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RUSSIAN GROUTING EXPERIENCE (U)

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REVIEWS AND APPROVALS

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1.0 EXECUTIVE SUMMARY

A final report documenting Russian experience in ambient temperature cement-based waste forms, in-tank waste treatment and grout/concrete decommissioning is provided in Attachment 1. The report is titled "The Review of the Russian Experience on Inorganic Binders for Waste Treatment and Tank Closure," and contains technical information concerning:

- An assessment of the properties of hardened cement materials that affect the performance of hazardous and radioactive waste forms.
- A list of additives used to modify the properties of cement-based waste forms.
- The effect of elevated temperature on hydrated portland cement as a function of time and temperatures up to 180°C.
- The effect of radiation exposure on hydrated cement materials as a function of radiation dose up to 6E+09 Rad. Radiolytic gas generation data is also presented and discussed.
- The results of an experimental investigation on the properties of special grout formulations developed as closure grouts.
- Descriptions of full–scale grouting for decommissioning vessels contaminated with radioactivity.
    - Inner tank space of the fuel storage unit on Lepse (service tanker/vessel for nuclear submarines). (Monolith for strength and contaminant immobilization).
    - In-place decommissioning of two nuclear submarine reactor compartments and associated equipment at the Russian Navy Training Center in Paldisk, Estonia. (Radiation shielding and contaminant immobilization)
- Description of in-tank waste solidification for a chemical tank using magnesium phosphate cement.

The Russian experience in aqueous waste stabilization/solidification appears to be limited. The examples discussed in the report are related to work conducted in the recent past or intended for future programs. To date, the Russians have not closed any high-level waste tanks. However, decontamination and repair of a large steel waste tank was discussed. The tank was returned to service after the repairs. In informal discussions, V. A. Starchenko stated that the Russians intend to empty and clean their high-level waste tanks and to then use them as "vaults" for storage of radioactive debris and/or packaged waste/waste forms rather than to fill the tanks with clean grout. Currently the Russians are planning for long-term storage of the low-level waste and for geologic disposal of high-level waste.

The Russian experience in decommissioning large tanks/vessels is also limited to three examples. One example involved pumping grout into the inner tank space in a Fuel Storage Unit on a tanker that serviced nuclear submarines. The concrete was a relatively high-strength concrete which was batched at a central plant and delivered to the docked tanker by truck. The concrete was pumped into the inter tank space. Approximately 6 hours was required to fill the 110 cubic meter void space.

The second example involved decommissioning two nuclear submarine reactor compartments at the Russia Navy training center in Paldisk, Estonia. Two concrete mixes

were used for "conservation" of these units. A highly fluid, pumpable concrete mix was designed for filling the compartments. This concrete also contained Shungizit porous aggregate which was intended to dissipate radiolytic gases formed in the high radiation fields associated with the inside of the reactor compartments. A concrete sarcophagi was also constructed over the reactor compartments as part of the "conservation" effort. This mix contained less cement and no porous aggregate since the radiation field was much lower.

Finally, magnesium phosphate cement was used to solidify the radioactive ferrocyanide sludge in a defective tank. The tank had a volume of 3200 cubic meters and it contained 70 cubic meters of sludge with a liquid to solid ratio of 2 to 1. The tank had no mixing capabilities so the magnesium oxide and phosphoric acid were simply added on top of the sludge. As a consequence, crust formed on top of the sludge after the first addition and precluded subsequent additions of the magnesium oxide. However, the solidification of the sludge was sufficient to stop the tank from leaking. Magnesium phosphate cement was selected for this application because of the strong stabilization/fixation properties of the resulting salts (barium, tin, zirconium and thorium phosphates are very insoluble).

The Russian regulatory requirements (in Russian) are provided in Attachment 2. These regulations have been in effect since January 1, 2001. In general the Russian requirements for cement-based waste forms are similar to the US Nuclear Regulatory Commission requirements. An abridged English summary is provided in Attachment 3, in the form of slides from a presentation on "Collecting, Reprocessing, Storage and Conditioning (Treatment) of Liquid Radioactive Waste" by V. A. Starchenko.

The Russian waste form requirements include limits on radionuclides, leaching rates, referred to as water resistance, compressive strength, radiation stability, resistance to thermal cycling, and a durability requirement based on the strength of a sample immersed in water for 90 days. In addition to prohibiting waste with fire or explosive characteristics, wastes that can react with the compounds in the cement to form toxic substances such as ammonium slats are also prohibited. Complete dehydration of high-salt aqueous low-level waste solutions is prohibited per the Russian regulations in order to prevent possible exothermic interaction of the compounds in the dry residue. This sensitivity to management of dry salt waste is applicable to treatment, storage and disposal of calcined waste at DOE facilities.

Strength is used as a measure of durability and performance. Additives used to improve leaching include sodium silicate to reduce overall permeability, zeolites to improve cesium leaching and organic polymers (polymer cements including latex-modified cements) to reduce porsoity/permeability and thereby improve tritium leaching. Pozzolans were discussed for improving matrix properties and alkaline additives were discussed as set regulators specifically for borate wastes.

An interesting concept employed by the Russian grout formulators is to include a porous aggregate, "Shungizit," as a ingredient in waste forms and/or decommissioning grouts exposed to high radiation doses. The purpose of the porous aggregate is to provide for gas permeability through the aggregate. In addition, the matrix portion of these grouts/waste forms is designed to have low permeability to water. This is achieved by a low water to cement ratio in the mix. This type of mix provides a means of venting the waste form so that radiolytic gases do not accumulate while at the same time minimizing the contact between the contaminants and leachate.

The final report on Russian Grouting experience provided an opportunity for international cooperation and access to Russian grouting/waste form experience. The data on radiolytic gas generation form grout mixtures was already used in evaluation of the source of hydrogen and methane generation detected in the sampling ports around the SRS high-level waste tanks in 2002. The concept of venting the radiolytic gases from a waste form by adding porous aggregate is being considered for future cement-based TRU waste forms at SRS.

2.0 BACKGROUND

2.1 OBJECTIVE

The objectives of this work were to document the Russian experience on grouting for waste forms and tank closures or other decommissioning applications. This task was designated as Task H-4 in TFA Work Element 923, SRS TTP SR16WT51 Subtask H. The milestone designation is 923-1.5-1.

The approach for obtaining the Russian experience in ambient temperature waste treatment and in decommissioning tanks and other vessels was to issue a subcontract to the American Russian Environmental Services, Inc. (ARES) for preparation of a report on the Russian experience. ARES contracted the report to Daymos, Ltd. V. A. Starchenko from Daymos was the project manager. N. I. Alexandrov and V. P. Popik were the principal investigators. The final report from Daymos/ARES is attached.

3.0 ACKNOWLEDGEMENTS

This effort was requested and supported by L. Bustard, Sandia National Laboratory as part of the Scope of Work for the Tank Focus Area, Enhanced Grout Program.

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4.0 ATTACHMENT A

Review of the Russian Experience on Inorganic Binders for Waste Treatment and Tank Closure - Final Report

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THE REPORT

" The Review of the Russian Experience on Inorganic Binders for Waste Treatment and Tank Closures "

The manager of the project
V. A . Starchenko
The responsible executors
N. I. Alexandrov
V. P. Popik

St-Petersburg 2001.

Summary.

Keywords: radioactive wastes, tanks for waste storage, radiation - dangerous objects, cementation of radioactive wastes, portland cement mixtures, radiation stability, radiolysis, radiolytical gases release, durability of the cement compositions, leaching of radionuclides, large scale experience.

In this report it is given the review of results of researches carried out in Radium Institute and in number of other Russian enterprises directed on elaboration of high-performance concrete - conserving agents designed for solidification of liquid radioactive wastes and for transfer into ecological - safe state of HAW storage tanks and decommissioned radiation - hazardous objects of nuclear power engineering. The existing Russian practical experience in the indicated area is also briefly described.

CONTENT.

INTRODUCTION.

The present review is prepared in accordance with subcontract No 8D6266-EE/2001-01 concluded with corporation " American Russian Environmental Services, Inc." The subcontract goal is to prepare a technical report on Russian experience with the use of inorganic binders for treatment of radioactive waste and for closing of waste tanks. For reflection of the Russian experience on use of cement grouts for waste treatment and tanks closure the following major subjects were determined, which should be reflected in the report:

- research of influencing of portland cement grouts composition on the mechanical strength and radiation stability of the solidified cement mixtures;
- studies of influencing of cement grouts composition on an yield of radiolytic hydrogen at high radiation doses of compositions;
- estimation of capability to improve cement grouts at their contact with groundwater due to introducing in them of the special additives;
- estimation of capability to decrease radionuclides leaching from cement grouts due to addition of the components, which can decrease migration rate of impurities in cement matrixes;
- practical experience on application of grouting methods for decommissioning of objects of nuclear power engineering and Navy nuclear radiation hazardous facilities;
- key technical problems regarding of improvement of cement grouts properties that might be resolved through an analytical and experimental research program during the potential next stage of the project;
- filled in survey of enhanced grout needs for tank closure provided by the SRTS customer.

The compositions on the basis of ordinary portland cement (OPC) and some other inorganic binders (blast furnaces slag (BPS) and pulverized fuel ash (PFA) are broad applying for solidification of low active liquid radioactive waste (1). In Russia and in USA cement grouts are applied also to conservation of radwaste tanks and other radiation - dangerous objects. The important direction of cement grouts usage is their application as construction materials and biological protection in containers and dry storages designed for long-term storage of burned up nuclear fuels.

For effective utilization in the mentioned above directions cement grouts should have a series of particular properties:

- high radiation stability that is especially important at usage of cement mixtures for decommissioning of objects with high level of irradiation (reactor compartments of nuclear powered vessels, burned up nuclear fuel storage);
- the structure of the cured grout should provide release of radiolytic gases without disturbance of monolith integrity;
- the structure of the cured grout should have low water permeability to ensure low leaching rate of radionuclides from the object in environment for all time of its long-term storage.

The special cement mixtures suitable for solidification of the liquid radwaste should have a following complex of properties:

- high radiation stability, that will allow to solidify radwastes with enough high level of activity;
- the structure of the cured cement mixture should have low water permeability to reduce to minimum release of the radionuclides from solidified radwastes in environment for all time of their long-term storage;
- the cured cement compositions should have high frost resistance;
- the cured cement mixtures should have low speed of leaching of the basic components of cement to provide high stability of the cured cement grout at contact with groundwater;
- the composition of cement grouts used for radwaste solidification should provide enough high mechanical strength at the high contents of soluble salts in cemented waste.

The primary goal which was put at realization of researches of cement mixtures with reference to treatment of liquid radwaste and decommissioning of radiation - hazardous objects including waste tanks was concluded in elaboration of cement mixture compositions which in the most degree satisfy to the presented above requirements.

Previously the literature on structure and properties of cured cement mixtures was studied with the purpose of selection of rational paths for looking up of cement mixtures composition possessing a given complex of properties.

The main features of structure and properties of hardening cement mixtures with reference to problems of radwaste treatment and radiation-hazardous objects decommissioning.

The term "cement" usually implies portland cement or materials containing portland cement. The cement produces by high temperature calcination with a partial melting of materials having high content of calcium oxide. The obtained product (clinker) is crushed up to rather high specific surface (approximately 4000 cm^2/g) to activate cement. The cement hardening begins when this powder is mixed with water. At preparation of hardening cement mixtures the water-cement weight ratio is usually selected within the limits of 0.25-0.6; the mixtures with smaller water-cement ratio become rigid and lose plasticity whereas at the greater ratio mixtures can be stratified and liberate water. The prepared mixtures must be used within 1-2 hours then their setting starts. The development of strength becomes noticeable only in 1-2 days. Then the strength fast accrues.

In general all processes of cementation have a" window" 1-2 hours, during which the prepared mixture should be used. In further it is necessary to allow to cement to hydrate chemically without mechanical effects. The decelerators and boosters of cement setting are known, but their long-time influencing on properties of cement especially in conditions of irradiation is not enough well established and their applying for cemetnation of radiation - dangerous objects can not be recommended and therefore it is necessary to work with cement mixtures in the mentioned above period (I).

Internal structure of cured cement mixtures.

In structure of the cured cement it is possible to mark three characteristic features: a matrix of solid phases, both crystalline and amorphous, system of pores of the different size and form and aqueous phase located in pores. The model of a microstructure of cured cement is shown) on fig. 1.

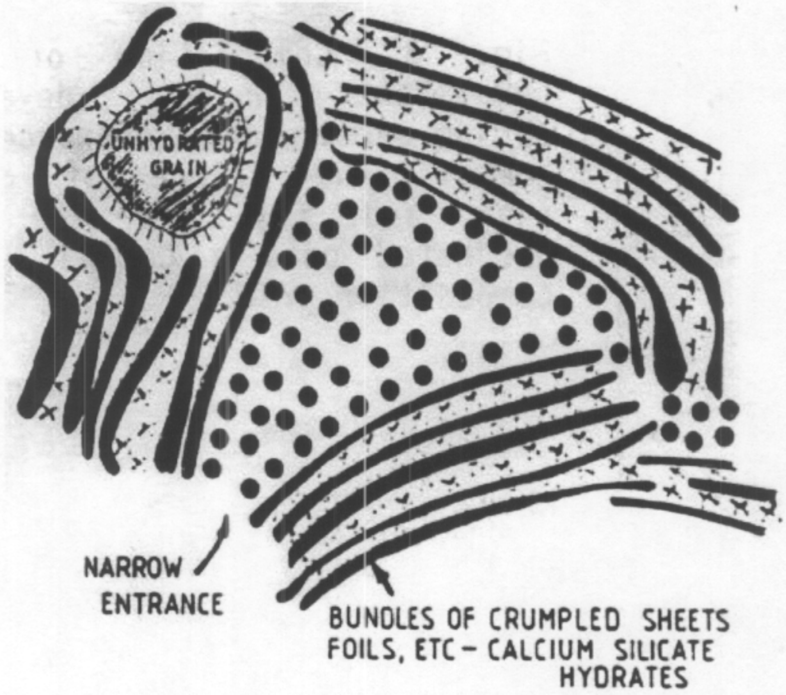

Fig. 1. Model of a microstructure of cured cement.

The gel of calcium hydrosilicate (C-S-H) is the most important component of solid phase of cured cement mixture as it contains main part of micropores. C-S-H phase is almost amorphous. As it can be seen from fig. 1, the phase of C-S-H gel consists from elements, the centers of which are grains of cement surrounded with products of its hydratation. Each element has flaky structure; the random packaging of these units creates a grid of micropores. It is not surprised that the most part of sorbtion potential of the cured cement is determined with micropores of C-S-H gel. The conducted researches show, that normally prepared samples of concrete, as a rule, do not contain pores with size > 1 micron and the main part of pores has the size < 0.1 microns. These pores determine the diffusion characteristics of cement mixtures in relation to radioactive ions therefore measured for them diffusion constants are not property of cement matrix, but concern to system pores - matrix. It is necessary to mark that in cured cement mixtures there are also other kinds of pores. In space between hydrated and not hydrated fragments of cement so-called capillary pores are formed. These pores have large sizes than pores of C-S-H gel. Common and capillary porosity is reduced with growth of degree of cement hydration, gel porosity increases a little because the volume of C-S-H gel (2) is augmented.

C-S-H gel acts as a main factor controlling pH of cement mixtures at the greater cure period. The Ca/Si ratio in C-S-H gel varies within the limits 1.8-0.9. At Ca/Si ratio 1.8. where C-S-H gel coexists with $Ca(OH)_2$, the equilibrium pH value practically equal to pH of saturated solution of calcium hydroxide. At decreasing of this ratio concentration of calcium in aqueous phase and pH are descended. The same decrease will be obtained as a result of leaching, as the cement will lose the most soluble calcium compounds. The pH of aqueous phase in pores of concrete has high importance for it properties as material for conservation of radiation-hazardous objects, since the high pH value promotes a decrease of solubility of a number of radionuclides, in particular, of most dangerous long-lived α-emitters (3) and favours to decrease of corrosion of carbon steel.

