples extracted from stacks and bushings, it was found that specific activity of ^{14}C and ^{36}Cl in the graphite of domestic PUGRs amounts to 10^3—$2{\cdot}10^6$ Bq/g and 10—$2{\cdot}10^4$ Bq/g, respectively [31]. Furthermore, the specific activity of ^{14}C in core graphite, including interchangeable parts (bushings, solid graphite rings, control rod followers), amounts to over 10^4 Bq/g. It should be noted that the specific activity of ^{14}C for PUGR and RBMK-1000 graphite given similar operating periods in the plane of the neutron fluxes maximum, do not differ fundamentally. In case of RBMK-1000, ^{14}C activity is 1.7—2 times lower as compared to the PUGR's one due to the lower contribution of ^{14}C to the formation of gaseous nitrogen contained in the gas purge mixture (for PUGR — pure N_2, for RBMK — N_2-He mixture). About 5% of graphite, mainly originating from PUGRs, contains a significant amount of "fuel" origin isotopes (Am, U, Pu, Cm, Np).

Complete data on the content of ^{14}C and actinides in reflectors are currently lacking. Experimental data obtained for one of the PUGRs showed that the specific activity of ^{14}C in samples taken from side reflector stack in the plane of maximum neutron flux exceeds 10^5 Bq/g even in the 3rd row of reflector columns (~ 60 cm from the core boundary).

On the whole, significant progress can be observed in this area, namely, in terms of better understanding of PUGR graphite cladding radiation characteristics with a significantly less progress achieved in terms of RBMK power reactors. The amount of data published on this topic is extremely limited. Information on AMB reactors is almost completely absent. Therefore, when assessing the radiation characteristics of graphite, authors refer to available calculated estimates made at the initial stage before real samples were studied [27]. Moreover, the available data show that the radiation-chemical characteristics of graphite can vary greatly depending on the nuclear unit considered and fundamentally differ from similar characteristics of PUGR graphite. Despite the fact that most of these reactors are still in operation, relevant arrangements providing for the implementation of preliminary studies should be started already to assess the volumes of graphite RW planned for disposal and to solve relevant challenges.

Under the second focus area of the program, comprehensive surveys (KIRO) of graphite claddings were performed by JSC PDC UGR experts with sampling procedure for irradiated graphite enabling its further study being developed. Based on experimental data and KIRO, as well as PUGR operational history, reactor graphite inventory was categorized according to RW classes: 1st class — 1—2%; 2nd class — 81—88%; 3rd class — 11—17%; 4th class — less than 1%. It was assumed that mass ratio for side reflectors ranged from 22 to 32% from the total mass of graphite cladding. Based on these estimates, it can be concluded that 82—90% of the graphite RW from Russian UGRs shall be subject to deep disposal (RW classes 1 and 2).

These estimates, given a certain degree of approximation, can be extended to RBMK-type reactor units that are to be decommissioned based on the dismantlement option. Given the voids and available package sizes, this part of graphite RW with an estimated total mass of 26 thousand tons can

correspond to a volume of disposed RW amounting to some 25—40 thousand m^3. Based on these conclusions, certain assumptions can be made regarding the initially estimated cost associated with the disposal of retrieved graphite assuming the use of NZK-150-1 5P containers with SM-500 insert for reactor graphite (the cost of one container was estimated being equal to some 225 thousand rubles given the prices of late 2017 with a packaging cost of some 60 thousand rubles). The number of containers was calculated based on the assumption that containers were almost full. It seems quite interesting to compare such estimates for two options: entombment with the implementation of activities similar to those performed under the EI-2 decommissioning project and retrieval. In the first case, the total costs amounted to some 2.5 billion rubles (excluding the full cost of UGR dismantlement). In the second, graphite disposal cost alone would amount to some 6.9 billion rubles with an estimated total cost of some 8 billion rubles. Thus, it appears to result in more than three-fold cost saving. The diagram (Figure 2) shows the distribution of the share of graphite RW from PUGRs and NPPs, based on the suggested methods of its management. The decision on the disposal method for graphite RW belonging to PUGR located at FSUE PA Mayak site, referred to as the zone of uncertainties and research, is considered to be of a fundamental nature. It enables to assess the opportunities for constructing geochemical barriers with minimal filtration coefficients. Bilibino NPP case is somewhat different — general decisions on SNF and RW management seem to be of great importance, including those associated with the long-term storage at the industrial site.

