Optimization of RW volumes from reprocessing of SNF from fast reactors. Fractionation options

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Abstract. In this work, are analyzed the volume of vitrified radioactive waste generated during the processing of spent nuclear fuel from fast reactors (using the example of processing mixed uranium-plutonium nitride spent nuclear fuel from the BREST-OD-300 reactor). The analysis showed:

- there is a minimum acceptable volume of HLW from spent nuclear fuel reprocessing, determined by the parameters of permissible thermal and radiation effects on the matrices used for RW utilization, as well as by the restriction on the chemical composition of the matrices, fixed in regulatory documents;
- the allocation of the fraction of minor actinides (MA) during the reprocessing of the fast reactor spent nuclear fuel significantly reduces the required volume of the matrix for disposal of HLW;
- the currently technically achievable level of MA (99.9%) and plutonium (99.975%) recovery during the reprocessing SNF allows reducing the time to achieve radiation equivalence of the generated RW and uranium feed used to produce mixed nitride uranium-plutonium nuclear fuel to 300 years;
- a complete rejection of the deep underground disposal of RW from SNF reprocessing (rejection of the formation of radioactive waste of the 1st and 2nd classes) is not economically justified, because leads to the need for the formation of a large volume of RW of the 3rd class;
- minimization of the RW volume to be deep disposed is achieved by a combination of fractionation (separation of the MA fraction) and the use of the principle of multi-stage processing of HLW, which assumes long-term intermediate controlled storage of HLW (50-60 years).

1. Introduction
One of the main regulatory documents regulating the amount of generated RW during the reprocessing of spent nuclear fuel is Federal Rules and Regulations for the Use of AE “Collection, Reprocessing, Storage and Conditioning of LRW. Safety Requirements” (NP-019-15) [1], approved on June 25, 2015 by Order No. 242 of the Federal Environmental, Industrial and Nuclear Supervision Service of Russia. When selecting borosilicate glass as a matrix material for conditioned HLW, one should be governed by the changes to NP-019-15 described in [2], which give the basic quality indicators of glass-like borosilicate compound containing RW.

Table 1 shows basic permissible characteristics of the borosilicate glass compound where it satisfies the regulatory quality requirements.
Table 1. Basic quality indicators of the borosilicate compound.

| Parameter                              | Value                        |
|----------------------------------------|------------------------------|
| Content of fission products oxides     | Not more than 10% by weight  |
| Weight content of trans-uranium elements | Not more than 0.6%            |
| Weight content of α-emitters           | Not more than 0.6%            |
| Threshold of β-, γ-radiation           | $10^9$ Gy                    |
| Threshold of α-radiation               | $10^{18} - 10^{19}$ α-decays/cm$^3$ |

Limitations in table 1 on the specific content of radionuclides with borosilicate glass compound result in the limitation of the amount of vitrified RW produced during the reprocessing of SNF, i.e. there is some minimum permissible amount of glass per 1 kg of reprocessed SNF that meets the regulatory requirements [1, 2]. Obviously, this minimum permissible amount depends on the initial isotopic composition of SNF reprocessed, on the SNF holding time prior to reprocessing, on the reprocessing method (technology) - on the purity of separation of the target fuel components during SNF reprocessing; on the degree of separation for intermediate storage or destruction (afterburning) of some fractions of isotopes present in the SNF.

2. Amounts of vitrified HLW in the reprocessing of MNUP SNF of fast reactors

Let us consider the required amounts of borosilicate glass compound (BSG) generated during the reprocessing of MNUP SNF irradiated in the BREST-OD-300 reactor unit with an average burnout of 8% and post-irradiation holding for 2 years. Isotopic composition of the above SNF is given in [3].

Radioactive isotopes in the SNF may be classified into 4 groups:

- Target products (fuel components recovered from spent fuel reprocessing);
- Minor actinides;
- Fission products;
- SFA (spent fuel assemblies) structural materials.

Let us examine separately each group of isotopes and determine the minimum volume of the matrix necessary for its utilization. It is proposed to use a borosilicate glass compound as a matrix for disposal of high-level radioactive waste, except for 14C disposal (which is immobilized using a cement matrix, since high-temperature reprocessing methods are not applicable to 14C disposal) and structural materials of irradiated fuel elements and SFA (which, as a result of reprocessing by melting or pressing, form a metal matrix).