Usage of the additions for modification of properties of cement mixtures.

Addition of different components for modification of properties of cement mixtures is widely used in practice. In the table I some materials frequently added in cement are presented and the effects which are obtained with addition of such component are shown too (I).

Table I.

Materials adding in cement mixtures for improvement of their properties.

Material.	Effect from the component.
Fly ash from coal combustion.	Decrease permeability, increase mixture fluidity, lower initial heat evolution.

Ground granulated blast furnaces slag.	Decrease permeability, lover internal E_h and initial heat evolution and increase mixture fluidity.
Natural puzzolans.	Increase of sorption.
Microsilica. Superplasticizers.	Decrease permeability, increase sorption. Reduce water content and permeability.
$Ca(OH)_2$, NaOH	Condition borate waste and ensure set.
Sodium silicate .	Precipitate heavy metals, decrease permeability.
Miscellaneous getters: chemical and structural.	Reduce solubility of specific radwaste species e.g. Ag^+ for iodine conditioning.
Organic polymers.	Decrease permeability, getter for tritium.

It is possible to divide materials adding in cement into three groups. One group including natural puzzolane materials, ash, microsilica and slag can be called puzzolanic because these materials are activited with cement, react with water and cement and become the integrated part of cement matrix. The size of particles of these materials are usually less a little than size of cement particles, so the formed matrix have a little more dense packing of fragments than cement matrix and provides smaller permeability. The siliceous fly ashes and slags have a high content of glass, which concerns to main puzzolanic components (4, 5). In the short term materials of this group are enough inert and more active part of mixture (cement or $Ca(OH)_2$) determine properties of mixtures in this period.

Other materials in the table I are mainly modifying agents for matrix. The cement is usually mixed with quantity of water which is sufficient to receive fluidity of mixture appropriate to the application. But such fluidity usually receives at excess of water over that is required for a chemical hydration, it leads to porous and permeable matrix. Critical water-cement ratio indispensable for full hydration of portland cement is equal approximately 0.24, but such mixture is too hard for practical usage. The demanded fluidity can be reached at low water content by adding of superplasicizers, which are the high-molecular polar organic compounds (for example sulphonated melamin, naphthalene formaldegyde polymers or lignosulfonates(6). As usual 0.2-2 % of plasticizer, depending on it type, is diluted in mixing water. It enables to receive normal fluidity of mixtures at low water content that give possibility to obtain less permeable product. The final destiny of plasticizers in cement is not completely known. Probably initially they are fast sorbed on the products of cement hydration with the subsequent desorption and partial irreversible precipitation.

Calcium and sodium hydroxides are used for acid wastes neutralization and to condition borates which have negative influence on cement setting and even can make it impossible; addition of hydroxides reduces borates influencing.

The applying of sodium silicate is described in the literature (7), but it is not used widely in practice. This compound is soluble in water and precipitates a wide range of metallic ions from solution usually in the amorphous form, however nature of precipitations in cured cement matrixes is poorly known.

Particular getters it is offered to use to reduce leaching components of the radwastes badly sorbed by cement, for example of cesium and iodine (8). For an iodine it is offered to use addition of silver or barium (9, 10). As getter for cesium it is possible to use zeolites (II). However capacity of such components is limited because of chemical reaction between cement and zeolites. Besides the zeolite capacity is decreased because of competition of K^+ and Na^+ ions presented in cement (12). The final result of it is the necessity to add in cement of high quantity of zeolite to lower Cs^+ leaching. It results in a number of disadvanages: the mixture has high consumption of water and low compression strength. Thus the getters application is rather specific and demands special consideration in each particular case.

The organic polymers also can be used for modification of cement matrixes. In general it is necessary to concern with caution impregnation of cement matrixes with organic polymers as the diffusion coefficients in polymers are rather high and long-term stability of polymers to irradiation and degradation under an environmental stress is either unsatisfactory or unknown. The possible advantage of polymeric -cement matrixes can be improvement of immobilization of tritium (13) .

Effect of elevated temperature and radiation.

As it was noticed above, major component determining long-term properties of cured cement mixtures is C-S-H gel, which at ordinary temperatures is metastable, but exists long, natural samples of such phase with age $20*10^5$ years (I) are known. At temperature 170-180 ^{0}C this phase crystallize in autoclave during 16-24 hours. In result of gel crystallization pH of pore liquid is decreased on 1-2 units in comparison with C-S-H gel having equivalent composition, besides a considerable change of this phase volume can take place in result of crystallization. In the literature practically there are no data on rate of crystallization of C-S-H gel at temperature below 100 ^{0}C, but it is possible to suppose that the considerable temperature rise inside of curing cement mixture is extremely undesirable because of hazard of C-S-H gel crystallization.

The information, available in the literature, demonstrates that even the high doses of absorbed γ- radiation have not negative influence on the properties of dry cement (14). The effect of irradiation on cured cement mixture is appeared basically through radiolysis of water presented in it.

Main result of water radiolysis is release of gases and, first of all, of hydrogen. Irradiation of cement mixture can also modify structure of it and increases degree of it crystallinity (15)

The yield of hydrogen at radiolysis of water in cement mixtures depends on their composition, in particular, from compos ition of binders, which was used at their preparation. So it was noticed that trend of hydrogen yield for mixtures in which as binders were used ordinary portland cement (OPC) and its mixture with blast furnace slag (BPS/OPC) and fly ash (PFA/OPC) is: :BPS/OPC > OPC> PFA/OPC.

This in part may be attributed to differences of composition of aqueous solutions in pores (16).

Presented above literature information on structure and properties of cured cement mixtures allows to dedicate following main directions in elaboration of optimal composition of cement mixtures designed for radwastes treating and conservation of radiation-hazardous objects:

- usage of superplasticizers with the purpose to decrease quantity of water which is necessary for obtaining cement mixtures with demanded fluidity and low permeability;
- looking up of the addition components and filling materials ensuring to cured mixtures a gas permeability enough for deleting of radiolytical gases at preservation of their high water tightness;
- looking up of mixtures composition with a low yield of radiolytical hydrogen;
- research of possibility of using of the components sorbing radionuclides for decrease their leaching from cement mixtures.

In this report the results of experimental researches carried out in these directions are presented.

2. EXPERIMENTAL ENVESTIGATION OF PROPERT IES OF SPECIAL CEMENT MIXTURES.

The main results of the carried out researches are reviewed below. In the given section of the report the techniques of experimental researches are briefly shown also. It is necessary to mark that during preliminary researches 22 cement mixtures of different composition were tested. The mixtures possessing the most favourable combination of properties were selected for further researches. The compositions of these mixtures are shown in the table 2.

Table 2.

Composition of investigated cement mixtures.

No of mixture	Main components of Mixture		Modifying Components.		Water : solid Ratio (W/S)
	Material	Content, % vol.	Material.	Content, % vol.	
1.	Cement Shungizit filling	66.0 33.7	S-3 SDO	0.7 0.05	0.24

	material				
2.	Cement Shungizit filling material Shungizit sand	40.0 20.0 40.0	S-3 SDO	0.7 0.05	0.35
3.	Cement Shungizit filling material Mordernite sand	40.0 20.0 40.0	S-3 SDO	0.7 0.05	0.30
9.	Cement Quartz sand	40.0 60.0			0.38
21.	Cement Shungizit sand Clinoptilo-lite Sand Microsilica	38.0 37.0 5.0 7.0	S-3	0.9	0.32

2.1. Experimental techniques.

2.1.1. Irradiation of cement mixtures.

2.1.1.1 The characteristics of the gamma -irradiation installation.

Irradiation of investigated cement mixtures was carried out on gamma - irradiation installation of V. G. Khlopin Radium Institute, which had following basic parameters:

- Source of a gamma-radiation - Co - 60;
- Sizes of irradiation chamber:
  height - 200 mm;
  diameter -120 mm;
- Calculated dose rate of gamma-radiation - 150 rad / sek.

The real dose rate in the irradiation chamber was measured with usage of chemical dosimetry method based on radiation oxidation of divalent iron ions in sulphate media saturated oxygen of air, so-called Frank's dosimeter (17) .

2.1.1.2. Technique of cement mixtures irradiation.

Samples of cement mixtures were placed in glass cylindrical vessels with inner diameter and height 21-22 mm and wall thickness I mm or in glass ampoules with diameter 15 mm and length 90 mm with the same wall thickness. Vessels with samples of mixtures destined for research of composition and quantity of radiolytic gases were placed in titanium hermetic capsule with wall thickness 0.5 mm.

2.1.1.3. Tests of mechanical properties of cured cement mixtures.

The determination of compressive strength were carried out according to GOST 10180-90. The hydraulic press PD-10 (GOST 8905-73) was applied to tests. The inaccuracy of measurements of load did not exceed 2 % rel. Compressive strength (Rcom) of samples was calculated according to:

Rcom = F/A

Where: F - destructive load;
A – cross section area of sample.

The mechanical tests irradiated and not irradiated samples of the same composition were done simultaneously.

2.1.1.4. Test of cement mixtures air permeability.

The device AGAMA-2R was used for evaluation of cement mixtures air permeability. This device gave possibility to measure time which was necessary to definite volume of air pass through testing sample into evacuated chamber. As all tested samples had the identical geometrical sizes for comparison of their air permeability the value of rate of air accumulation in vacuum chamber was used.

2.1.1.5. Test of waterproof of cement mixtures.

Testing of waterproof of cured cement mixture samples was done in according to GOST 12730.5-84 on which the measure of waterproof is the water pressure (W) at which on a surface of test cylindrical sample there is no yet wet spot. The increase of water pressure was made by steps 0.2 MPa during 1-5 min After each step the stop during 16 hours was made.

2.1.1.6. Research of cement mixtures structure changes at irradiation.

For research of influencing of irradiation on structure of cement mixtures the methods X-ray phase analysis and differential thermal analysis were used.

X-ray phase analysis.

For identification of crystalline phases in investigated samples the powder method permitting to receive the information on chemical composition and structure was used . For obtaining diffraction pictures the X-ray diffractometer DRON-1,5 was used with copper anode X-ray tube at anode V = 30 kV and I = 25 mA. CuK α – radiation was filtered through Ni – foil.

The identification of crystalline phases was made by comparison of data obtained from diffractograms, with data presented in catalogue 1CPD5 issued by Integrated Committee of the Powder Diffraction Standards (USA).

Differential thermal analysis of cement mixtures.

The differential thermal analysis (DTA) of cured cement mixtures was carried out with help of derivatograph Q 1500 D of the Hungarian production.

During DTA it was fixed weight changes of grinded specimen of investigated mixtures at controlling heating from 20 to 1000 ^{0}C.

2.1.1.7. Research of quantity and composition of gases which are evolving at irradiation of cement mixtures.

The chemical analysis of gas phase accumulating in capsules and ampoules during irradiation of cement mixtures was made with gas chromatographic method. The gas chromatograph "Tsvet-100" was used for gas analysis. The conditions of chromatography and system of detecting were selected recognizing that the main anticipated components of gas phase evolving at cement mixtures irradiarion are: O_2, H_2, CH_4, CO_2. The accuracy of definition of the contents of these components in samples of gas was 5-10 %.

2.1.1.8. Research of leaching of radionuclides from cured cement mixtures.

Raduonuclides (Pu-239, Am-241, Cs-137, Sr-90) were added in water used for cement mixtures preparation as nitrates. Before beginning of leaching experiments all specimens of cement mixtures were cured during 28 days.

The technique of experiments on definition of speed of leaching of radionuclides from hardening compositions on the basis of cement corresponded to the guidelines of IAEA (18) and GOST 2914-91 (19). The distilled water was used as leaching liquid.

The concentration of radionuclides in samples of leaching water was determined radiometrically on α-radiation for Pu-239 and Am-241, on β-radiation for Sr-90 and on γ-radiation for Cs-137. The accuracy of activity measurement of samples was 5-10 % rel.

2.2. RESULTS AND THEIR DISCUSSION.

Influencing of gamma-radiation on mechanical properties, air permeability and waterproof of cement grouts of different compositions.

As it was already marked in the introduction, primary goal of researches, the results of which are reviewed in this report, is looking up of cement mixtures (concrete) compositions which properties meet to demands to concrete - conserving agents used for conservation (closing) of radiation - hazardous objects, including waste tanks. In this connection influencing of gamma-irradiation on a mechanical strength, air permeability and waterproof of concrete of different compositions was studied. The obtained results are shown in the table 2.2.1

The table 2.2.1.

Influencing of gamma-radiation on mechanical properties, air permeability and waterproof of concrete of different compositions.

No of concrete compo-sition	Time of curing, days	Dose of Gamma -radia-tion, Mrad.	Compre-ssive strength, MPa.	Air perme-ability, Cm^3/sec.	Water-proof W, tm.
1	28	266	43.5	0.037	12
	90	855	41.6		
	180	1710	56.0		
	270	2565	45.9		
	690	6000	40.2		
1c	28		27.5	0.022	12
	90		25.0		
	180		20.1		
	270		29.7		

2	28	266	25.4	0.263	10
	90	855	25.0		
	180	1710	28.0		
	270	2565	26.7		
	690	6000	33.9		
2c	28		16.0	0.04	2
	90		16.8		
	180		14.7		
	270		17.4		
3	28	266	20.5	0.11	2
	90	855	21.0		
	180	1710	21.1		
	270	2565	23.8		
	690	6000	19.4		
3c	28		15.5	0.322	2
	90		15.0		
	180		13.7		
	270		16.3		
21	28	266	31.3	0.029	> 12
	90	855	34.9		
	180	1710	39.5		
	270	2565	38.8		
21c	28		37.5	0.03	> 12
	90		35.0		
	180		39.2		
	270		39.0		

As it is visible from the data of table 2.2.1., irradiation of compositions 1-3 at their curing during 28-690 day (the absorbed dose 266-6000 Mrad) results in increase of compressive strength in comparison with control samples of cement grouts curing in normal conditions (1c-3c). For composition N 21 some decrease of strength marked at curing period 28-180 days under effect of gamma-radiation, however at the greater curing period strength of sample hardening in normal conditions and under irradiation are equalized. The obtained results allow to draw a conclusion that the gamma-irradiation of investigated compositions on the basis of portland cement with dose up to 3000 - 6000 MRad will not render negative effect on mechanical strength of the cured compositions.

Evolution of radiolytical gases at gamma-irradiation of cement mixtures.

For definition of conditions ensuring safe long-term storage of decommissioned radiation - hazardous objects, the knowledge of composition and quantity of gases evolved at irradiation of cement mixtures used at their closure is necessary.

With this purpose two series of experiments were conducted, the goal of the first of which was estimation of composition of gases released at gamma-irradiation of cement mixtures, and the goal of the second series was determination of quantity of hydrogen evolved at irradiation of mixtures of different compositions.

The results of analysis of gas phase in titanium capsules are shown in the table 2.2.2.

The table 2.2.2.

Composition of gas phase in capsules at different radiation time.

Time, days.	Dose of Gamma-radiation, Mrad.	No of concrete composition	Composition of gas phase in capsule, % vol.			
			H_2	O_2	CO_2	CH_4
0	0	1	< 0.0001	78.1	0.03	-
0	0	2	< 0.0001	78.0	0.02	-
0	0	3	< 0.0001	77.9	0.03	-
28	266.1	1	3.3	10.0	0.01	< 0.1
		2	0.1	18.6	0.1	0.02
		3	12.0	13.1	0.01	0.2
36	342.1	2	0.1	18.6	0.1	0.02
64	608.2	1	1.2	14.4	0.06	0.15
		2	0.5	11.1	0.02	0.16
		3	8.5	15.6	0.03	1.1

The results of the analysis of gas phase composition in capsules have shown that at irradiation of cement mixtures that evolution of hydrogen and small amounts of methane take place. Evolution of hydrogen is connected with radiolysis of water containing in cement mixtures. Methane evolution is the result of radiolysis of organic components (superplasticizer S-3 and component SD0), added in mixtures during their preparation. Literature data show that methane is one of the main products of radiolysis of complex organic compounds (20). Apparently that at estimation of safety of long-term storage of radiation - hazardous objects filled with cement mixtures, it is possible to consider only hydrogen release, since quantity of other fire hazard gas - methane is insignificant.