Referring to the option suggesting deep disposal of graphite RW, it should be noted that the approach

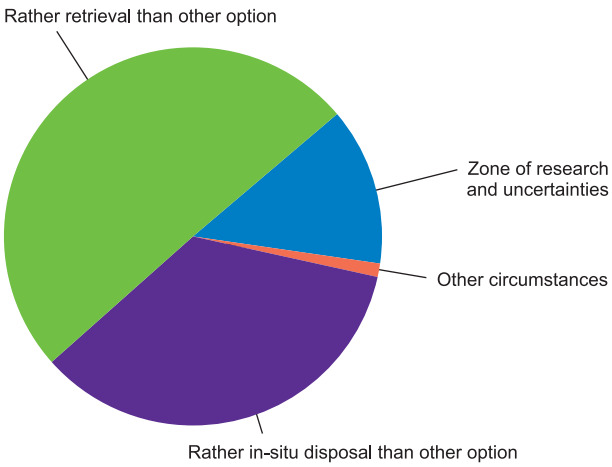

Figure 2. Break down of irradiated graphite inventory considering different management scenarios

applied in Russia seems to be fundamentally different from the solutions developed in France and Great Britain providing for the disposal of such RW at a shallower depth. Obviously, this is not due to particular hazard associated with the domestic graphite, but rather to the imperfection of the classification system in terms of this and other relevant aspects [28]. Yet another point to note is that the emergence of graphite RW of Class 1 is mainly due to the concentrations of individual nuclides. Thus, segregation of graphite waste streams into several different classes seems to be quite feasible if associated savings exceed the segregation costs.

Thus, current RW classification system literally ruls out the opportunities for making any moves allowing to optimize the graphite RW disposal concept. Given this fact, two options are possible. The first is to agree with the inevitable costs associated with the deep disposal of such RW, and the second, which seems to be more balanced, - to review the current classification system and to optimize it based on safety criteria, especially since problems with setting disposal classes are also encountered in other areas.

Today, a need has arose on discussing and adjusting the activities stated under the fourth focus area of the Program associated with graphite processing. The results obtained, both in Russia and abroad, do not give reason for any optimism in assessing the effectiveness of various engineering solutions. For example, UDC UGR experts proposed a “graphite evaporation” technology in a low-temperature plasma (when graphite is heated to over 3,000 °C) [29], which is also discussed in [30]. Experimental studies on further development of graphite incineration technology using pilot USG unit showed the possibility of its practical implementation. However, a significant (more than eight-fold) increase in the volume of secondary waste using the selected treatment system makes such processing economically infeasible.

Nevertheless, some data is available showing that at a significantly lower temperature (600 °C) thermal extraction of ^{14}C in an inert or slightly oxidizing medium allows further reduction in the rate of isotope leaching from graphite [31].

However, a neater statement of the final goal is required to assess the prospects for further development of the processing method.

If the goal is reduced to obtaining an RW matrix for disposal, then graphite can be considered as a readily available one and, moreover, the best-known matrix. The only possible exception is the graphite RW resulting from nuclear accidents containing some spills of fuel fragments and having high oxidation state due to relevant incidents.

Liquid and thermal deactivation of PUGR stack graphite results in up to 25 % extraction of ^{14}C during low temperature annealing or reagent treatment [32—34]. In this case, the mass loss of graphite is less than 10 %. This stage enables to remove some weakly bound ^{14}C atoms located on the surface. This method is not suitable for complete or significant ^{14}C extraction. However, in this case a significant fraction of radiocarbon being weakly bound in the graphite structure is released with a relatively low mass loss of graphite itself. It should be noted that it is this fraction of radiocarbon that is most susceptible to leaching when coming into contact with liquid media. Thus, by removing this part of ^{14}C using thermal methods, a significant increase in the resistance of irradiated graphite to such leaching can be achieved [31].