2.1. Residues of target products under-recovered in the reprocessing of MNUP SNF

The target fuel components in the reprocessing of MNUP YAT determined by the upper level technical specifications of the Proryv project direction include U, Pu and Np isotopes. The target recovery rate for fuel components from SNF is 99.9%. The actual recovery rate of fuel components that is technically achievable with modern technologies is higher: from nitrate acid solutions of SNF containing heavy metals (uranium and plutonium) coming for hydrometallurgical reprocessing - 300 g/l (ratio of plutonium to uranium – 3:20, that is, a solution contains about 40 g/l of plutonium and 260 g/l of uranium) uranium and plutonium are recovered to levels of 1 mg/l. That is, plutonium is extracted to the level of 2.5E-05, and that of uranium is 4.0E-06.

Table 2 shows the list of U, Pu and Np isotopes present in MNUP SNF and an estimate of the required minimum glass volume for disposal of each isotope, based on the stability requirements of the resulting glass-like matrix (number of α-decays/s - 5.0E + 18; weight content of α-emitters - no more than 0.6%). From table 2 it follows that the minimum possible volume of glass-like matrix for the disposal of under-recovered fuel components during the reprocessing of SNF is about 0.0013 l BSG/kg of SNF and is limited by the amount of α-decays per unit volume of BSG.
2.2. **Inclusion of minor actinides in the BSG**

Upper level technical specifications of the Proryv Project provide for the recovery of MA during the reprocessing of MNUP SNF to the level of 99.90%, the separation of americium and curium and taking:

- americium to afterburning by homogeneous inclusion in the fuel composition (or fabrication of oxide uranium-americium fuel elements);
- curium (in oxide form mixed with uranium) to long-term controlled storage for the purpose of subsequent recovery of plutonium that accumulates as a result of disintegration of curium isotopes.

Let us examine the minimum permissible amount of BSG required for the disposal of MA under recovered during reprocessing of MNUP SNF. Data on the amount of MA in MNUP SNF and evaluation of the required amount of BSG are given in table 3. Analysis of the data shows that for disposal of MA under recovered from RW, an insignificant amount of BSG (less than 1 cm³ BSG per kg of SNF) is required, which is determined by the limitation on the amount of α-decays per unit volume of the matrix.

If MA are not recovered from RW, it will lead to an increase in the matrix volume for their disposal. According to table 3, the required volume of BSG is 0.7 l per kg of SNF.

**Table 2.** Content of U, Pu and Np isotopes in the waste from MNUP SNF reprocessing (average burnout is 8%, and the post-irradiation holding is 2 years) and an estimate of the minimum volume of BSG for their disposal.

| Isotope     | g/kg of SNF | Half-life, years | g/kg of SNF in RW | α-activity of RW | Estimated required volume of BSG, l/kg of SNF |
|-------------|-------------|------------------|-------------------|----------------|---------------------------------------------|
| U-235       | 0.473       | 7.04E+08         | 1.89E-06          | 1.51E-01       | 0                                           |
| U-236       | 0.248       | 2.40E+07         | 9.92E-07          | 2.38E+00       | 0                                           |
| U-238       | 696         | 4.47E+09         | 2.78E-04          | 3.46E+01       | 0                                           |
| Np-237      | 0.359       | 2.14E+06         | 1.44E-06          | 3.73E+01       | 0                                           |
| Pu-238      | 0.947       | 87.7             | 2.37E-05          | 1.50E+07       | 0.83E-05                                    |
| (daughter U-234) | (2.48E+05) | 5.48E+03         |                   | 0              |                                             |
| Pu-239      | 86.9        | 2.40E+04         | 2.17E-03          | 4.98E+06       | 3.13E-04                                    |
| Pu-240      | 34.2        | 6.60E+04         | 8.55E-04          | 7.18E+06       | 4.55E-04                                    |
| Pu-241      | 4.72        | 14.3             | 1.18E-04          | -              | -                                           |
| (daughter Am-241) | (432.6) |                   | 1.50E+08          | 4.08E-04       | 7.28E-06                                   |
| (daughter Np-237) | (2.14E+06) |                   | 3.08E+04          | 0              |                                             |
| Pu-242      | 4.45        | 3.87E+05         | 1.11E-04          | 1.62E+04       | 0                                           |
| (daughter Np-237) | (2.14E+06) |                   |                   | 0              | 6.85E-06                                   |
| **Total**   |             |                  | **1.26E-03**      | **3.72E-04**   |                                             |