For more reliable determination of hydrogen quantity released at gamma-irradiation of cement mixtures, the experiments on their irradiation in soldered glass ampoules were done that eliminated a capability of leakage of gases during irradiation. The obtained results are shown in table 2.2.3.

The table 2.2.3.

Evolution of radiolytic hydrogen at irradiation cement mixtures in glass ampoules.

No of concrete composition	Irradiation time, days.	Dose of Gamma-radiation, Mrad.	Hydrogen evolution rate, l/m^3 of mixture per hour.
1	5	47.5	0.018
	15	142.6	0.106
	29	275.6	0.064
2	5	47.5	0.034
	15	142.6	0.012
	29	275.6	0.058
9	5	47.5	0.025
	15	142.6	0.01
	29	275.6	0.01

As it is visible from the data of the table 2.2.3. the rate of hydrogen evolution does not exceed 0.11 l / m^3 of mixture per hour at dose rate 110 rad / sek. Apparently that the evolution of radiolytiical hydrogen is necessary to take into account as the dangerous factor only at conservation with the help of cement mixtures of objects with high radiation level.

Research of influencing of addition of boron carbide on release of radiolytical hydrogen from cement mixture at irradiation.

At concreting of the objects containing fissile materials for maintenance of nuclear safety it can be necessary to add in cement mixtures used for their conservation matters – neutron poisons. One of the most known matters of such type is the carbide of boron - SiC. In this connection the experiments on influencing of addition of SiC in mixture No 2 on radiolytical hydrogen release were made. With this purpose the irradiation of mixture No 2 with addition of SiC in soldered ampoules was made. The powder of SiC in quantity 50 g/kg of cement mixture was added in dry mixture of components of composition No 2 before mixing with water.

The results of determination of quantity of radiolytical hydrogen evoluted at irradiation of composition No 2 without and with addition of SiC presented in table 2.2.4.

The table 2.2.4.

Influencing of addition of SiC on evolution of hydrogen at irradiation of composition No 2.

Irradiated Composition	Weight of com-position in ampoule, g.	Irradiation time, days.	Dose of Gamma-radiation, Mrad.	Hydrogen Evolution rate, L/m^3 of mixture per hour.
No2	12.3	5	47.5	0.034
	14.6	15	142.6	0.012
	10.0	29	275.6	0.052
No2 + SiC	10.7	7	66.5	0.028
	12.5	28	266.1	0.051
	12.7	70	665.3	0.053

As it is visible from the data of the table 2.2.4, the introducing of SiC in composition of cement mixture practically has not effect on rate of hydrogen evolution at gamma-irradiation of mixture.

Determination of quantity of radiolytical hydrogen which is held back by concrete.

At irradiation of compositions on the basis of cement radiolytic hydrogen is generating inside cement mixture and than goes out from mixture by diffusion through a system of cement mixture pores. It was possible to expect that the part of radiolytical hydrogen will remain in mixture phase. In this connection the research of content of hydrogen in samples of the irradiated cement mixtures was carry out.

The results of experiments on definition of hydrogen contents in samples of mixtures having compositions No 1, 2 and 3, irradiated 90 days (gamma-radiation dose 855 Mrad), are presented in the table 2.2.5.

Table 2.2.5.

The residual contents of hydrogen in samples of irradiated compositions.

Temperature of Mixture Heating, ^{0}C.	No of composition					
	1		2		3	
	Quantity of H_2		Quantity of H_2		Quantity of H_2	
	% from common.	l/t of mixture.	% from common.	l/t of mixture.	% from common.	l/t of mixture.
200	1.0	0.3	3.0	1.4	-	-
250	-	-	-	-	3.0	0.4
400	67.0	18.3	65.0	31.4	-	-
500	99.0	26.9	100.0	48.4	37.0	5.3
700	100.0	27.1	-	-	96.0	13.9
900	-	-	-	-	100.0	14.5

From the data, presented in the table 2.2.5, it is visible that all studied samples of the irradiated mixtures are containing significant amounts of hydrogen, and at heating the output of the main quantity of hydrogen is occurred at temperature 400 -500 ^{0}C for mixtures I and 2 and 500-700 ^{0}C for mixture 3. The fact that at temperature 200-250 ^{0}C the output of hydrogen is a little indicates that the binding strength of hydrogen with cement matrix is higher than at it physical adsorption. The data of the table 2.2.5 allow to suppose that the main part of hydrogen leaves from mixture at temperature at which destruction of crystallohydrates obtained at hydration of cement takes place. These crystallohydrates can retain hydrogen by formation of solid solutions similarly to chalkstone (21). Probably also holding of hydrogen by samples because of it including into closed micropores (22).

Research of influencing of the additives on leaching rate of radionuclides.

The results of experiments on influencing of addition into cement mixtures of matters which are capable to sorb radionuclides, on rate of their leaching are shown in the table 2.2.6.

The table 2.2.6.

Influencing of the addition of sorbents on leaching rate of Cs-137 and Sr-90 from cement mixtures.

Leaching time, days.	Total fraction of a leached radionuclide, % from initial quantity.							
	Composition I		Composition II		Composition III		Composition IV	
	Cs-137	Sr-90	Cs-137	Sr-90	Cs-137	Sr-90	Cs-137	Sr-90
1	0.05	<0.01	<0.01	0.15	0.26	<0.01	<0.01	0.11
4	0.18	<0.01	0.05	0.15	0.75	0.44	0.08	0.15
7	0.24	<0.01	0.12	0.28	0.92	0.72	0.08	0.19
13	0.50	<0.01	0.16	0.28	1.29	0.80	0.08	0.27
26	0.80	<0.01	0.16	0.28	2.28	0.84	0.08	0.33

The matters added to cement mixtures were:

- composition I – without addition;
- composition II – natural zeolite – clinoptilolite;
- composition III- antimony phosphate;
- composition IV- tin phosphate.

The selection of these inorganic sorbents was determined by that they are capable effectively to sorb radionuclides from solutions of miscellaneous compositions, including solutions, close on composition to pore solutions of cured cement mixtures (23, 24, 25).

As it is visible from the data of the table 2.2.6, the addition in cement mixture of the mentioned above components has essential influencing on leaching rates of Cs-137 and Sr-90.

The introducing in cement mixture of 10 % of clinoptilolite and tin phosphate results in decrease of fraction Cs-137 leached during 26 day in 5 and 10 times accordingly. The addition of antimony phosphate lead to increase of Cs-137 leaching.

In the case of Sr-90 the minimum leaching rate is watched for cement mixture without any additives, the introducing in a structure of composition any of the tested components results in increase of leaching rate, which is most significant in case of addition of antimony phosphate.

The results of the carried out researches have shown that the addition in cement mixtures of matters which are capable to sorb radionuclides, can reduce their leaching. However, the influencing of such components on properties of cured cement mixtures has complex nature and in some cases can result not in decrease, but even in increase of leaching rate which can be connected with negative effect of additives on structure of cured cement mixtures.

The results of done researches have shown that the composition of portland cement mixtures has high influence on properties of these mixtures. Changing composition of cement mixtures it is possible to obtain mixtures possessing complex of properties in the most degree adequate to areas of their applying.

In the present review we consider two fields of application of portland cement mixtures:

- conservation (closure) of radiation - hazardous objects of atomic engineering, in particular of waste tanks, reactor compartments of vessels with nuclear energy installations etc.;
- solidification and preparing for long-term storage of the liquid radwastes.

The optimal complex of properties of cement mixtures designed for applying in each of introduced above areas can be various.

So at usage of these mixtures for conservation of radiation - hazardous objects with high radiation level the important value has a high gas permeability of using mixtures because that allows to ensure going out of radiolytical gases without disturbance of concrete monolith integrity. For cement mixtures intended for solidification of the liquid radwastes middle and low level of activity the requirements of low leaching rate of radionuclides and high stability of mixture at affect of groundwater and other factors of environment of radwastes storage are going on the foreground.

Considering from these positions results of researches of portland cement mixtures it is possible to make following conclusions.

For conservation (closure) of radiation-hazardous objects with high level of radiation for which the release of significant amount of radiolytical hydrogen is possible, the most favourable combination of properties has the composition No 2, as it has a high gas permeability, that provides output of hydrogen from massive of the cured composition without disturbance of it integrity.

For conservation of radiation-hazardous objects with low radiation level, for which quantity of emanation of radiolytical hydrogen is insignificant, and for solidification of the liquid radwastes the preference is necessary to give to composition No 21, which has a low permeability, low rate of radionuclides leaching and highest stability to affect of groundwater from studied compositions on the basis of portland cement

Evaluation of service time of the cured cement mixtures at long-term storage under affect of gamma-radiation and factors of environment.

The different factors of an environment of storage act on the cured compositions utilized for conservation of radiation-hazardous objects or for cementation of radwastes prepared to long-term storage. These factors cause corrosion of cement mixtures that results in decreasing of their durability. As the cured compositions should ensure safe long-term storage of radiation-hazardous objects and cemented radwastes it is important to have a capability of the evaluation of durability of cement mixtures in conditions of their storage.

For estimation of demanded service time of the cured cement mixtures which is necessary for ecological safe handling with conservated (closed) radiation-hazardous objects and cemented radwastes it is possible to start with following reasons.

The radionuclides which can be contained in radiation-hazardous objects and radwastes can be divided into three groups

- short-lived nuclides with a half-life up to 2-3 years;
- the long-lived fission products, basic of which are Cs-137 and Sr-90, having a half-life about 30 years;
- superlong-lived nuclides (actinides, Pd-107, Tc-99, I-129) the half-life of which reaches hundreds thousand and even millions years.

For nuclides of the first group for their practically full disintegration it is enough several tens years and the reliability of cement mixtures on this period does not cause doubts.

For nuclides of the second group for practically full disintegration (the decrease of activity in 100 000 times) it is necessary time about 500 years.

Nuclides of the third group for their disintegration need time measured in millions of years that far leaves the frameworks of the most optimistic estimations of durability of the cured mixtures on the base of portland cement.

Outgoing from above-stated, is reasonable to esteem cementation with mixtures on the basis of portland cement as reasonable method for maintenance of safe storage of the radwastes containing nuclides of the first and the second groups for it is necessary that the durability of mixtures was not less than 500 years. Apparently, that for estimation of suitability cement mixtures for this purpose it is desirable to have possibility to predict their durability in conditions of affect on them of gamma-radiation and storage environment.

The analysis of reasons which can lead to decrease of strength and impairment of other physicochemical properties of concretes has shown that the primary one is the interaction with groundwater and temperature regime of storage. The negative influencing on durability of concrete can be done also by irradiation, but this factor can become essential only for objects with high level of radiation (absorbed dose more than 2000 Mrad).

Interaction with groundwater leads to leaching from concrete dissoluble calcium compounds that results in change of structure of solid phases of cement matrix and, as consequent of it, to decrease of strength and change of other physicochemical properties of concrete. Besides the contact with groundwater is a reason of leaching of radionuclides from cement matrixes.

The negative influencing on durability of concrete can render also periodic cooling it up to temperature at which freezing a liquid in pores of cement matrix can take place that can lead to development of microcracks in structure of concrete that results in decrease of it strength, acceleration of leaching of components which are included in structure of concrete and radionuclides too. This factor may be important when the cemented objects are stored without weather protection in climatic conditions in which in winter period the probably long-lived temperature fall is lower -18-20 ^{0}C. Such situation is represented rather improbable. Besides the conducted special researches have shown that designed cement mixtures have the class of frost resistance not below F=300 according to GOST 10060-87 that guarantees preservation of properties of concrete on predictable period.

Thus as the main factor determining durability of concrete, used for conservation of radiation-hazardous objects, including waste tanks , and for solidification of the liquid radwastes, it is necessary to consider interaction of concretes with groundwater.

As a parameter of properties of concrete at estimation of their durability was chosen their compression strength because the change of this parameter mirrors changes happening in structure of concrete.

Corrosion of the cured cement mixtures at interaction with groundwater.

At contact of the cured cement mixtures with groundwater the leaching of $Ca(OH)_2$ takes place, rate of which is limited by Ca^{+2} diffusion through system of matrix - pores of cement mixture.

For estimation of service time of the cured mixtures on the basis of cement it is necessary to have the information on leaching rate of calcium from it. The data on leaching rate of calcium from mixtures No 2 and 21 are shown in the table 2.2.7.

Table 2.2.7.

Leaching rate of calcium from mixtures No 2 and 21.

No of composition	Time of leaching, days	Total fraction of leached ?? $(OH)_2$
2	3	0.0144
	6	0.0203
	10	0.00263
	20	0.00371
	30	0.00456
	40	0.00526
	50	0.00589
	60	0.00645
21	3	0.000181
	6	0.000235
	10	0.00033

	20	0.000465
	30	0.000572
	40	0.00066
	50	0.000738
	60	0.00081

The diffusion coefficients of calcium in mixtures No 2 and 21, calculated on the basis of the data of table 2.2.7. are equal:

For mixture No 2 - $3.1*10^{-15}$ m^2/sec;

For mixture No 21 – $3.8*10^{-16}$ m^2/sec

These values of a diffusion coefficients of calcium in mixtures No 2 and 21 were used for estimation of service time of these mixtures at a contact with groundwater.

. For calculation of speed of leaching Ca it is accepted:

- the package with the cemented radwaste has the form of barrels with diameter 1 m which are located in storage as a dense packing thus the volume of groundwater directly interacting with composition is equal to 0.215 m^3/m^3 of cement mixture;
- the leaching rate of Ca depends on speed of groundwater current in storage therefore calculations were made for three speeds of water current : 1, 10, and 100 m / years;
- for calculation of Ca leaching rate it was adopted the model, according to which the cylindrical units of cement mixtures are completely filled up with groundwater, which is contacting with mixture definite time dependent on speed of water current in storage then it is substituted on a fresh portion of groundwater and the procedure repeats;
- the concentration of Ca in groundwater is small in comparison with it concentration in pores of cement mixture that corresponds to maximum rate of Ca leaching from mixture.

The solution of a diffusion equation for this case are known (25). In the table 2.2.8. service time of mixtures No 2 and 21 calculated on the basis of above presented reasons are shown.

Table 2.2.8.

Calculated service time of compositions N 2 and 21 at different speeds of water current in storage of cemented radwaste .
(The packages of cemented radwaste have the form of barrels with diameter 1 m).

No of composition	Speed of water current , m / year	Service time of composition, years
2	1	$2.5*10^3$
	10	$7.9*10^2$
	100	$2.5*10^2$
21	1	$7.0*10^3$
	10	$2.2*10^3$
	100	$7.0*10^2$

As it is visible from the table 2.2.8.,estimated service time of composition N 2 at speed of groundwater current in storage of radwastes up to 10 m / years will exceed demanded value 500 years, however at more high speed of groundwater in storage it stability to calcium leaching can be not enough to guarantee safe storage of the cemented radwastes on given term. Considerably best parameters has the composition No 21 for which even at speed of groundwater current 100 m / years calculated service time is 700 years, that notably exceeds time (500 years) of radwastes storage containing Cs-137 and Sr-90, demanded for practically full decay of these radionuclides.

Research of cementation of simulated radioactive pulps on MCC.

The researches on usage of cementation for solidification of radioactive pulps was made on Mining-Chemical-Combine. The portland cement M 400 was used in these researches. Permissible quantity of pulps entered in cement mixture is determined, which allows to obtain the cured cement grout corresponding to the Russian requirements (RD 9510497-93 “Quality of compounds, obtained at cementation of low and middle active liquid radioactive wastes. Specifications. MINATOM of Russian Federation, 1993).

The Main results of carried out researches are presented below.