Thermal treatment [31] in an inert and slightly oxidizing medium, allows the extraction of up to 50—60 % of ^{14}C with a graphite mass loss of less than 3 %. At this stage, ^{14}C implanted into the crystal lattice as a result of the nuclear reaction $^{14}N(n, p)^{14}C$ with gaseous nitrogen is removed. At the last stage, the yield of ^{14}C exceeds 90 % with a proportionally decreasing mass of graphite.

Such dynamics is due to the fact that at the early stages of thermochemical treatment, graphite oxidation predominantly occurs along the (micro) cracks and other defects facilitating the removal of relatively weakly bound ^{14}C, for example, from the pore space. To remove ^{14}C from the crystal lattice, the graphite lattice must be destroyed, for example, by heat treatment at high temperatures.

General conclusions of the authors seem to be quite noteworthy. They state that the low leaching rates of radionuclides: (^{14}C—10^{-8}—10^{-7}, ^{36}Cl — 10^{-5}—10^{-4} g/cm^2·day) and the stability of physical and chemical properties of irradiated graphite allows it to consider as a matrix with radionuclides inclusion. To implement the option of near-surface disposal, additional studies are required to be done given geological and geochemical conditions at the proposed sites.

It should be noted that this conclusion is based on systematic studies of various options providing for graphite RW decontamination performed both by Russian [19] and international experts [7—12].

Thus, today all necessary prerequisites are in place allowing for the revitalization and rearranging the pre-disposal management of graphite based on the dismantlement option, followed by relevant assessment of options suggesting near-surface and deep disposal of graphite RW.

Activities stated under the fifth focus area of the Program for the period until 2021 did not provide for a wide range of siting efforts, although relevant tasks featured the following: development of criteria defining the conditions for graphite disposal (deep or surface) and justification studies of potentially suitable disposal sites.

To date, largely due to the deployment of in-depth studies aiming to demonstrate the long-term deep disposal safety [33], it is possible to set the task more clearly and, accordingly, adjust the content of activities suggested under the fifth focus area of the Program.

The rationale behind such an adjustment is based on three groups of circumstances:

- preliminary calculations of long-term consequences for three types of facilities: deep, near-surface and shallow disposal facility;
- cost estimates for graphite disposal;
- the need to formulate a target order for FSUE "NO RAO".

Isolation of radionuclides during the time period associated with their potential hazard in case of their disposal (near-surface or deep) is ensured, first of all, based on indicators characterizing the strength of radionuclides fixation in graphite itself, anti-migration and anti-filtration properties of natural and technogenic barriers (already existing and constructed ones) on the path of possible radionuclide release, as well as existing conditions (hydrogeological, climatic, demographic, etc.) at the site proposed for the construction of a particular facility. To demonstrate the safety and feasibility of any disposal facility, predictive assessments of possible radionuclide migration under the conditions considered as typical for the developed repository are to be done. But besides this, it is still necessary to analyze the consequences of the so-called FEPs (Features, Events and Processes). Such estimates, made in the first approximation, taking into account a suitable geological environment, revealed more than 20 FEPs with guaranteed negative result for a near-surface repository.

Based on scenario-based cost estimates, some research priorities can be identified. By structure, the total cost associated with graphite management in case of its retrieval, can be split into several components, including conditioning (container cost and packing cost), transportation to disposal site and disposal itself.

The transportation cost depends linearly on the average distance from the RW retrieval facilities to the disposal facilities and the tariff. Several possible sites for graphite disposal were considered, given that today it is mainly considered as RW of class 2: Zheleznogorsk (the average distance between the retrieval sites and the repository amounts to some 3.8 thousand km) and three regions in the European part of Russia. All of them seemed to be much

more attractive with the Obninsk region being considered as an optimal geometric center (with average distance from the retrieval sites accounting for some 0.9 thousand km). Given a conservative assumption suggesting a tariff of 20 rubles/(km·m^3), the total price results in almost ten billion rubles. Perhaps, the tariff can be also optimized resulting in several fold reductions. The second optimization component is the conditioning cost. Apparently, it can be reduced mainly by reducing the container cost. The most important component of optimization is the reduction of the disposal cost from the existing tariff for RW class 2 (636 thousand rubles/m^3) to some 250—350 thousand rubles/m^3 given that the waste is disposed of in a shallow repository at a depth of some 50—100 m. In this case final estimates can be reduced by about 2 times.