2.3. **SNF fission products**

The main limiting factor in determining the minimum permissible volume of BSG in the disposal of fission products of MNUP SNF is the dose accumulated in the weight unit of the BSG matrix - $10^9$ Gray [2]. Since the main value of the absorbed dose (up to 95%) is related to the absorption of the radiation of $^{90}$Sr and $^{137}$Cs isotopes by the borosilicate matrix, we estimate the required volume of BSG due to the absorption of the radiation of these isotopes (table 4). Table 4 shows that the minimum
The permissible volume of BSG determined by the fission products of SNF is 3.04 kg of the matrix or 1.13 liters of BSG per kg of SNF.

A specific product of MNUP SNF irradiation in a reactor facility is $^{14}$C. In the hydrometallurgical reprocessing of MNUP SNF (during the operation of voloxidation) $^{14}$C goes into a gaseous phase in the form of carbon dioxide. As a result of its alkaline capture and transformation into a form suitable for burial, a calcium carbonate-based pulp is generated, which is subject to cementing. The permissible $^{14}$C content in the RW matrix is determined based on the radiation resistance of the cement matrix [1] and the permissible activity level of the RW being reprocessed in the cementing facility and amounts to $10^{10}$ Bq/kg of cement matrix, which corresponds to the weight of the cement matrix 4.6 kg/kg of SNF or 2.3 l/kg of SNF (2nd class RW). Currently, R&D is being conducted to substantiate a more compact arrangement of $^{14}$C in RW matrices (substantiation of the properties of matrices made of magnesium-phosphate ceramics to allow for $^{14}$C with specific activity over $10^{10}$ Bq/kg).

Table 3. MA content in the waste from reprocessing of MNUP SNF (average burnout of 8%, and post-irradiation holding is 2 years) and an estimate of the minimum volume of BSG for their disposal.

| Isotope       | g/kg of SNF | Half-life, years | g/kg of SNF in RW | $\alpha$-activity of RW, (transfer share in RW: $2.5E-05$ by $^{244}$Pu; $4.0E-06$ by $^{243}$U and $^{237}$Np) | Estimated required volume of BSG, l/kg of SNF | 0.6 weight% of $\alpha$-emitters or TUE |
|---------------|-------------|------------------|-------------------|-----------------------------------------------|---------------------------------------------|------------------------------------------|
| Am-241 (daughter Np-237) | 1.35E-00   | 430              | 1.35E-03          | 1.71E+08                                      | 4.62E-04                                   | 8.33E-05                                 |
| Am-242m (daughter Pu-242) | 6.03E-02   | (2.14E+06)       | 6.03E-05          | 3.52E+04                                      | 2.23E-06                                   | 3.72E-06                                 |
| Am-243         | 7.42E-01   | (3.87E+05)       | 7.42E-04          | 5.48E+06                                      | 2.55E-06                                   | 4.58E-05                                 |
| Cm-242 (daughter Pu-238) | 9.44E-04   | 7.4E+03           | 9.44E-07          | 1.16E+08                                      | 0                                           | 5.85E-08                                 |
| Cm-243 (daughter U-234) | (87)       | 1.50E+00           | 1.35E+00          | 5.97E+05                                      | 3.27E-07                                   |                                          |
| Cm-244 (daughter Pu-239) | 6.07E-03   | (2.48E+05)       | 6.07E-06          | 1.14E+07                                      | 2.01E-06                                   | 3.75E-07                                 |
| Cm-244 (daughter Pu-240) | (2.4E+04)  | (3.87E+05)       | 2.53E-04          | 2.45E+02                                      | 1.54E-08                                   |                                          |
| Cm-245         | 7.28E-03   | (6.6E+04)         | 7.28E-06          | 2.13E+00                                      | 8.64E-05                                   | 1.56E-05                                 |
| Total          |             |                  |                   |                                               | 4.92E-08                                   | 1.50E-04                                 |