In the table 2.2.9 the data on influence of quantity of the pulp entered into cement mixture on strength of cured mixture are shown. 10 % of bentonite from weight of cement was entered into cement mixture as the component for decreasing of leaching rates of radionuclides from cured cement nixtures..

The table 2.2.9

Influencing of quantity of pulp on strength of cured cement mixtures.

Composition of cement mixture.			Weight water/ cement ratio	Compressive Strength, MPa
Cement / Bentonite ratio	The contents, % weight.			
	Cement + Bentonite	Pulp		
10: 0	100	0	0.7	18.0
9: 1	100	0	1.0	14.5
9: 1	90	10	1.0	17.0
9: 1	80	20	1.0	18.0
9: 1	70	30	1.0	13.0
9: 1	60	40	1.0	3.0
9: 1	20	80	1.0	1.0

The data, showed in the table 1, demonstrate that satisfactory compression strength, equal to 13 MPa, have samples of cured cement mixtures containing up to 30 % weight of pulp. Increasing of content of pulp in cement mixture above this limit leads to sharp decreasing of compressive strength and these mixtures does not fulfil conditions of the Russian standards of safety (RD 9510497-93).

The conducted researches have allowed to recommend for solidification of radioactive pulps the following composition of cement mixture:

- binding – portland cement M400;
- bentonite - 10 % from weight of cement;
- radioactive pulp - about 30 % from weight of cement;
- water/cement ratio 1: 1.

The leaching rate of Cs-137 and Pu-239 was studied for cement mixtures, compositions of which are shown in the table 2.2.10

The table 2.2.10

Composition of cement mixtures for research of leaching rate of Cs-137 and Pu-239.

N Of sample	Composition of cement mixtures, weight %						
	Cement	Bentonite	Clinoptilolite	Rad-waste pulp	Water/ cement ratio	Radio-nuclide	Activity Ci/kg
1	100	-	-	35	1.0	Cs-137	2.0x10-3
2	90	10	-	33	1.1	Cs-137	1.8x10-3
3	95	5	-	34	1.1	Cs-137	1.9x10-3
4*	91	9	-	33	1.0	Cs-137	1.8x10-3
5	91	-	9	33	1.0	Cs-137	1.8x10-3
6	100	-	-	35	1.0	Pu-239	1.3x10-2
7	91	9	-	33	1.1	Pu-239	1.3x10-2
8*	91	9	-	33	1.0	Pu-239	1.2x10-2

The notice - * in samples 4 and 8 the bentonite dried up to constant weight at temperature 100 ^{0}C was added.

The data on leaching rate of Cs-137 and Pu-239 from investigated cement mixtures are presented in the tables 2.2.11 and 2.2.12.

The table 2.2.11.

Leaching rate of Cs-137 from cement mixtures (composition of samples presented in table 2.2.10).

Contact time of sample with water, days.	Leaching rate of Cs-137 from samples of cement mixtures, g/sm^{2}xday				
	1	2	3	4	5
15	$8.5x10^{-3}$	$1.3x10^{-3}$	$3.1x10^{-3}$	$1.6x10^{-4}$	$1.6x10^{-3}$
28	$5.1x10^{-3}$	$8.8x10^{-4}$	$1.8x10^{-3}$	$8.0x10^{-5}$	$5.0x10^{-4}$
58	$2.6x10^{-3}$	$4.1x10^{-4}$	$8.2x10^{-4}$	$3.0x10^{-5}$	-
150	$2.0x10^{-3}$	$8.5x10^{-4}$	$3.8x10^{-4}$	$8.0x10^{-5}$	-
190	$1.0x10^{-3}$	$1.3x10^{-4}$	-	$2.5x10^{-5}$	-
265	$1.0x10^{-3}$	$1.0x10^{-4}$	-	$1.0x10^{-5}$	-
350	$1.0x10^{-3}$	-	-	-	-

The table 2.2.12

Rate of Pu-239 leaching from cement mixtures (composition of samples presented in table 2.2.10).

Contact time of sample with water, days.	Rate of Pu—239 leaching from samples of cement mixtures, g/sm^{2}xday.		
	6	7	8
15	$9.2x10^{-5}$	$3.3x10^{-5}$	$1.0x10^{-5}$
48	$2.1x10^{-5}$	$1.0x10^{-5}$	$4.0x10^{-7}$
140	$2.0x10^{-7}$	$2.0x10^{-7}$	$1.0x10^{-7}$

The results of the carried out experiments have shown that the addition of clinoptilolite and bentonite, especially bentonite dried up to constant weight at 100 ^{0}C, essentially decreases leaching rate of Cs-137 from cement mixtures. In case of plutonium the influencing of bentonite addition on leaching rate is not so significant.

The experiments, results of which presented above, were carried out without exchange of contact solution that imitated emergency submergence of radwastes storage.

Magnesium-phosphate cement for solidification of the liquid radwastes and pulps directly in waste tanks.

In a number of cases there is a necessity of solidification of the liquid radwastes or radioactive pulps directly in waste tanks in which the stirring is very difficult or is impossible. The carried out researches have shown that for this purpose it is perspective to use cements on the phosphate basis which is forming at interaction of phosphoric acid with compounds of different metals, in particular with caustic magnesite.

In work /27/ the main positions reflecting influence of chemical composition of phosphate compounds on capability and on conditions of development of astringent properties were formulated:

1. The phosphate cements are received at interaction of phosphoric acid with powdery materials (oxides, hydroxides, phosphates etc.) in broad range of their composition.
2. The intensity of development of astringent properties in system " oxide - phosphoric acid " is objective function of value of ionic potential (relation of a charge to effective radius) cation of oxide.
3. The major factor determining a capability of obtaining of phosphate cement is the right selection of ratio of reaction rate of phosphoric acid with oxide (generation rate of germs of crystalls) and speed of gelation of cement.

Application of cements of phosphate solidification for an immobilization of radioactive waste is perspective from the point of view of strong fixation of radionuclides in structure of cement mixture. The large group of mineral ion exchangers is known on the basis of indissoluble salts of phosphoric acids

/28/, such as phosphates of barium, tin, zirconium, thorium etc ., which effectively retain different radionuclides. It gives the basis to guess that after immobilization of radioactive waste in phosphate cement there will be a strong fixation of radionuclides.

On the basis of above-stated it was offered technique /28/ of solidification of liquid radioactive wastes with applying as binding of magnesium-phosphate cement.

In according to this technique in radioactive waste the concentrated ortophosphoric acid and caustic magnesite must be sequentially added at weight ratio radwaste pulp: ortophosphoric acid: caustic magnesite equal to 1:0,3:0,5 accordingly. After that the mixture is maintained during time which is necessary for solidification. The plant tests /30/ on solidification of 70 m^3 of radioactive ferrocyanide pulp directly in defective radwaste storage tank - AG - 8301/1 having volume 3200 m^3 at radiochemical plant MCC was made. On technological calculations for solidification of 70 m^3 of pulp having ratio solid to liquid phase equal 1:2, it was required 25,8 t of concentrated phosphoric acid and 42 t of caustic magnesite. The distinctive feature of technological process of solidification was that in the tank the system of mixing was absent that complicated process of solidification. As a result of it, after supply in the tank of all quantity of phosphoric acid and the first portion (3,5 tons) of caustic magnesite the dense layer of magnesium-phosphate cement was formed on the surface of pulp that precluded entry of caustic magnesite into mixture of pulp and phosphoric acid and because of that has not given capability to finish process of solidification.

In laboratory conditions the researches were carried out to choice the conditions of the introducing in waste tank of phosphoric acid and caustic magnesite which would give a capability to make solidification in all volume of waste. The results of these experiments show that for this purpose it is necessary:

- concentration of phosphoric acid in waste must be 90-106 g/l;
- all demanded quantity of caustic magnesite should be entered in one portion during possible short time;
- after the addition of magnesite the system must be stayed during 100 day for solidification of the formed cement.

In accordance with recommendations prepared on the basis of results of laboratory researches, the following technological decisions were adopted:

- to add in tank calculated quantity of water for decreasing of density of aqueous phase of pulp from 1.412 g/sm^3 to 1.32 g/sm^3 and decreasing of phosphoric acid concentration from 196 g/l up to 106 g/l;
- the addition of all demanded quantity of magnesite carry out in one portion during possible short time;
- after magnesite addition to add in tank 1.5 t of phosphoric acid to guarantee obtaining of magnesium-phosphate cement on the surface of waste.

After carrying out of all these operations reaction mixture in tank was stored during 100 days. The samples, which were taken after that, showed that magnesium-phosphate cement monolith was formed in whole volume of radwaste in tank.

The carried out works have allowed to solidificate 70 m^3 of radioactive pulp that gave possibility to localize radionuclides in cured cement mixture and to eliminate leakage of radionuclides from defective tank in ground waters.

The installations for cementation of radioactive wastes developed in Russian Federation.

The method of cementation is considered in Russian Federation now as the main method for reprocessing of low and middle active radwastes. In this connection the significant attention in Russia was paid to developing of equipment for this purpose

To the present time the number of such installations is designed, the brief description and the characteristics of which are presented below.

1. Installation of cementation on RTP "ATOMFLOT".

The installation is designed for cementation of radioactive wastes which are obtained at exploitation of atomic-powered icebreakers.

The radwastes of different chemical and phase composition can be processed on this installation:

- saline solutions including brines and concentrate from the installation of membrane cleaning;

- hydroxide pulps;
- pulps of inorganic sorbents.

Cement mixtures obtained on the installation, are packaged in protective concrete containers UNZK-150-1.5P with capacity 1.5 m^3. Activity of cement mixtures must be not more than 6×10^{-3} Ci/kg.

Structure of the installation.

The installation consists of five blocks:

- system of cement feed in the block of mixture preparation;
- system of radwastes preparing and ts feed in the block of cement mixture preparation;
- block of cement mixtures preparation and discharging of it in the container;
- system of containers transportation;
- the control system.

Characteristics of the installation.

1.	Concentration of salts in radwaste solutions, g/l	Up to200
2.	pH of solutions	7 - 10
3.	Specific activity of solution, Bq/l	Up to 1.86×10^7
4.	Concentration of solid phase in pulp, g/l	Up to 200
5.	Specific activity of pulp, Bq/l	Up to 2.22×10^8
6. 7.	Consumption of cement, kg/hour Consumption of the additions, kg/hour	600-1200 60-120
8.	Consumption of compressed air (6 atm.), kg/hour	Up to 50
9.	Yield of waste gases, kg/hour	Up to 60
10.	Specific activity of overflow gases, Bq/m^3	Not more than 3.7
11.	Weight water-cement ratio	0.4-07
12.	Weight fraction of salts in cement mixture, %	Not less than 7
13.	Weight fraction of pulps solid phase in cement mixture, %	7-8.5
14.	Degree of the container filling, %	Not less than 85
15.	Consumed electrical power	No more than 32.5 kw
16.	Temperature of air at exploitation	5 – 40 ^{0}C

2. Installation for cementation of the ash from incineration of combustible solid radwastes on Smolensk Nuclear Power Plant..

The installation of cementation is designed for cementation of the ash which is obtained at incineration of solid combustible radioactive wastes. It contains up to 99 % of radionuclides from their quantity in an initial combustible solid radioactive waste.

The installation includes:

- bunker for cement;
- vessel for water;
- vessel for collecting of ash with volume 200 l;
- the automotive protective container;
- auger feeder for ash;
- feeder for batching of cement;
- unit for mixing of ash, cement and water.

Characteristics of the installation.

Filled with ash the barrel located in the automotive protective container, goes on the installation of cementation.. The barrel with ash is connected to the cover having the electric drive with mixer and admissions for cement and water.

The cement feeds from bunker with portions up to 70 kg.

The water feeds from vessel with portions up to 70 l.

The consumption of cement - 0.6 - 1.0 kg / hour.

Quantity of cement compound - 1.6 - 2.6 kg / hour.

Specific activity of cement compound - up to $1x10^{-2}$ Ci/kg.

3. Installation of cementation liquid radwastes in building 101 NITI.

The installation is designed for cementation of liquid radwaste concentrates. The solidification is made directly inside primary packaging (barrel).

The installation can be utilized on objects, on which the volume and regularity of liquid radwastes formation make stationary installation economically inexpedient.

The portland cement M500 (GOST 10178-85) will be used for cementation of liquid radwaste concentrates. The different synthetic and natural inorganic sorbents (nickel ferrocyanide on silicogel (NGA- "Celeks - CFN" TU 95-2385-92, bentonite clay (GOST 7032-75), clinoptilolite, vermiculite etc.) in quantity 5-15 % from weight of cement can be applied as sorbing additives.

The modular design is adopted for installation that allows to convey it on different objects.

Characteristics of the installation.

1.	Installation capacity: - on concentrate of liquid radwases , m^3/year - on cement compound, m^3/year:	 100 Up to 142
2.	Concentration of salts in radwaste solutions, g/l	Up to 200
3.	Quantity of components on cementation of one barrel: - concentrate of liquid radwastes , l - cement, kg - sorbing components, kg	 126 180 10
4.	Weight of barrel with cement compound, kg	Up to 350
5.	Specific weight of cement compound, kg / l	1.8-2.0
6.	Volume of barrel withcement compound, l	200
7.	Radiochemical composition of liquid radwastes	Co-60, Sr-90, Cs-137
8.	Average chemical compos ition of liquid radwaste concentrates : HCO3 Cl- SO4-2 NO3- Ca+2 Mg+2 Na + K + NH4 + Fe+3 Petroleum PH Density, g/l	 25-35 % 18-25 % 10-15 % 1-2 % 8-12 % 1-5 % 8-12 % 4-8 % 0.1-0.3 % 1-3 % 1.5 ?/? 6.5-8.5 1.045
9.	Specific activity of liquid radwastes concentrate , Ci/l	$1x10^{-5}$
10.	Consumed electrical power, kw	Up to 5
11.	Operational mode of the installation	Periodic
12.	The design of installation	Modular
13.	Quantity of modules at transpotation	2
14.	Overall dimensions, mm Module 1 Module 2	 3500x1350x2000 3500x2600x2000
15.	Weight of the installation (net), kg Module 1 Module 2	 1000 2000

4. Installation for cementation radwastes on Moscow NPO "Radon".

The modular installation of cementation with the vortex mixer designed for processing of liquid radwastes concentrates, pulp of ion-exchange resins and inorganic sorbents.

Cement compound obtained on the installation with activity not more than $5x10^{-4}$ Ci/kg is packaging in steel barrels with volume 200 l.

The installation consists of following modules:

- preparation of cement mixture;
- transport;
- preparation of pulps;
- preparation of inorganic sorbents ;
- preparation of ionexchange resins;
- pumps-batchers;
- ventilation;
- control panel;
- electric switchboard.

Characteristics of the installation.

1.	Capacity on cement grout, m^3/hour	1.5
2.	Capacity on liquid radwastes (at (water/cement ratio – 0.75), m^3/hour	1.0
3.	Concentration of salts in radwaste concentrates , g/l	Up to 1000
4.	Cementing material	Portland cement M400 and M500
5.	Density of cement grout, kg / l	1.5-2.2
6.	Rate of Cs-137 leaching, g/ sm^2xday	$2x10^{-3}$ – $4x10^{-5}$
7.	The sorbing additives	Bentonite, natural zeolites
8.	Quantity of the sorbing additives, % of weight of cement grout	1 – 5 %
9.	Quantity of superplasticizer, % of weight of cement grout	0.1 – 1.0 %
10.	Water/cement ratio	0.4 - 0.8
11.	Specific activity of reprocessed solutions, Ci/l	Up to $1x10^{-3}$
12.	Operational mode	Periodic or continuous
13.	Process control	Manual or automatic
14.	Primary packaging	Steel barrel V=200 l

15.	Service life of the installation, years	Not less than 10
16.	Total mass, metric ton	27
17.	Consumed electrical power, kw	150

The modular installation of cementation with the vortex mixer provides high quality of cement grout, has capability of fast change of its structure depending on kind of reprocessed radioactive waste, allows to reprocess waste with the considerable contents of solid suspended matters and enables to lower quantity of secondary waste (decontamination water used for washing of the mixing chamber).