The first stages of USS RW development were associated with significant uncertainties regarding the requirements on the disposal system capacity. This resulted in the existing RW classification and disposal facilities being developed exclusively based on the disposal classes with all the resulting costs. Thus, attempts were made to demonstrate the long-term safety based on possible amounts of generated waste with relevant theoretical difficulties being encountered since a case suggesting the disposal of radioactive waste with activity indicators of all radionuclides corresponding to the upper activity limit for the class considered seems practically impossible.

At present, all conditions have been set to allow a shift providing for a totally new quality in this area, namely: planning for the development of disposal facilities designed for specific RW emplacement. The case of graphite waste disposal is considered as a most indicative one for the following reasons:

1. The base amount of graphite will start to be transferred for disposal in the next 20—30 years;

2. The amount of graphite subject to disposal amounts to some 30—40 thousand m^3 which is considered a quite sufficient amount deserving the development of a separate repository. In addition, other radioactive waste can be emplaced into the repository, namely those the disposal of which in a deep repository seems inadequate from a safety perspective whereas the near-surface disposal seems insufficient in the long run. This will provide a due increase in terms of personnel and equipment loading during the entire period of irradiated graphite emplacement into the repository (about 20 years) and beyond this period.

Taking into account the significant knowledge accumulated to date on PUGR graphite characteristics, as well as the timing for the siting stage, it seems possible to implement a set of studies on RBMK, AMB, EGP graphite characteristics in a timely manner and develop the repository design given a detailed inventory of RW with a disposal tariff amounting to some 250—300 thousand rubles/m^3. Russian studies and global experience will allow us to develop relevant siting terms of reference in a limited time.

Conclusion

Implementation of the industry-wide program on graphite allowed us to achieve a significant progress in obtaining reference data on the option of its in-situ disposal, as well as a large amount of data on its radiation characteristics. At the same time, it formed the basis for further progress and refinement of the research and development plan regarding graphite disposal. These initial data can be stated as follows:

1. A quite balanced approach is required to arrange the funding for the development (testing) of graphite decontamination methods in cases when these do not provide for stable and more compact matrices with decontamination waste compared to the original graphite.

2. Graphite itself is considered by far the best matrix for radioactive carbon. From radiation safety perspective, any operations associated with the destruction of graphite do more harm than good if they are not associated with nuclear safety.

3. To optimize the design solutions associated with pre-disposal treatment of reactor graphite, it is necessary:

a) To arrange targeted studies of graphite from all types of power reactor units (RBMK, EGP) and PA Mayak PUGR to identify a more precise distribution of ^{14}C and ^{36}Cl in the claddings, their integral amount, and the release rates from the matrix during the period associated with the potential hazard of such waste;

b) To elaborate on the RW classification system for disposal with relevant estimates on the distribution of graphite accounting for the "newly established classes", including the in-situ disposal option;

c) To search for and develop effective packaging solutions and to optimize the certification procedures. In the future, simplified certification procedures may be introduced, also by means of adjusting the regulatory framework for the purpose of establishing a purpose-designed disposal facility.

4. Decisions are required on the management of graphite RW: disposal in deep, shallow repository or in-situ disposal. As for FSUE PA Mayak facilities, a more in-depth consideration of an option suggesting the construction of impermeable barriers seems reasonable and necessary.

5. The safety of graphite disposal can be ensured both in case of deep and near-surface disposal of such waste. In the former case, it will be much more expensive, also due to significant transportation costs.

6. Terms of reference are to be developed for the siting efforts aimed at finding a suitable site for a shallow type repository.

7. All RW considered as unsuitable for their disposal in near-surface repositories are to be studied and analyzed to identify their amounts and to account them for during the siting efforts mentioned under pp. 5.

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