2.4. Structural materials of irradiated fuel elements and SFA

EP-823 steel of martensite-ferrite class is used as a structural material for the manufacture of fuel elements and fuel assemblies of the BREST-OD-300 reactor plant. The compositions of steel and induced activity during irradiation in the reactor are given in paper [4]. The main finding is that the amount of metal (EP-823 steel) activated in the core of the BREST-OD-300 reactor is approximately 1 kg per kg of SNF (disregarding the top end fittings and bottom nozzles of fuel assemblies that are located outside the reactor core and during the reactor life gain activity by 1-2 orders of magnitude lower than shells of the fuel elements and structural materials of SFAs located in the reactor core.

Table 5 presents data on the activity of long-lived (half-life of more than 31 years) isotopes accumulated in the structural materials of fuel elements and SFAs.
Table 4. Content of main dose-generating fission products in MNUP SNF (average burnout 8%, with post-irradiation holding of 2 years) and an estimate of the minimum volume of BSG for their disposal.

| Isotope | g/kg of SNF | Half-life, years | Activity, Bq/kg of SNF | Radiation energy absorbed per unit weight of the matrix, Gy/kg of SNF (during 1000 years) | Minimum permissible weight of the matrix, kg of matrix/kg of SNF | Estimated required volume of BSG, l/kg of SNF |
|---------|-------------|------------------|------------------------|-------------------------------------------------------------------------------------|------------------------------------------------------------------|---------------------------------------------|
| 90Sr    | 0.84        | 28.5             | 4.31E+12               | 1.02E+09                                                                            | 1.02                                                             | 0.38                                        |
| 137Cs   | 3.55        | 30.2             | 1.14E+13               | 2.02E+09                                                                            | 2.02                                                             | 0.75                                        |
| Total   |             |                  |                        |                                                                                     |                                                                  | 3.04                                        |

Thus, structural materials of fuel elements and SFAs irradiated in the reactor core of BREST-OD-300 are classified as 1st class RW and are subject to deep geological burial. This radioactive waste is processed (compacted) by melting or pressing. The final amount of processed metal RW is 0.15-0.20 l/kg of SNF.

Table 5. Specific activity of structural materials of fuel elements and SFAs (EP-823 steel) due to long-lived (half-life more than 31 years) isotopes.

| Alloy grade | Long-lived isotope (half-life is shown in brackets), activity, Bq/kg of metal | Total induced specific activity of long-lived isotopes, Bq/kg of metal |
|-------------|--------------------------------------------------------------------------------|---------------------------------------------------------------------|
|             | C-14 (5730 years) | Ni-63 (96 years) | Mo-93 (4000 years) | Nb-94 (20 thousand years) | Ni-59 (76 thousand years) | Tc-99 (212 thousand years) |                                      |
| EP-823      | 6.0E+08            | 8.2E+09          | 1.4E+09            | 2.0E+09                  | 1.4E+08                  | 4.7E+07                  | 1.2E+10                                |

2.5. Summary of RW from reprocessing of MNUP SNF in BREST-OD-300 reactor

Summarizing the data of sections 1.1-1.5, it is possible to determine the total amount of RW from reprocessing of MNUP SNF of the BREST-OD-300 reactor (average burnout is 8%, and the post-irradiation holding is 2 years) subject to deep burial. The result is given in table 6.

3. Radiation equivalence between the generated radioactive waste and uranium raw materials used to fabricate nuclear fuel.

One of the basic requirements of the Proryv Project applicable to the generated RW is the requirement to minimize the time to achieve radiation equivalence between the RW generated by the reprocessing of SNF (and to be buried) and raw materials extracted from the subsoil to fabricate nuclear fuel. Equivalence is considered in terms of negative impacts on the environment.
Table 6. Data on the amounts of RW from reprocessing of MNUP SNF of the BREST-OD-300 reactor (average burnout 8%, with post-irradiation holding of 2 years) subject to deep burial (without regard to the volume of packages)

| RW source                                               | RW category | Matrix | RW amount, l/kg of SNF |
|---------------------------------------------------------|-------------|--------|------------------------|
| Residues of target products (fuel components under-     | 1st class   | BCG    | 0.0013                 |
| recovered during reprocessing)                          |             |        |                        |
| Minor actinides under-recovered during SNF reprocessing | 1st class   | BCG    | 0.0007                 |
| Fission products                                        | 1st class   | BCG    | 1.13                   |
|                                                         | 2nd class (C-14) | cement | 2.30                   |
| Structural materials of fuel elements and SFA           | 1st class   | metal  | 0.20                   |
|                                                         |             |        | **Total**              |
|                                                         |             |        | **3.63**               |