5. Installation of cementation of liquid radioactive waste "ATOMmash".

The modular installation of cementation is designed for solidification of concentrates obtained at evaporation of liquid radwastes, pulps of filtering maretials and liquid organic waste of low activity level.

Cement grout with specific activity no more than $1x10^{-4}$ Ci/kg obtained on the installation is packaged in steel barrels with volume 200 l.

Characteristics of the installation.

1.	Capacity on cement grout, m^3/hour	1.0
3.	Concentration of salts in radwaste concentrates , g/l	Up to 200
4.	Cementing material	Portland cement M400 and M500
5. 6.	Density of cement grout, kg / l The sorbing additives	1.5-2.2 Bentonite, natural zeolites
7.	Quantity of the sorbing additives, % of weight of cement grout	1 – 5 %
8. 9.	Quantity of superplasticizer, % of weight of cement grout Water/cement ratio	0.1 – 1.0 % 0.4 - 0.8
10.	Specific activity of reprocessed solutions, Ci/l	Up to $5x10^{-4}$
11.	Operational mode	Periodic or continuous
12.	Process control	Manual or automatic
13.	Primary packaging	Steel barrel V=200 l

In conclusion the basic characteristics of the installations of cementation designed in Russia are presented below.

Basic characteristics of the Russian installations of cementation.

The characteristics of the installations	Place of location of the installation				
	RTP "ATOMFLOT"	Smolensk NPP	RNC "NITI"	Moscow SPA "RADON"	ATOMmash
Composition of the cemented radwastes	Liquid radwases with salt content up to 200 gl, hydroxide pulps and pulps of inorganic sorbents	Ash from incineration of combustible solid radwastes.	Liquid radwastes with salt content up to 200 7?, pulp of ion-exchange resins and inorganic sorbents	Liquid radwastes with salt content up to 1000 g/l, pulps of ion-exchange resins and inorganic sorbents	Residues from evaporation of liquid radwastes, pulps of filtrating materials, liquid organic Low level radwastes
Capacity on cement – grout	Up to 1.0 m^3/hour	Up to 2.6 Kg / hour	Till 0.13 m^3/hour	Up to 1.5 m^3/hour	Up to 1.3 m^3/hour
Specific activity of cement grout, Ci/kg	Up to $6x10^{-3}$	Up to $1.0x10^{-2}$	Up to $1.0x10^{-5}$	Up to $5x10^{-4}$	Up to $4x10^{-4}$
Type of cement grout package	The protective concrete container UNZK-150-1.5P	Steel barrel V=200 l	Steel barrel V=200 l	Steel barrel V=200 l	Steel barrel V=200 l
Design features of the installation	Fixed location in building with radiation protection.	Fixed location in building with radiation protection.	The modular installation	The modular installation	The modular installation
Development stage	Installation is mounted on RTP "ATOMFLOT" and now is in industrial testing	The working documentation is designed and confirmed by regulatory autorities.	The cold tests of experimetal Installation was carried out	Installation is mounted on Moscow NPO "RADON" and now is in industrial exploitation	The working documentation is designed and confirmed by regulatory autorities.
Enterprise-Designer	SverdNIIchim-mash	GI VNIPIET	RNC "NITI"	Moscow NPO "RADON"	"ATOMMASH"

3. LARGE SCALE EXPERIENCE OF APPLICATION OF CEMENTATION METHODS FOR CONSERVATION OF RADIATION-HAZARDOUS OBJECTS IN RUSSIA.

Up to the present time in Russia there is no experience on applying of cementation for conservation (closure) of waste tanks. It is connected with that waste tanks in which the main part of the accumulated radwastes is stored in Russia manufactured from corrosion-resistant stainless steel that reduces to minimum hazard of affect of these wastes on environment.

The radiation-hazardous objects of nuclear fleet removed from exploitation have significantly larger hazard to environment therefore on conservation of these objects it is paid prime attention now.

To the present time the large scale works on conservation of two objects of nuclear fleet were made:

- floating technical base "Lepse" (ship assigned to maintenance of atomic ice breakers) of Murmansk marine shipping company;
- two reactor compartments of nuclear submarines in former training center of the Russian Navy in Paldisk, Estonia.

1. Cementation of intertank space of burned up fuels storage on "Lepse".

For adjustment of radiation conditions on "Lepse" in accordance with the present Russian standards in 1990 it was accepted the decision to fill in intertank space of burned up fuels storage on this ship with concrete - conserving agent.

The monolith formed by concrete - conserving agent should become an engineering barrier ensuring increase of strength of all construction and also as immobilization barrier to prevent possible migration of radionuclides from burned up fuels storage in environment.

1.1. Design of intertank space of burned up fuels storage on «Lepse».

The burned up fuels storage on "Lepse" represents located in the nose of the ship rectangular compartment with metallic walls inside of which there are placed two cylindrical tanks with burned up fuel assemblies. Assemblies are placed in capsules arranged by concentric series and intercapsule space is filled by water serving for cooling of capsules. The water circulation is provided with special system. The tanks are closed by rotary covers permitting selectively to open access to capsule and assembly which must be overload.

Burned up assemblies storage has the internal sizes 4800x10150 mm and walls with thickness 420-450 mm. The tanks having diameter 3600 mm and height 3440 mm are made from stainless steel with thickness 10 mm.

For creation of engineering barrier in intertank space it was necessary to fill it with 102 m^3 of cement mixture. It was made with the purpose of strengthening of storage in view of possible accidents at all technological stages of the management of burned up assemblies and at long-term storage in repository.

1.2. General requirements to materials of engineering and immobilization barriers.

To the cured cement mixtures:

- to provide protection of metallic construcrions of burned up assemblies storage from corrosion under affect of the external factors; do not accelerate and do not instigate corrosion of available protective contours of storage;
- to be steady against long-lived radiation effect;
- to have high durability, the value of compressive strength of concrete - conserving agent at the end of calculated storage time (500 years) should be not less than 100 kg /cm^2 (10 MPa);
- to be non-toxic, flame safety and fireproof.

The mixtures prepared for conservation of radiation-hazardous objects must:

- to have flow characteristics indispensable for full filling of space of the complex configuration;
- the used mixtures should be not toxic, are explosion-proof and fireproof;
- mixtures and their rheological characteristics should provide a capability of their mechanical preparation, transport, supply and stacking with usage of equipment serially produced and used in building;
- the materials should be not deficient

1.3. General requirements to technology of conservation of the object:

- the technology of object filling with cement mixtures should be completely mechanized and remotely operated;
- the technological circuit for object cementation should be highly reliable;
- mounting of the part of technological circuit for packing of mixtures in radiation-dangerous zone and process of packing of mixtures should requires of minimum workers participation at carrying out of all requirements and standards of radiation safety.

1.4. Materials and technology for cementation of intertank space.

In accordance with data of radiation safety service of Murmansk marine shipping company the integral radiation dose of concrete in intertank space of burned up assemblies storage for estimated time of storage of "Lepse" will not exceed $1x10^8$ rad. Thus for this object it is not required applying of special highly radiation stable concrete. Outgoing from this for filling of intertank space on "Lepse" the concrete on the basis of portland cement and customary fillers (sand, Breakstone) satisfying to GOST 10260-80 was selected. Such concrete mixtures also fit to all, formulated above, requirements: they are not toxic, are fire- and explosion-proof, the materials to their preparation are accessible and mixtures preparation, transport, supply and stacking can be made with usage of serially produced and widely applicable in building equipment.

Due to modern achievements of concrete technology, it is possible to receive not stratified homogeneous mixtures possessing high fluidity, capable to stuff completely internal volumes of objects of the complex configuration.

At the same time, it, as a rule, demands of the heightened consumption of cement. In this connection special attention was paid to limitation of mixture temperature growth during cementation and subsequent curing of concrete monolith to except thermal crackforming in concrete. On the basis of the carried out experimental works for the solution of this problem it was determined to do cementation of intertank space of burned up assemblies storage on "Lepse" in October when climatic conditions near Murmansk allow to receive concrete mixtures with temperature 5-10 ^{0}C without addition cooling. This measure and usage for cooling of curing concrete of cooling system of burned up fuel storage has allowed to limit temperature rise in concrete no more than 35-40 ^{0}C that completely eliminated hazard of dangerous thermal stresses in concrete monolith.

For cementation of intertank space on "Lepse" the concrete mixture on the basis of low aluminate portland cement was used the composition of which is shown in the table 1.1.

Table 3.1.

Composition of concretes used for cementation of intertank space of burned up assemblies storage on "Lepse"

The naming of materials, parameters of concrete mixture	Units of measurement	The consumption of materials on 1 m^3 of concrete	
Portland cement:			
of the mark: "400"	Kg	415	-
"500"	Kg	-	380
Sand	Kg	650	740
Breakstone (size up to 20 mm)	Kg	975	1020
Water : cement ratio		0,48÷0,5	0,42÷0,45
Additives:			
Superplasticizer		S-3	S-3

Concrete mixtures were prepared on usual concrete plant and transportation of concrete to "Lepse" was made with help of automobile concrete mixers.

The system consisting of the "Wartington" concrete pump and concrete line connected the pump placed on a coast with a receiving hutch on "Lepse" was used for stacking of concrete. In total in intertank space was pumped 110 m^3 of concrete, on what it was required about 6 hours.

The check tests have shown that the concrete monolith has no defects, the compression strength of check samples of concrete in the age of 28 day was equal to 27.6mPa. Thus it is possible to draw a conclusion, that the conservation of burned up assemblies storage on "Lepse" has passed successfully.

2. Decommissioning and preparing to safe storage during 50 years of reactor compartments nuclear submarines in training center of Russian Navy in Paldisk Estonian Republic.

Training center of the USSR Navy in Paldisk was put into exploitation in 1967. There were two real reactor compartments of nuclear submarines and necessary power equipment, (steam generators, turbines etc.). The nuclear power plants in training center were working up to 1989.

After finding by Estonia of the status of the independent state it was raised the question about liquidation of training center of Russian Navy in Paldisk. In accordance with intergovernmental Agreement between Russia and Estonian Republic nuclear objects of training center must be decommissioned and prepared to long-term (50 years) safe storage till 30 September, 1995.

In accordance with this agreement it was carried out complex of works including:

- comprehensive engineering inspection of the objects;
- elaboration and adjustment of the concept of decommissioning and preparation to safe storage of nuclear objects of training center;
- elaboration of the project documentation for carrying out of the works and technology of cementation of reactor compartments and other systems of nuclear power plants;
- hermetic sealing of reactor compartments;
- conservation of reactors, equipment and systems of nuclear power plants;
- building of protective shelters (sarcophagi).

The complex of buildings and facilities of a training center provided realization of all technological operations indispensable for exploitation of reactors and other systems of nuclear power plants in conditions maximum approximated to real.

At exploitation of the installations the carrying out of the following operations was required:

- storage and audit of fresh fuel assemblies, rods of management and protective system etc.;

- recharge of active zones of the reactors;
- replacement of separate components of reactors and equipment of the steam generating installations;
- cooling of the burned up fuel assemblies;
- cooling of radioactive components of reactors and equipment of steam generating installations;
- loading of cooled burned up fuel assemblies on an external transport;
- disposal of the radioactive equipment.

The systems of training center provided carrying out of all these operations.

2.1. Comprehensive engineering inspection of the nuclear objects of the Navy training center.

Comprehensive engineering inspection of nuclear objects of the Navy training center was made to obtain the data necessary for designing of technology and documentation for decommissioning of these objects. The inspection was conducted in two stages. The first stage was done in January, 1994 for elaboration of the concept of decommissioning of nuclear objects and second stage was made after discharge of nuclear fuel from rectors and disposal of liquid and gaseous mediums from systems and equipment to obtain the indispensable addition data for designing technology of objects conservation and preparing of project-budget documentation.

Engineering inspection included:

- estimation of actual condition of buildings, equipment and systems of reactor compartments and other equipment of nuclear power plants;
- itemization of the design and technological solutions on equipment and systems disassembly and their preparing to conservation and long-term storage;
- full-scale measurements of overall dimensions of a reactor compartments necessary for designing of protective shelters;
- radiation examination of reactor compartments, buildings and territory.

On the base of results of engineering inspection the following conclusions were made:

- the dose rate of γ- radiation on territory of training center is equal to 16 -

23 μR/h and corresponds to background values for the given terrain;

- technical condition of buildings and equipment is satisfactory;
- the radiation examination has shown, that in all placement, excepting reactor compartments, there is no excess of background values β- and γ-activity;
- γ -radiation dose rate in reactor compartments is equal to 0,1 - 23 mR/h.

The obtained results of engineering examination have allowed to accept the optimal and economically reasonable solutions at elaboration of technology and project documentation.

2.2. The concept of decommissioning and preparation to safe storage of nuclear objects of training center

The modern concepts of decommissioning and preparation for long-term storage of nuclear objects envisage their deep decontamination and disassembly, including constructions with induced radiation. The realization of the similar concepts is connected with formation big quantity of highly active solid radioactive wastes and secondary liquid radioactive wastes. Processing and solidification of these wastes are very complex, expensive and long-timed processes requiring of creation of new productions.

In conditions of a nuclear training center in Paldisk the indicated concept could not be used because of very short period of time which was given on all works (nuclear objects of training center must be delivered to Estonian party till September, 1995) and also because of the limited financial capabilities of Russian party. All this dictated necessity of acceptance of new more optimal solutions.

In view of IAEA principles of safety and technical criterions for underground disposal of the radioactive wastes (serial of issues on safety, No 99, Vienna 1990, the section 3 " Principles of safety ") was necessary to find a solution answering to the following requirements:

- providing of safety ;
- liability before the future generations;
- consequences in the future: " It is necessary to ensure a degree of isolation of highly radioactive wastes at such level that there were absent predictable kinds of risk for people health or consequence for an

environment in the future, which would not be acceptable today ".

For carrying out of the indicated conditions and requirements, on fifty years period of reactor compartments storage in accordance with really existing radiation situation, it was necessary to create a number of engineering barriers precluding migration of radionuclides in environment and eliminating unauthorized admittance of the people into reactor compartments.

The following system of rector compartm ents preparation to long-term storage was adopted to realization:

- preparing of devices and systems of reactor compartments and steam generating installation to conservation with the help of concrete - conserving agents;
- creation immobilizing and engineering barriers inside reactor compartments;
- building protective sarcophagi outside of reactor compartments designed for protection of reactor compartments from extreme impacts natural and technical origin within 50 years.

The preparation of devices and submarine inner compartments and of steam generating installation to constitutes barriers for disassembly and deleting of the uncontaminated equipment was done prior to sealing reactor compartments. The hermetic sealing involved plugging pipe lines, holes in body of reactor compartments with grout prior to the compartment test on air-tightness. Besides these works dehumidifying of air inside reactor compartments and deposition of outside protective coatings on bodies of compartments were made.

In result of engineering inspection the list of equipment and devices of reactor compartments must be conservated with help of special concrete mixtures was determined.

On the basis of results of researches of properties of special portland cement mixtures, presented in this review, for conservation of reactor compartments the mixture No 2 was selected, in which as the main components of mixture except of cement will be used finely divided shungizit and shungizit sand. For giving high fluidity to concrete mixture in it the superplasticizer S-3 was added, and for increase of concrete gas permeability the component SDO was introduced.

The composition of concrete, which were used for conservation of reactor compartments and the creation of external shelters (sarcophagi) are shown below.

Composition of concrete mixture No 2 for conservation of reactor compartments (consumption of materials on 1 m^3 of concrete):

Portland cement of Pikalev plant M 500 - 726 kg
Shungizit filling material - 259 kg
Shungizit sand - 621 kg
Water - 372 kg
Superplasticizer S-3 - 5.4 kg
SDO - 0.4 kg

Mean density of concrete of 2000 kg / m^3

For building external shelters the concrete of the following composition was used (consumption of materials on 1 m^3 of concrete):

Portland cement of Pikalev plant of M 400 - 400 kg
Sand - - 512 kg
Breakstone (fraction no more than 10 mm) - 836 kg
Water - 246 kg
Superplasticizer S-3 - 6.0 kg

Mean density of concrete of 2000 kg / m^3.