From the point of view of modern classification of RW which takes into account the negative impact of ore on the environment, the dumps of uranium ore are separately allocated into a separate class —6th class RW [5] and allow near-surface burial without the use of additional packages (engineering barriers). Only hollows of ore deposits formed during the extraction of ore are used as engineering barriers restricting the spread of radionuclides from the uranium ore dumps.

RW from the reprocessing of SNF hypothetically may also be in a form that allows for their near-surface burial similar to the uranium ore in terms of safety requirements. Such RW are classified as the 4th class [5] and are characterized by the following parameters:

- they contain low-level radioactive waste, including radionuclides with a half-life of up to 31 years with specific activity of:
  - $10^6$ to $10^7$ Bq/kg - for radioactive waste containing beta-emitting radionuclides (except for tritium);
  - $10^5$ to $10^6$ Bq/kg - for radioactive waste containing alpha-emitting radionuclides (except for transuraniums);
  - $10^4$ to $10^5$ Bq/kg - for radioactive waste containing transuranic radionuclides;
- they contain very low level radioactive waste with a half-life of more than 31 years containing radionuclides with specific activity of:
  - up to 106 Bq/kg - for radioactive waste containing beta-emitting radionuclides (except for tritium);
  - up to 105 Bq/kg - for radioactive waste containing alpha-emitting radionuclides (except for transuraniums);
  - up to 104 Bq/kg - for radioactive waste containing transuranic radionuclides;

As an indirect confirmation of the achievement (or absence) of radiation equivalence, let us estimate the amount (weight) of the generated RW during MNUP SNF reprocessing, if these RW are classified as 4th class, and compare it with the amount (weight) of uranium ore required for the production of nuclear fuel.

Considering (estimated) uranium content in uranium ore at the level of 0.1%, we may assume that 1 ton of uranium ore should be processed to extract 1 kg of natural uranium. 1 kg of natural uranium contains 7 g of U-235. About 50 g of U-235 is required to produce 10 g of plutonium in a VVER-1000 reactor, therefore, 625 g of U-235 or about 90 kg of natural uranium are required to produce 1 kg of MNUP nuclear fuel (to produce 125 g of plutonium). In further estimates, we will use a rounded value
- 100 kg of natural uranium or 100 tons of uranium ore for the production of 1 kg of MNUP nuclear fuel.

An important factor is the time to achieve radiation equivalence. We will consider the characteristics of the generated radioactive waste 300 years after removal from the reactor facility.

3.1. Fuel components under-recovered during MNUP SNF reprocessing
Table 7 gives data on the activity of fuel components under-recovered during MNUP SNF reprocessing after 300 years of storage and weight estimation of the 4th class RW which is generated on their basis.

3.2. MA under-recovered during MNUP SNF reprocessing
Table 8 gives data on the activity of MA under-recovered during MNUP SNF reprocessing after 300 years of storage and weight estimation of the 4th class RW which is generated on their basis.

3.3. Content of basic fission products in MNUP SNF affecting the weight of the 4th class RW matrix
Table 9 gives data on the activity of fission products after 300 years of storage and weight estimation of the 4th class RW generated on their basis.

3.4. Weight of the 4th class matrix for structural materials
Based on the data in table 5, it can be concluded that the matrix of the 4th class RW for the structural materials of irradiated fuel elements and SFAs (after 300 years of storage) should be at least 5.0E+03 kg/kg of SNF.

Table 10 shows data on the minimum weight of the 4th class RW from various components of MNUP SNF reprocessing.

The data in table 10 show that after 300 years of storage RW from reprocessing of MNUP SNF (subject to recovery of fuel components, in particular plutonium, to the level of 2.5E-05, and MA to the level of 1.0E-03) reach equivalence with uranium extracted from subsoil for fabrication of MNUP SNF.

Table 7. Estimated weight of the 4th class RW based on U, Pu and Np isotopes from reprocessing of MNUP SNF (average burnout - 8%), RW storage period - 300 years.

| Isotope       | g/kg of SNF | Half-life, years | g/kg of SNF in RW (transfer share in RW: 2.5E-05 by Pu; 4.0E-06 by U and Np) | a-activity of RW after 300 years of storage Bq/kg of SNF | Estimated required weight of the 4th class RW matrix, kg/kg of SNF |
|---------------|-------------|-----------------|--------------------------------------------------------------------------------|----------------------------------------------------------|------------------------------------------------------------------|
| U-235         | 0.473       | 7.04E+08        | 1.89E-06                                                                      | 1.51E-01                                                 | 1.51E-06                                                         |
| U-236         | 0.248       | 2.40E+07        | 9.92E-07                                                                      | 2.38E+00                                                 | 2.38E-05                                                         |
| U-238         | 696         | 4.47E+09        | 2.78E-04                                                                      | 3.46E+01                                                 | 3.46E-04                                                         |
| Np-237        | 0.359       | 2.14E+06        | 1.44E-06                                                                      | 3.73E+01                                                 | 3.73E-03                                                         |
| Pu-238 (daughter U) | 0.947 | 87.7         | 2.37E-05                                                                      | 1.59E+06                                                 | 1.59E+02                                                         |
| Pu-239        | 86.9        | 2.40E+04        | 2.17E-03                                                                      | 4.98E+06                                                 | 4.98E+02                                                         |
| Pu-240        | 34.2        | 6.60E+04        | 8.55E-04                                                                      | 7.18E+06                                                 | 7.18E+02                                                         |
| Pu-241 (daughter Am-241) | 4.72 | 14.3         | 1.18E-04                                                                      | 9.27E+07                                                 | 9.27E+03                                                         |
| Pu-242 (daughter Np-237) | 4.45 | 3.87E+05     | 1.11E-04                                                                      | 1.62E+04                                                 | 1.62E+00                                                         |
| Total         |             |                 |                                                                              | 1.07E+04                                                 |                                                                  |
Table 8. Estimated weight of the 4th class RW based on U, Pu and Np isotopes from reprocessing of MNUP SNF (average burnout - 8%), RW storage period - 300 years.

| Isotope | Isotope group | Half-life, years | g/kg of SNF in RW | g/kg of SNF in RW (transfer share in RW - 1.0E-03) | α-activity after 300 years of storage, Bq/kg of SNF in RW | Estimated required weight of the 4th class RW matrix, kg/kg of SNF |
|---------|---------------|------------------|------------------|-----------------------------------------------|-----------------------------------------------|-------------------------------------------------|
| Am-241  |               | 1.35E-00         | 430              | 1.35E-03                                       | 1.05E+08                                       | 1.05E+04                                       |
|         | (daughter Np-237) | (2.14E+06)       | 141              | 6.03E-05                                       | 3.52E+04                                       | 3.52E+00                                       |
| Am-242m |               | 6.03E-02         | 3.87E+05         | 8.52E+03                                       | -                                              | -                                              |
| Am-243  |               | 7.42E-01         | 7.42E-04         | 5.48E+06                                       | 5.48E+06                                       | 5.48E+02                                       |
| Cm-242  |               | 9.44E-04         | 4.44             | 9.44E-07                                       | 0                                              | 0                                              |
|         | (daughter U-234) | (87)             | 6.07E-03         | 9.04E+03                                       | 2.45E+02                                       | 2.45E+02                                       |
| Cm-243  |               | 6.07E-03         | 29.1             | 6.07E-06                                       | 9.04E+03                                       | 9.04E+02                                       |
|         | (daughter Pu-238) | (2.48E+05)       | 1.24E+08         | 1.24E+02                                       | -                                              | -                                              |
|         | (daughter Pu-239) | (2.4E+04)        | 1.40E+04         | 1.40E+00                                       | -                                              | -                                              |
| Cm-244  |               | 2.53E-01         | 18.1             | 2.53E-04                                       | 7.79E+03                                       | 7.79E+02                                       |
|         | (daughter Pu-240) | (6.6E+04)        | 2.13E+06         | 2.13E+02                                       | -                                              | -                                              |
| Cm-245  |               | 7.28E-03         | 8.5E+04          | 7.28E-06                                       | 4.63E+04                                       | 4.63E+00                                       |
|         |               |                  |                  |                                               | Total                                          | 1.13E+04                                       |