All activities on conservation of the equipment of reactor compartments and building of external shelters were finished to the end of September, 1995 and were adopted by the Estonian party which stated that the carried out works guaranteed safety of decommissioned reactor compartments on demanded period (50 years).

CONCLUSION.

The results of done researches have shown that the composition of portland cement mixtures has high influence on properties of these mixtures. Changing composition of cement mixtures it is possible to obtain mixtures possessing complex of properties in the most degree adequate to areas of their applying.

In the present review we considered two fields of application of portland cement mixtures:

- conservation (closure) of radiation - hazardous objects of atomic engineering, in particular of waste tanks, reactor compartments of vessels with nuclear energy installations etc.;
- solidification and preparing for long-term storage of the liquid radwastes.

The optimal complex of properties of cement mixtures designed for applying in each of introduced above areas can be various.

So at usage of these mixtures for conservation of radiation - hazardous objects with high-level of radiation the important value has a high gas permeability of using mixtures because that allows to ensure going out of radiolytical gases without disturbance of concrete monolith integrity. For cement mixtures intended for solidification of the liquid radwastes middle and low level of activity the requirements of low leaching rate of radionuclides and high stability of mixture at affect of groundwater and other factors of environment of radwastes storage are going on the foreground.

Considering from these positions results of researches of portland cement mixtures it is possible to make following conclusions.

For conservation (closure) of radiation-hazardous objects with high level of radiation for which the release of significant amount of radiolytical hydrogen is possible, the most favourable combination of properties has the composition No 2, as it has a high gas permeability that provides output of hydrogen from massive of the cured composition without disturbance of it integrity.

For conservation of radiation-hazardous objects with low radiation level, for which quantity of emanation of radiolytical hydrogen is insignificant, and for solidification of the liquid radwastes the preference is necessary to give to composition No 21, which has a low permeability, low rate of radionuclides leaching and highest stability to affect of groundwater from studied compositions on the basis of portland cement. Estimation of possible service time of composition No 21 showed that even in disadvantageous condition (groundwater flow in repository up to 100 m/year) durability of it must be more 500 years – the time which is necessary for practically full decay of Cs-137 and Sr-90.

The technologies of cementation of radiation-hazardous objects of nuclear fleet was elaborated with accounting of results of carried out investigation. This permitted to make up to date the large scale works on conservation with help of concrete of two objects of nuclear fleet:

- floating technical base "Lepse" (ship assigned to maintenance of atomic ice breakers) of Murmansk marine shipping company;
- two reactor compartments of nuclear submarines in former training center of the Russian Navy in Paldisk, Estonia.

In the short review it is impossible to present all results obtained in cement mixtures researches carried out in Russia. But presented results permits to formulate key technical questions that can be the matter of analytical and experimental investigations in the potential Part 2 of the project:

- elaboration of new compositions of mixtures on the base of cement and other inorganic binders for liquid radwastes solidification and for closure of tanks and other radiation-hazardous objects;
- looking for additives to cement which can strongly retain in cement matrixes the certain radionuclides (Cs-137, Sr-90, Tc-99, I-129, Np-237);
- investigation of radiation-chemical processes taking place at cement mixtures irradiation to look up of the ways of decreasing of hydrogen generation.

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5.0 ATTACHMENT B

Russian Low-Level Radioactive Waste Regulations (in Russian)
January 1, 2002

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ФЕДЕРАЛЬНЫЙ НАДЗОР РОССИИ
ПО ЯДЕРНОЙ И РАДИАЦИОННОЙ БЕЗОПАСНОСТИ
(ГОСАТОМНАДЗОР РОССИИ)

ПОСТАНОВЛЕНИЕ

27 сентября 2000 г. МОСКВА № 7

Об утверждении и введении в действие федеральных норм и правил в области использования атомной энергии НП-019-2000 "Сбор, переработка, хранение и кондиционирование жидких радиоактивных отходов. Требования безопасности"

Федеральный надзор России по ядерной и радиационной безопасности

ПОСТАНОВЛЯЕТ:

Утвердить и ввести в действие с 1 января 2001 г. федеральные нормы и правила в области использования атомной энергии НП-019-2000 " Сбор, переработка, хранение и кондиционирование жидких радиоактивных отходов. Требования безопасности ".

Начальник
Госатомнадзора России Ю.Г. Вишневский

Невскому С.Я.
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для обеспечения [illegible]
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Федеральный надзор России по ядерной и радиационной безопасности
(Госатомнадзор России)

ФЕДЕРАЛЬНЫЕ НОРМЫ И ПРАВИЛА
В ОБЛАСТИ ИСПОЛЬЗОВАНИЯ АТОМНОЙ ЭНЕРГИИ

УТВЕРЖДЕНЫ
постановлением
Госатомнадзора России
от 27 сентября 2000 г.
№ 7

СБОР, ПЕРЕРАБОТКА, ХРАНЕНИЕ И КОНДИЦИОНИРОВАНИЕ ЖИДКИХ РАДИОАКТИВНЫХ ОТХОДОВ. ТРЕБОВАНИЯ БЕЗОПАСНОСТИ

НП - 019 - 2000

ВВЕДЕНЫ в действие
с 1 января 2001 г.

Москва 2000

УДК 621.039.58

СБОР, ПЕРЕРАБОТКА, ХРАНЕНИЕ И КОНДИЦИОНИРОВАНИЕ ЖИДКИХ РАДИОАКТИВНЫХ ОТХОДОВ. ТРЕБОВАНИЯ БЕЗОПАСНОСТИ

Госатомнадзор России
Москва, 2000

Настоящие федеральные нормы и правила "Сбор, переработка, хранение и кондиционирование жидких радиоактивных отходов. Требования безопасности" устанавливают требования к обеспечению безопасности при сборе, переработке, хранении и кондиционировании жидких радиоактивных отходов на ядерных установках, радиационных источниках, в пунктах хранения ядерных материалов и радиоактивных веществ, хранилищах РАО.

Нормативный документ выпускается впервые.

Нормативный документ разработан в Научно-техническом центре по ядерной и радиационной безопасности при участии Захаровой К.П., Масанова О.Л. (ВНИИНМ им. А.А Бочвара) Киселева В.В. (ФУМБ и ЭП при Минздраве России), Непейпиво М.А., Шарафутдинова Р.Б. (НТЦ ЯРБ).

При разработке нормативного документа рассмотрены и учтены замечания: ФУМБ и ЭП при Минздраве России, Госкомэкологии России, УЭ и СЭЯО Минатома России, ДБЭ и ЧС Минатома России, ВНИИНМ им. А.А. Бочвара, МосНПО "Радон", ВНИПИЭТ, ГНЦ "Институт Биофизики", Горно-химического комбината, Сибирского химического комбината, ПО "Маяк", концерна "Росэнергоатом", Ленинградской АЭС и др.

СОДЕРЖАНИЕ

нормами и правилами в области использования атомной энергии. Отнесение указанных веществ, материалов, изделий, приборов, оборудования и объектов к радиоактивным отходам определяется эксплуатирующей организацией и обосновывается в проекте ядерной установки, радиационного источника и пункта хранения.

17. **Переработка ЖРО** - технологические операции по сокращению объема, изменению агрегат-ного состояния и (или) физико-химических свойств ЖРО.

18. **Программа обеспечения качества** - документально оформленный комплекс организа-ционно-технических и других мероприятий по обеспечению качества, позволяющих руководству эксплуатирующей организации и(или) организаций, выполняющих работы и предоставляющих услуги эксплуатирующей организации, убедиться в том, что вся деятельность, влияющая на ядерную и радиационную безопасность, осуществляется в соответствии с требованиями федеральных норм и правил в области использования атомной энергии и других нормативных документов.

19. **Сбор ЖРО** - сосредоточение ЖРО в специально оборудованных емкостях.

20. **Упаковка РАО** - упаковочный комплект (контейнер) с помещенными в него РАО, подготов-ленный для транспортирования, и (или) хранения, и (или) захоронения.

21. **Цементирование ЖРО** - включение ЖРО в цементный матричный материал.

22. **Хранение ЖРО** – временное содержание ЖРО в емкостях (хранилищах), обеспечивающих защиту от радиации и изоляцию ЖРО, с намерением последующего извлечения ЖРО.

2. НАЗНАЧЕНИЕ И ОБЛАСТЬ ПРИМЕНЕНИЯ

2.1. Настоящий документ устанавливает требования к обеспечению безопасности при сборе, переработке, хранении и кондиционировании ЖРО на ядерных установках, радиационных источниках, в пунктах хранения ядерных материалов и радиоактивных веществ, хранилищах РАО (далее - пункты хранения).

2.2. Настоящий документ распространяется на проектируемые, сооружаемые, эксплуатируемые и выводимые из эксплуатации ядерные установки, радиационные источники и пункты хранения при сборе, переработке, хранении и кондиционировании ЖРО.

2.3. Настоящий документ не распространяется на:

- обращение с ЖРО, образующимися при добыче и обогащении руд радиоактивных веществ и других полезных ископаемых;
- обращение с ЖРО, накопленными в поверхностных водоемах объектов ядерного топливного цикла.

3. ОБЩИЕ ТРЕБОВАНИЯ К ОБЕСПЕЧЕНИЮ БЕЗОПАСНОСТИ ПРИ СБОРЕ, ПЕРЕРАБОТКЕ, КОНДИЦИОНИРОВАНИИ И ХРАНЕНИИ ЖИДКИХ РАДИОАКТИВНЫХ ОТХОДОВ

3.1. Технические средства и организационные меры по обеспечению радиационной безопасности при сборе, переработке, хранении и кондиционировании ЖРО на ядерной установке, радиационном источнике и в пункте хранения должны определяться исходя из максимальной допустимой активности ЖРО на этих объектах и ограничивать радиационное воздействие на работников (персонал), население и окружающую среду уровнями, установленными Нормами радиационной безопасности (НРБ-99) и другими федеральными нормами и правилами в области использования атомной энергии и нормативными документами.

3.2. В проекте ядерной установки, радиационного источника и пункта хранения должны быть предусмотрены конкретные технические средства безопасного сбора, переработки, хранения и кондиционирования ЖРО, разработанные в соответствии с требованиями настоящего документа, других федеральных норм и правил в области использования атомной энергии и нормативных документов.

При отсутствии необходимых нормативных документов предлагаемые конкретные технические решения устанавливаются и обосновываются в проекте ядерной установки, радиационного источника и пункта хранения в соответствии с достигнутым уровнем науки и техники.

3.3. Требования к конструированию, изготовлению и монтажу оборудования, предназначенного для сбора, переработки, хранения и кондиционирования ЖРО, проектированию соответствующих систем (элементов) ядерных установок, радиационных источников и пунктов хранения, а также классификация систем (элементов) и оборудования, предназначенных для сбора, переработки, хранения и кондиционирования ЖРО, по назначению, влиянию на безопасность и по характеру выполняемых ими функций безопасности, категориям сейсмостойкости, пожаро- и взрывоопасности устанавливаются федеральными нормами и правилами в области использования атомной энергии, регламентирующими обеспечение безопасности ядерных установок, радиационных источников и пунктов хранения и настоящим документом.

3.4 Устройство и надежность систем (элементов) ядерной установки, радиационного источника и пункта хранения, документация и работы по сбору, переработке, хранению и кондиционированию ЖРО должны являться объектами деятельности эксплуатирующих организаций и (или) организаций, выполняющих работы и предоставляющих услуги эксплуатирующим организациям, по обеспечению качества в соответствии с программой обеспечения качества эксплуатирующей организации, требованиями федеральных норм и правил в области использования атомной энергии и других нормативных документов.

3.4.1. Программа обеспечения качества должна быть направлена на:

- организацию эффективной системы подготовки, переподготовки, повышения квалификации и аттестации работников (персонала);
- минимизацию образующихся ЖРО по величине их активности, массе и объему;
- контроль качества поставляемого оборудования, комплектующих изделий и материалов;
- получение достоверной и полной информации о количественном и качественном составе ЖРО в местах их образования, сбора, переработки, хранения и кондиционирования;

- организацию контроля качества проведения технологических процессов при сборе, переработке, хранении и кондиционировании ЖРО;
- установление системы критериев качества ЖРО, которым ЖРО должны отвечать после сбора, переработки, хранения и кондиционирования;
- использование метрологически аттестованных методик контроля качества ЖРО и испытаний упаковок кондиционированных отходов;
- организацию контроля качества ЖРО и упаковок кондиционированных отходов;
- организацию эффективной системы записей и хранения документации при сборе, переработке, хранении и кондиционировании ЖРО, включая идентификационную маркировку упаковок кондиционированных отходов.

3.4.2. В зависимости от стадии обращения с ЖРО при установлении критериев качества ЖРО должны учитываться основные характеристики ЖРО, контейнера и упаковки ЖРО.

3.4.2.1. Характеристики ЖРО:

- химический состав и фазовое состояние;
- величина суммарной активности;
- радионуклидный состав, величина удельной альфа- и бета-активности.

3.4.2.2. Характеристики отвержденных ЖРО:

3.4.2.2.1. Битумный компаунд:

- радионуклидный состав, величина удельной альфа- и бета-активности, мощность эквивалентной дозы;
  - содержание свободной воды в компаунде;
  - водоустойчивость;
  - термическая устойчивость;
  - радиационная устойчивость;
  - биологическая устойчивость.

3.4.2.2.2. Цементный компаунд:

- радионуклидный состав, величина удельной альфа- и бета-активности, мощность эквивалентной дозы;
  - водоустойчивость;
  - механическая прочность;
  - радиационная устойчивость;
  - термическая устойчивость.

3.4.2.2.3. Стеклоподобные материалы:

- радионуклидный состав, величина удельной альфа- и бета-активности, мощность эквивалентной дозы;
  - состав фосфатных материалов;
  - однородность отвержденного материала;
  - тепловыделение;
  - водоустойчивость;
  - термическая стойкость;
  - радиационная стойкость;
  - механическая прочность;
  - теплофизические константы (теплопроводность, коэффициент термического расширения).

3.4.2.4. Характеристики контейнера ЖРО:

- коррозионная стойкость, радиационная стойкость, конфигурация (геометрические размеры) - для металлического контейнера;
- плотность, пористость, водопроницаемость; газопроницаемость, морозостойкость, радиационная стойкость, стойкость к микроорганизмам, плесени и грибкам, пожароустойчивость, конфигурация (геометрические размеры) - для железобетонного контейнера;
- иные характеристики, определяющие изолирующую способность контейнера.

3.4.2.5. Характеристики упаковки ЖРО:

- радионуклидный состав, величина удельной альфа- и бета-активности, мощность эквивалентной дозы;
- величина суммарной активности;
- однородность;
- механическая прочность (статические, динамические, ударные нагрузки);
- устойчивость к тепловым нагрузками и термическим циклам;
- радиационная устойчивость.

3.4.3. Система контроля качества ЖРО и кондиционированных отходов должна включать контроль качества:

- процесса сбора ЖРО;
- ЖРО, направляемых на переработку;
- процесса переработки ЖРО;
- матричных материалов;
- процесса отверждения ЖРО;
- отвержденных ЖРО;
- упаковок кондиционированных отходов.

Объем контроля качества устанавливается в проекте ядерной установки, радиационного источника и пункта хранения и должен обеспечивать получение достоверной информации о характеристиках ЖРО, матричных материалов, отвержденных ЖРО и упаковках кондиционированных отходов.

3.4.4. В программе обеспечения качества должны быть установлены порядок и процедуры регистрации нарушений критериев качества ЖРО и кондиционированных отходов, а также организации сбора, обработки и анализа данных о нарушениях и причинах их возникновения.

По результатам анализа причин нарушений должны разрабатываться и приниматься корректирующие меры по предотвращению их повторения.