Table 9. Estimated weight of the 4th class RW based on fission products, MNUP SNF (average burnout - 8%), RW storage period - 300 years.

| Isotope group | Half-life, years | Activity after 300 years of holding, Bq/kg of SNF | Estimated required weight of the 4th class RW matrix, kg/kg of SNF |
|---------------|------------------|---------------------------------------------------|---------------------------------------------------------------|
| Cs-137+Sr-90  | About 30 years   | 3.07E+10                                          | 3.07E+03                                                      |
| Cs-135        | 2.3E+06          | 1.24E+08                                          | 1.24E+02                                                      |
| C-14          | 5.7E+03          | 4.62E+10                                          | 4.62E+04                                                      |
| Se-79         | 6.5E+04          | 2.28E+07                                          | 2.28E+01                                                      |
| Zr-93         | 1.5E+06          | 1.03E+08                                          | 1.03E+02                                                      |
| Te-99         | 2.1E+05          | 1.04E+09                                          | 1.04E+03                                                      |
| Pd-107        | 6.5E+06          | 1.68E+07                                          | 1.68E+01                                                      |
| Sn-126        | 1.0E+05          | 9.67E+07                                          | 9.67E+01                                                      |
| I-129         | 1.5E+07          | 3.57E+06                                          | 3.57E+00                                                      |
| Sm-151        | 87               | 1.90E+10                                          | 1.90E+04                                                      |
| TOTAL         |                  | Total                                              | 6.95E+04                                                      |

Table 10. Required weight of the 4th class RW from various components of 4th class RW generated during reprocessing of MNUP SNF (average burnout - 8%), RW storage period - 300 years.

| RW source | Weight of the 4th class RW, t/kg of SNF |
|-----------|---------------------------------------|
| Residues of target products (recovery rate of plutonium during reprocessing - 99.9975%) | 11 |
| Residual minor actinides (recovery rate during reprocessing – 99.90%) | 11 |
| Fission products | 70 |
| Structural materials of fuel elements and SFAs | 5 |
| TOTAL | 97 |
| Weight of uranium ore to produce 1 kg of MNUP SNF | 100 |
4. Another section of your paper

Today there are several approaches to the fractionation of HLW:

- **Without fractionation** - all nuclides except for uranium and plutonium are solidificated into a single matrix;
- **“Proryv”** - recovery of actinides from RW by 99.9% (U, Pu, Np, Am, and Cm) and solidification of all other fission products, except for volatile and structural materials of irradiated fuel elements and SFAs, into borosilicate glass (1 class RW);
- **“Proryv +”** - the “Proryv” approach is supplemented with vitrification of fission products in 2 stages: initially into glass-like granulate with exceeding the standards for the content of fission products, followed by reprocessing the granulate into a monolithic matrix after 55-60 years of storage (during this time 3/4 of the dose from Cs, Sr and short-lived fission products will be absorbed) [5];
- **“Fractionation of MA + fission products”** - recovery of actinides from RW at 99.9% to 99.99% level (Pu, Np, Am into fuel, and Cm after 70 years of intermediate storage), separation of the Cs-Sr fraction for solidification into the matrix as III class RW (after 300 years after reprocessing) and solidification of all other fission products in I class RW.
- **“Abandon of I Class”** - recovery of all actinides followed by afterburning of MA (transmutation) and solidification of residues in RW (Cs, Sr and short-lived fission products) into the matrix for long-term storage (300 years) and transfer to Class III RW (no underground burial of RW);

Subsequent estimates are based on the following assumptions:

- calculations are made for reprocessing MNUP SNF in BREST reactor with a burnout of 8% and post-irradiation holding of 2 years;
- carbon is considered as a separate element released into the gaseous phase and constituting a separate type of radioactive waste;
- calculations were performed in accordance with NP-019-15 and annex to NP-019-15.
- Estimates of the amount of RW in various fractionation options, as well as estimates of transferring RW to the national operator are given in table 11.

Table 11. Required weight of the 4th class RW from various components of 4th class RW generated during reprocessing of MNUP SNF (average burnout - 8%), RW storage period - 300 years.

| Option                        | Fractions                  | Cost of burial in prices of 2017 (thousand rubles per kg of SNF) |
|-------------------------------|----------------------------|---------------------------------------------------------------|
|                               | Actinides | Fission products | Structural materials | C-14 |                               |
| Without fractionation         | 0.69      | 1.13           | 0.20                | 2.3  | 4.0                           |
|                               | 1 class   | 1 class        | 1 class             | 2 class |
|                               | 0.002     | 1.13           | 0.20                | 2.3  | 3.1                           |
|                               | 1 class   | 1 class        | 1 class             | 2 class |
|                               | 0.002     | 0.3            | 0.20                | 2.3  | 2.0                           |
|                               | 1 class   | 1 class        | 1 class             | 2 class |
|                               | 0.2       |                |                     |       |                               |
|                               | 1 class   | 1 class        | 0.20                | 2.3  | 2.7                           |
|                               | 1 class   | 5.7            | 1 class             | 2 class |
|                               | 3 class (Cz+Sr) |                |                     |       |                               |
| Abandon of I and 2 classes    | 850       | 750            | 750-1000            | 2,300 | 645-680                       |
|                               | 3 class   | 3 class        | 3 class             | 3 class |                             |
The data in table 11 does not take into account the cost of 60 years of storage of HLW and repeated overmelting of glass in the “Proryv +” option, as well as the cost of 300 years of storage of HLW and MLW for “Fractionation of MA + fission products” and “Abandon of 1 and 2 classes” options.

5. Conclusion
The Proryv Project establishes the following fractionation options in the MNUP SNF reprocessing:
- recovery of fuel components (uranium, plutonium, neptunium) up to 99.975% for the purpose of subsequent use in the fuel cycle;
- recovery of minor actinides up to 99.9%; separation of americium (for the purpose of subsequent reactor afterburning) and curium (for the purpose of long-term controlled storage, followed by recovery of accumulated plutonium and reactor afterburning of the remaining curium);

These requirements correspond to the 300-year period for achieving the time of radiation equivalence between the generated radioactive waste and uranium raw materials extracted from the subsoil for the production of nuclear fuel.

When using nitride nuclear fuel, it is necessary to substantiate the matrix material for compaction of C-14 under the conditions of the 1st class RW (candidate matrix is magnesium phosphate ceramics).

In case of immobilization (vitrification) of Cs and Sr as 1st class RW, it is necessary to estimate the costs of the two-stage process (granulation + final vitrification in 55-60 years)

Separation of other fractions during reprocessing of MNUP SNF does not seem appropriate from the point of view of optimizing the amounts of final forms of RW and the costs of their burial.

References
[1] NP-019-15 Nuclear and Radiation Safety Russia
[2] 2016 Amendments to NP-019-15, Nuclear and Radiation Safety. Draft regulations 3 (81) Russia p 1-2.
[3] 2017 Isotopic Composition of MNUP SNF of BREST-OD-300 with an average burnup at the height of the core 8% and 12% (No. 124TC, ITCP “Proryv”)
[4] Kascheev V A Handling of structural materials of fuel elements and fuel assemblies during reprocessing of SNF from BREST-OD-300 reactor 2015 Problems of Atomic Science and Technology (PAST) Materials science and new materials series 3(82) pp 71-79
[5] Decree of the Government of the Russian Federation dated October 19, 2012 No. 1069
[6] On Criteria for Classifying Solid, Liquid and Gaseous Wastes as Radioactive Wastes Criteria for Classifying Radioactive Wastes as Special Radioactive Wastes and Disposable Radioactive Wastes, and Criteria for Classifying Disposable Radioactive Wastes
[7] 2017 Options for vitrification of HLW from the reprocessing of MNUP SNF in reprocessing module of Pilot Demonstration Energy Center (PDEC) with BREST-OD-300 reactor in order to minimize the amount of final forms of RW (No. 59TC, ITCP “Proryv”)