3.4.5. Эксплуатирующая организация должна контролировать эффективность реализации программы обеспечения качества на ядерной установке, радиационном источнике и в пункте хранения путем проведения проверок (инспекций), включающих:

- верификацию ведения технологических процессов при сборе, переработке, кондиционировании и хранении ЖРО в рамках установленных проектом параметров в соответствии с требованиями федеральных норм и правил в области использования атомной энергии и условиями действия лицензии государственного органа регулирования безопасности при использовании атомной энергии,
- проверку работоспособности систем управления технологическими процессами и их контроля;

- проверку соответствия качества ЖРО и упаковок кондиционированных отходов критериям качества.

По всем выявляемым при проверках (инспекциях) случаям :оответствия должны быть приняты корректирующие меры.

3.5. При сборе, переработке, хранении и кондиционировании ЖРО іжно обеспечиваться:

- поддержание требуемого уровня безопасности при обращении с ЖРО как с источниками ионизирующего излучения;
- исключение необоснованного облучения работников (персонала);
- сведение к разумно достижимому низкому уровню облучения работников (персонала) и населения с учетом санитарных правил, норм и гигиенических нормативов, экономических и социальных факторов;
- предотвращение возможных аварий с радиационными последствиями и ослабление их последствий в случае их возникновения;
- сокращение объема ЖРО;
- подготовка ЖРО к хранению и (или) захоронению после их кондиционирования.

3.6. Проектом ядерной установки, радиационного источника и пункта інения должна быть установлена классификация помещений, едназначенных для сбора, переработки, хранения и кондиционирования ?О, по взрыво- и пожарной безопасности в соответствии с требованиями :деральных норм и правил в области использования атомной энергии.

Конкретные технические решения и организационные меры по еспечению взрывозащиты и противопожарной защиты при сборе, ·реработке, хранении и кондиционировании ЖРО устанавливаются и юсновываются в проекте ядерной установки, радиационного источника и ·нкта хранения.

3.7. Помещения, предназначенные для сбора, переработки, хранения и ндиционирования ЖРО, должны быть оборудованы системой вентиляции, ·едотвращающей загрязнение воздушной среды помещений и окружающей ·еды радиоактивными веществами и поддерживающей климатические ·ловия, необходимые для нормальной эксплуатации оборудования. ;аляемые из помещений загрязненный воздух и из оборудования газы перед ібросом в атмосферу должны подвергаться очистке.

3.8. При сборе, переработке, хранении и кондиционировании ЖРО олжны быть предусмотрены:

- технические средства и организационные меры по обеспечению физической защиты ЖРО;
- технические средства и организационные меры по предотвращению протечек ЖРО и иных процессов, приводящих к поступлению радионуклидов в окружающую среду в количестве, превышающем пределы, установленные санитарными правилами, нормами и гигиеническими нормативами, федеральными нормами и правилами в области использования атомной энергии;
- радиационный контроль, включающий: контроль загрязненности поверхностей помещений, оборудования и трубопроводов, мощности эквивалентной дозы, удельной активности и радионуклидного состава ЖРО.

Средства и объем радиационного контроля устанавливаются в проекте ядерной установки, радиационного источника и пункта хранения в соответствии с требованиями санитарных правил, норм и гигиенических нормативов, федеральных норм и правил в области использования атомной энергии.

3.9. При сборе, переработке, хранении и кондиционировании ЖРО должна быть исключена возможность:

- неконтролируемого изменения агрегатного состояния ЖРО, в том числе образование осадков и отложений;
- неконтролируемого возникновения экзотермических реакций;
- неконтролируемого образования коррозионно агрессивных веществ.

3.10. При сборе, переработке, хранении и кондиционировании ЖРО, содержащих ядерно-опасные делящиеся материалы, должно быть исключена возможность возникновения СЦР.

Конструкция и геометрические размеры оборудования, предназначенного для сбора, переработки, хранения и кондиционирования ЖРО, содержащих ядерно-опасные делящиеся материалы, а также порядок проведения работ не должны приводить к возникновению СЦР.

Содержание ядерно-опасных делящихся материалов в кондиционированных ЖРО и геометрическая форма их упаковок должны исключать возможность возникновения СЦР.

Помещения, в которых находится оборудование, предназначенное для сбора, переработки, хранения и кондиционирования ЖРО, содержащих ядерно-опасные делящиеся материалы, должны быть оснащены системой аварийной сигнализации САС, которая должна эксплуатироваться в режиме постоянной готовности обнаружения СЦР.

Обеспечение ядерной безопасности при сборе, переработке, хранении и кондиционировании ЖРО, содержащих ядерно-опасные делящиеся материалы, регламентируется федеральными нормами и правилами в области использования атомной энергии, определяющими правила ядерной безопасности.

3.11. При сборе, переработке, хранении и кондиционировании ЖРО должна быть предусмотрена возможность дезактивации оборудования, трубопроводов, контейнеров и помещений. Оборудование, трубопроводы и поверхности помещений, предназначенные для сбора, переработки, хранения и кондиционирования ЖРО, должны обладать коррозионной стойкостью в агрессивных средах, низкой сорбирующей способностью по отношению к радиоактивным веществам и легко дезактивироваться.

3.12. Сбор, переработка, хранение и кондиционирование ЖРО совместно с нерадиоактивными отходами не разрешается.

3.13. Сбор, переработка, хранение и кондиционирование ЖРО должны документироваться в соответствии с программой обеспечения качества. Каждая партия (упаковка) ЖРО на всех этапах обращения должна сопровождаться документацией, содержащей ее основные характеристики, в том числе:

3.13.1. Сбор ЖРО:

- источник образования;
- количество;
- химический состав и фазовое состояние;
- величина суммарной активности;

ПЕРЕЧЕНЬ СОКРАЩЕНИЙ

ЖРО - жидкие радиоактивные отходы

РАО - радиоактивные отходы

САС - система аварийной сигнализации

СЦР - самоподдерживающаяся цепная реакция деления

ХЖО - хранилище жидких радиоактивных отходов

1. ОСНОВНЫЕ ТЕРМИНЫ И ОПРЕДЕЛЕНИЯ

1. Битумирование ЖРО - включение РАО в битумный матричный материал.

2. Водоустойчивость - способность компаунда (упаковки) сохранять свои свойства и удерживать включенные в него радионуклиды при контакте с водой.

3. Выдержка ЖРО - хранение ЖРО с целью снижения радиоактивности и тепловыделения за счет распада короткоживущих радионуклидов.

4. Достигнутый уровень науки и техники - комплекс научных и технических знаний, техно-логических, проектных и конструкторских разработок в определенной области науки и техники, который подтвержден научными исследованиями и практическим опытом и отражен в научно-технических материалах.

5. Компаунд - матричный материал с включенным в него РАО.

6. Кондиционирование ЖРО – операции по изготовлению упаковок отходов, пригодных для безопасного хранения и (или) транспортирования и (или) захоронения. Кондиционирование может включать перевод ЖРО в стабильную форму, помещение ЖРО в контейнеры.

7. Контейнер для РАО - емкость, используемая для сбора, и (или) транспортирования, и (или) хранения, и (или) захоронения РАО.

8. Корректирующие меры - деятельность, с помощью которой устраняются несоответствия и предотвращается их повторение.

9. Материал матричный - нерадиоактивный материал, используемый для иммобилизации ЖРО в монолитную структуру.

П р и м е ч а н и е. Примерами матричных материалов являются битум, цемент, стеклоподобные материалы.

10. Обращение с ЖРО - все виды деятельности, связанные со сбором, транспортированием, переработкой, кондиционированием, хранением и (или) захоронением ЖРО.

11. Обеспечение качества при обращении с РАО - планируемая и систематически осущест-вляемая деятельность, направленная на то, чтобы все работы по обращению с РАО, влияющие на ядерную и радиационную безопасность, проводились в соответствии с требованиями федеральных норм и правил в области использования атомной энергии и других нормативных документов, а их результаты удовлетворяли предъявленным требованиям.

12. Остекловывание ЖРО - перевод РАО в стеклоподобное состояние.

13. Отверждение ЖРО - перевод ЖРО в твердое агрегатное состояние с целью уменьшения возможности миграции радионуклидов в окружающую среду.

14. Отходы жидкие радиоактивные - РАО в виде жидких продуктов (водных или органических) или пульп, содержащих радионуклиды в растворенной форме или в виде взвесей.

15. Отходы жидкие радиоактивные органические - ЖРО в виде масел, эмульсий масел в воде, растворов детергентов, экстрагентов и т.п.

16. Отходы радиоактивные - не подлежащие дальнейшему использованию вещества в любом агрегатном состоянии, материалы, изделия, приборы, оборудование, объекты биологического происхождения, в которых содержание радионуклидов превышает уровни, установленные федеральными

- радионуклидный состав, величина удельной альфа- и бета-активности, дата их определения;
- тип контейнера (для упаковки ЖРО);
- дата упаковки (для упаковки ЖРО);
- мощность эквивалентной дозы (для упаковки ЖРО);
- поверхностное загрязнение контейнера (для упаковки ЖРО);
- идентификационный знак упаковки (для упаковки ЖРО);
- место хранения;
- соответствие критериям качества.

3.13.2. Переработка ЖРО:
- источник образования;
- методы переработки;
- количество;
- химический состав и фазовое состояние;
- величина суммарной активности;
- радионуклидный состав, величина удельной альфа- и бета-активности, дата их определения;
- тип контейнера (для упаковки ЖРО)
- дата упаковки (для упаковки ЖРО);
- мощность эквивалентной дозы (для упаковки ЖРО);
- поверхностное загрязнение контейнера (для упаковки ЖРО);
- идентификационный знак упаковки (для упаковки ЖРО);
- место хранения.

3.13.3. Отверждение ЖРО:
- источник образования;
- методы переработки;
- количество;
- величина суммарной активности;
- радионуклидный состав, величина удельной альфа- и бета-активности, дата их определения;
- тип контейнера;
- дата упаковки;
- мощность эквивалентной дозы от упаковки ЖРО;
- поверхностное загрязнение контейнера;
- идентификационный знак упаковки;
- место хранения.

3.13.4. Кондиционирование ЖРО:
- источник образования;
- количество;
- методы переработки;
- метод кондиционирования;
- величина суммарной активности;
- радионуклидный состав, величина удельной альфа- и бета-активности, дата их определения;
- тип и номер контейнера;
- дата упаковки;
- поверхностное загрязнение контейнера, мощность эквивалентной дозы от упаковки и дата их определения;
- идентификационный знак упаковки;
- место хранения.

4. ТРЕБОВАНИЯ К ОБЕСПЕЧЕНИЮ БЕЗОПАСНОСТИ ПРИ СБОРЕ ЖИДКИХ РАДИОАКТИВНЫХ ОТХОДОВ

4.1. Сбор ЖРО должен являться обязательным этапом подготовки их к переработке, хранению и кондиционированию и обеспечивать исключение поступления радионуклидов в окружающую среду выше пределов, установленных санитарными правилами, нормами и гигиеническими нормативами, федеральными нормами и правилами в области использования атомной энергии, путем сосредоточения ЖРО в специальном оборудовании.

4.2. Сбор ЖРО должен проводиться раздельно в зависимости от:

- периода полураспада радионуклидов (менее 15 суток, более 15 суток);
- величины удельной активности;
- концентрации альфа-активных радионуклидов;
- химического состава;
- фазового состояния;
- предполагаемого способа переработки.

4.3. Органические взрыво- и пожароопасные ЖРО должны собираться отдельно от других видов ЖРО.

4.4. При сборе неорганических ЖРО должны собираться раздельно:

- малосолевые водные растворы (с концентрацией солей менее 1 г/л);
- высокосолевые водные растворы (с концентрацией солей более 1 г/л);
- щелочные металлы, использованные в качестве теплоносителя;
- сильные окислители;
- коррозионно-активные вещества;
- химически неустойчивые вещества;
- ионообменные смолы;
- перлит, вермикулит и др.;
- титановые сорбенты;
- шламы.

4.5. Сбор ЖРО должен производиться с одновременным учетом требований пп. 4.1–4.4 в последовательности, обеспечивающей минимально возможное облучение работников (персонала). Последовательность операций по сбору ЖРО устанавливается и обосновывается в проекте ядерной установки, радиационного источника и пункта хранения.

4.6. Сборники (емкости, контейнеры и т.д.) ЖРО должны располагаться как можно ближе к месту образований отходов.

4.7. ЖРО, содержащие только радионуклиды с периодом полураспада менее 15 суток, должны собираться отдельно и подлежат выдержке в местах временного хранения ЖРО до снижения величины их удельной активности и величины их суммарной активности до значений, при которых радиоактивные вещества, освобождаются от регламентации Нормами радиационной безопасности (НРБ-99).

4.8. Для сбора ЖРО должна быть предусмотрена система специальной канализации (спецканализация). Если количество образующихся ЖРО не превышает 200 л/сут, для их сбора могут использоваться контейнеры (сборники). Требования к контейнерам (сборникам) устанавливаются нормативными документами.

4.9. Сброс ЖРО в хозяйственно-фекальную канализацию, производственно-ливневую канализацию, в поверхностные водоемы, поглощающие ямы, колодцы, скважины, на поля орошения, поля фильтрации и на поверхность земли запрещается.

5. ТРЕБОВАНИЯ К ОБЕСПЕЧЕНИЮ БЕЗОПАСНОСТИ ПРИ ПЕРЕРАБОТКЕ ЖИДКИХ РАДИОАКТИВНЫХ ОТХОДОВ

5.1. Переработка ЖРО должна обеспечивать очистку жидкой фазы ЖРО и концентрирование радионуклидов в меньшем объеме.

Не допускается полное обезвоживание высокосолевых водных растворов ЖРО в случае возможного экзотермического взаимодействия компонентов сухого остатка ЖРО.

Конкретные технические методы и средства переработки ЖРО устанавливаются и обосновываются в проекте ядерной установки, радиационного источника и пункта хранения.

5.2. При передаче (транспортировании) солевых концентратов (кубовых остатков) ЖРО к месту их хранения и отверждения должны быть приняты меры по предотвращению образования отложений в трубопроводах и оборудовании.

5.3. Образующиеся в результате переработки ЖРО солевые концентраты, отработавшие сорбенты, шламы, осадки должны быть кондиционированы в соответствии с требованиями настоящего документа.

5.4. Если концентрация радионуклидов и вредных веществ в образующихся в результате переработки ЖРО очищенных водах не превышает допустимых концентраций, установленных в соответствии с требованиями санитарных правил, норм и гигиенических нормативов, федеральных норм и правил в области использования атомной энергии, то они могут быть использованы для собственных нужд в системе оборотного водоснабжения ядерной установки, радиационного источника и пункта хранения или сбрасываться в открытую гидросеть через промежуточную контрольную емкость.

6. ТРЕБОВАНИЯ К ОБЕСПЕЧЕНИЮ БЕЗОПАСНОСТИ ПРИ ХРАНЕНИИ ЖИДКИХ РАДИОАКТИВНЫХ ОТХОДОВ

6.1. При хранении ЖРО должно обеспечиваться исключение:

- необоснованного облучения работников (персонала);
- облучения населения выше установленных пределов;
- поступления радионуклидов в окружающую среду выше пределов, установленных федеральными нормами и правилами в области использования атомной энергии и другими нормативными документами.

6.2. В проекте ядерной установки, радиационного источника и пункта хранения должны быть предусмотрены технические средства и организационные меры по безопасному хранению ЖРО, а также установлены и обоснованы допустимые объемы ЖРО, их радионуклидный состав, величина активности и сроки хранения ЖРО.

6.3. Хранение больших объемов ЖРО должно осуществляться в специально оборудованных хранилищах с системой барьеров, ...отвращающей поступление радионуклидов в окружающую среду выше пределов, установленных федеральными нормами и правилами в области использования атомной энергии и другими нормативными документами. Технические барьеры устанавливаются и обосновываются в проекте ядерной установки, радиационного источника и пункта хранения в соответствии с требованиями настоящего документа и других федеральных норм и правил в области использования атомной энергии.

6.3.1. Конструкция и конструкционные материалы ХЖО должны:

- предотвращать выход радионуклидов в окружающую среду выше пределов, установленных федеральными нормами и правилами в области использования атомной энергии;
- обеспечивать срок службы ХЖО не менее срока эксплуатации ядерной установки, радиационного источника и пункта хранения, на котором оно размещено.

Объем емкостей ХЖО должен обеспечивать необходимую технологическую выдержку ЖРО до их переработки и (или) распада короткоживущих радионуклидов.

6.3.2. Емкости для хранения ЖРО должны быть оснащены:

- трубопроводами и арматурой для приема ЖРО, направления их на кондиционирование, полного опорожнения;
- средствами контроля технологических параметров (температуры, давления, уровня в емкости), включая системы сигнализации о превышении верхнего уровня в емкости и контроля протечек ЖРО из емкости;
- радиационным контролем;
- пробоотборными устройствами, позволяющими производить отбор проб по всему объему емкости;
- устройствами для определения толщины (высоты) осадка;
- устройствами для диспергирования и удаления шлама (осадка) и отложений;
- оборудованием и трубопроводами для передачи растворов, шламов, сорбентов и смол из одной емкости в другую;
- трубопроводом перелива, объединенным с резервной емкостью, с диаметром большим, чем у приемного трубопровода;
- технологической сдувкой под разрежением, связанной с системой технологических сдувок и предотвращающей образование повышенного давления в свободном объеме емкости;
- средствами контроля водорода, предупредительной и аварийной сигнализацией, автоматическими средствами пожароизвещения и при необходимости пожаротушения;
- устройствами, не допускающими повреждение емкости из-за повышения в них давления или их вакуумирования.

6.3.3. В емкостях для хранения ЖРО высокого уровня активности должны быть дополнительно предусмотрены технические методы и средства для предотвращения:

- разогрева и выпаривания ЖРО;
- накопления взрывоопасных газообразных веществ.

6.3.4. Конструкция емкости для хранения ЖРО должна позволять поиск мест протечек из емкости и выполнение ее ремонта.

6.3.5. Передача ЖРО из одной емкости для хранения ЖРО в другую должна осуществляться с использованием статического давления жидкости или газа (без применения насосов).

6.3.6. Помещения, предназначенные для размещения емкостей для хранения ЖРО, должны иметь не менее чем трехслойную гидроизоляцию и облицовку из нержавеющей стали. Объем облицованного помещения должен вмещать все количество ЖРО, находящихся в емкостях.

6.3.7. На территории вокруг помещений с емкостями для хранения ЖРО должны быть предусмотрены контрольно-наблюдательные скважины для отбора проб грунтовых вод. Количество и расположение наблюдательных скважин устанавливается в соответствии с нормативными документами.

6.3.8. В помещениях, в которых находятся емкости для хранения ЖРО, должны быть предусмотрены:

- сигнализация протечек из емкостей;
- система сбора и возврата протечек;
- вентиляция;
- радиационный контроль;
- средства для дезактивации.

6.3.9. Водно-химический режим в емкостях для хранения ЖРО должен исключать интенсивные коррозионные процессы.

6.3.10. Помещения, в которых находятся емкости с органическими ЖРО, должны быть снабжены устройствами пожарной сигнализации и средствами пожаротушения. Совместное хранение в помещениях органических ЖРО со средами, содержащими окислители, не допускается.

6.3.11. Проектом ядерной установки, радиационного источника и пункта хранения должны быть предусмотрены резервные емкости для хранения ЖРО, образовавшихся в результате аварий. Минимальный резервный объем емкостей для хранения ЖРО должен быть обоснован в проекте. На резервные емкости для хранения ЖРО и помещения, в которых они находятся, распространяются те же требования, что и на основные емкости для хранения ЖРО.

6.4. Хранение малых объемов ЖРО должно осуществляться в специально оборудованных помещениях. Расположение помещений, оборудование помещений для хранения малых объемов ЖРО и условия их хранения должны соответствовать требованиям Основных санитарных правил обеспечения радиационной безопасности (ОСПОРБ –99).

7. ТРЕБОВАНИЯ К ОБЕСПЕЧЕНИЮ БЕЗОПАСНОСТИ ПРИ ОТВЕРЖДЕНИИ ЖИДКИХ РАДИОАКТИВНЫХ ОТХОДОВ

7.1. Технологический процесс отверждения ЖРО должен обеспечивать получение продуктов с показателями качества, установленными в настоящем документе. Конкретные технические методы и средства отверждения ЖРО устанавливаются и обосновываются в проекте ядерной установки, радиационного источника и пункта хранения.

7.2. Отверждение ЖРО должно производиться методами цементирования, битумирования и остекловывания.

При выборе метода отверждения ЖРО должны учитываться:

- физические и химические характеристики ЖРО;
- свойства матричного материала;

- предполагаемый способ хранения и (или) захоронения кондиционированных отходов.

Допускается использование других методов отверждения ЖРО, разработанных в соответствии с достигнутым уровнем науки и техники.

7.3. Процесс отверждения ЖРО должен быть пожаро- и взрывобезопасным и не сопровождаться образованием значительного количества вторичных РАО.

7.4. При отверждении ЖРО методом цементирования должны выполняться следующие основные требования:

7.4.1. Установка цементирования должна находиться в отдельном помещении, снабженном системой вентиляции.

7.4.2. Используемые неорганические вяжущие (цемент, портландцемент, шлакопортландцемент и др.) должны обеспечивать качество цементной матрицы в соответствии с требованиями настоящего документа.

7.4.3. В цементную матрицу не могут включаться ЖРО, содержащие вещества, взаимодействующие с цементом с образованием токсичных веществ (например, соли аммония).

7.4.4. С целью предотвращения разлива в помещении цементного компаунда при его расфасовке в контейнеры должны быть предусмотрены:

- контроль размещения контейнера для цементного компаунда под сливным патрубком;
- контроль заполнения емкости цементным компаундом;
- устройство, исключающее возможность разлива во время транспортирования контейнера с цементным компаундом от места заполнения до места выдержки для отверждения.

7.4.5. Оборудование для перемешивания цементного теста с ЖРО должно обеспечивать получение гомогенного цементного компаунда с равномерным распределением радионуклидов по его объему.

7.4.6. При цементировании должно быть обеспечено управление технологическими параметрами процесса и контроль за ними, обеспечивающими получение цементного компаунда со следующими основными показателями качества:

Показатель качества	Допустимые значения
Удельная активность компаунда:	
бета-активность	$<3{,}7 \cdot 10^{10}$ Бк/кг ($1 \cdot 10^{-3}$ Ки/кг)
альфа-активность	$<3{,}7 \cdot 10^{7}$ Бк/кг ($1 \cdot 10^{-6}$ Ки/кг)
Водоустойчивость (скорость выщелачива-ния радионуклидов по Cs-137 и Sr-90)	$<1 \cdot 10^{-3}$ г/см^{2}сут
Механическая прочность (предел проч-ности при сжатии)	≥ 50 кгс/см^2
Радиационная устойчивость	Механическая прочность не менее 50 кгс/см^2 после облучения дозой 10^6 Гр (10^8 рад)
Устойчивость к термическим циклам	Механическая прочность не менее 50 кгс/см^2 после 30 циклов замораживания и оттаивания (-40 ...

	+40 °C)
Водостойкость	Механическая прочность не менее 50 $кгс/см^2$ после 90-дневного погружения в воду

Требования к методам контроля качества цементного компаунда станавливаются нормативными документами.

7.5. При отверждении ЖРО методом битумирования должны ыполняться следующие основные требования:

7.5.1. Установка битумирования должна находиться в отдельном омещении, снабженном системой вентиляции, пожарной сигнализацией и редствами пожаротушения.

7.5.2. Используемый в качестве матричного материала битум должен довлетворять следующим основным требованиям:

- температура вспышки не ниже 200°C;
- температура воспламенения не ниже 250°C;
- температура самовоспламенения не ниже 400°C.

7.5.3. В битумную матрицу не должны включаться ЖРО, компоненты оторых вступают с ней в химическое взаимодействие, сопровождающееся:

- экзотермическими эффектами;
- образованием токсичных или взрывоопасных веществ;
- ухудшением качества образующегося компаунда.

7.5.4. Солевые концентраты, направляемые на битумирование, должны довлетворять следующим требованиям:

- концентрация сильных окислителей (нитраты трехвалентных металлов, марганцевокислый калий и т.п.) в ЖРО не должна превышать 5% от массы сухого остатка;
- содержание нитрата аммония в ЖРО не должно превышать 12% от массы сухого остатка;
- величина pH ЖРО должна находиться в пределах 6,5-11,5;
- удельная активность ЖРО не должна превышать $3{,}7 \cdot 10^{10}$ $Бк/дм^3$ (1 $Ки/дм^3$)

7.5.5. ЖРО не должны содержать органических веществ, которые в условиях проведения процесса битумирования могут образовать легколетучие соединения в количествах, способных создать взрывоопасную концентрацию в газовой фазе. Должен быть обеспечен контроль за содержанием таких соединений в отходящих газах.

7.5.6. С целью предотвращения разлива в помещении битумного компаунда при его расфасовке должны быть предусмотрены:

- контроль размещения контейнера для битумного компаунда под сливным патрубком;
- контроль заполнения емкости битумным компаундом;
- устройство, исключающее возможность разлива во время транспортирования контейнера с битумным компаундом от места заполнения до места выдержки для остывания.

7.5.7. Параметры процесса битумирования должны обеспечивать получение гомогенного битум-ного компаунда с равномерным распределением по его объему радионуклидов.

7.5.8. При битумировании должно быть обеспечено управление технологическими параметрами процесса и контроль за ними, обеспечивающими получение битумного компаунда со следующими основными показателями качества:

Показатель качества	Допустимые значения
Удельная активность компаунда:	
бета- активность	$<3,7\cdot10^{10}$ Бк/кг ($1\cdot10^{-3}$ Ки/г)
альфа-активность	$<3,7\cdot10^{7}$ Бк/кг ($1\cdot10^{-6}$ Ки/г)
Водоустойчивость (скорость выщелачи-вания радионуклидов по Cs-137 и Sr-90)	$<1\cdot10^{-3}$ г/см^2·сут
Содержание свободной влаги в компаунде	< 3 %
Термическая устойчивость	t° вспышки >200 °C t° воспламенения >250 °C; t° самовоспламенения >400 °C
Радиационная устойчивость	Увеличение объема менее 10% после облучения дозой 10^6 Гр (10^3 рад)
Биологическая устойчивость	Отсутствие роста грибков

Требования к методам контроля качества битумного компаунда устанавливаются нормативными документами.

7.6. При отверждении ЖРО методом остекловывания должны выполняться следующие требования:

7.6.1. Установка остекловывания должна находиться в отдельном помещении, снабженном системой вентиляции.

7.6.2. С целью предотвращения разлива стеклоподобного материала при его расфасовке должны быть предусмотрены:

- контроль размещения контейнера для стеклоподобного материала под сливным патрубком;
- контроль заполнения емкости стеклоподобным материалом;
- устройство, исключающее возможность разлива во время транспортирования контейнера со стеклоподобным материалом от места его заполнения до места выдержки для остывания.

7.6.3. Концентрация плутония в ЖРО не должна превышать 0,03 г/ дм^3.

7.6.4. При остекловывании должен быть обеспечен контроль концентраций радионуклидов и концентраций H_2, CO и других газов, отходящих из печи.

7.6.5. Химический состав ЖРО, используемые материалы и параметры процесса остекловывания должны обеспечивать получение гомогенного стеклоподобного материала с равномерным распределением по его объему радионуклидов.

7.6.6. При остекловывании должно быть обеспечено управление технологическими параметрами процесса и контроль за ними, обеспечивающими получение стеклоподобного материала со следующими основными показателями качества:

Показатель качества	Допустимые значения
Состав отвержденных ЖРО	<24 – 27% мас. Na_2O и оксидов одновалентных нуклидов; <20 - 24% мас. Al_2O_3 и оксидов многовалентных нуклидов, в том числе < 0,2% мас. трансурановых элементов; <50 –52% мас. P_2O_5
Однородность	Равномерность состава блока по макрокомпонентам в пределах ± 10%, отсутствие выделения дисперсных фаз, особенно для альфа-излучателей. Количество альфа-излучателей < 0,2% мас.
Тепловыделение	< 5 кВт /м^3
Водоустойчивость (скорость выщела-Чиваемости радиону-Клидов по Cs^{137}, Sr^{90}, Pu.)	10^{-5} - 10^{-6} г/см^2·сут Cs^{137}; 10^{-6} г/см^2·сут Sr^{90}; 10^{-7} г/см^2·сут Pu
Термическая стойкость	Отсутствие изменений структуры и водостойкости в результате хранения при температуре до 450^{0}C
Радиационная Стойкость	Неизменность структуры и водоустойчивости при значениях: а) дозы ~ 10^8 Гр (10^{10} рад) (по β, γ-излучению), б)10^{18} - 10^{19} α-распадов/см^3
Механическая прочность:	
Прочность на сжатие	(0,9 -1,3) кгс/мм^2 (0,9 -1,3) ·10^7 Н/м^2
Прочность на изгиб	(4,1 - 4,7) кгс/мм^2 (4,1 - 4,7) ·10^7 Н/м^2
Модуль Юнга	> 5400 кгс/мм^2 (> 5,4·10^{10} Н/м^2)
Теплофизические Константы:	
Коэффициент термического Расширения	(8-15) ·10^{-6} 1/^{0}C

Коэффициент теплопроводности	Изменения в пределах 0,7-1,6 Вт/мК в интервале температур 20-500 °C
Газовыделение	Не допустимо

7.6.6. Требования к методам контроля качества стекломассы устанавливаются нормативными документами.

8. ТРЕБОВАНИЯ К ОБЕСПЕЧЕНИЮ БЕЗОПАСНОСТИ ПРИ КОНДИЦИОНИРОВАНИИ ЖИДКИХ РАДИОАКТИВНЫХ ОТХОДОВ

8.1. Кондиционирование ЖРО должно обеспечивать перевод ЖРО в формы, пригодные для последующего транспортирования, и (или) хранения, и (или) захоронения.

8.2. В зависимости от характеристик ЖРО и способов последующего обращения с кондиционированными ЖРО, в том числе их транспортирование, и (или) переработка, и (или) хранение, и (или) захоронение, кондиционирование ЖРО должно включать в себя одну из следующих операций или их совокупность:

- размещение ЖРО в контейнере;
- отверждение ЖРО и размещение отвержденных ЖРО в контейнере;
- размещение упаковки ЖРО в дополнительном контейнере.

8.3. В проекте ядерной установки, радиационного источника и пункта хранения должны быть обоснованы принятые в соответствии требованиями настоящего документа, других федеральных норм и правил в области использования атомной энергии методы и средства кондиционирования ЖРО.

8.4. При установлении методов и средств кондиционирования ЖРО должны учитываться:

- характеристики ЖРО, подлежащих кондиционированию;
- способы последующего обращения с кондиционированными ЖРО, в том числе их переработка, и (или) транспортирование, и (или) хранение, и (или) захоронение;
- установленные для последующего обращения с ЖРО критерии качества.

8.5. Радионуклидный состав, удельная активность радионуклидов, суммарная величина активности в упаковке РАО, мощность эквивалентной дозы на поверхности контейнера, величина радиоактивного загрязнения наружной поверхности контейнера должны соответствовать критериям качества ЖРО для последующего этапа обращения с ними. Упаковка РАО должна предотвращать неприемлемое распространение радионуклидов в окружающую среду.

8.6. Упаковка кондиционированных РАО не должна содержать:

- сильных окислителей и химически неустойчивых веществ;
- коррозионно-активных веществ;
- ядовитых, патогенных и инфекционных веществ;

- биологически активных веществ;
- легковоспламеняющихся и взрыво- и пожароопасных веществ;
- веществ, способных к детонации или взрывному разложению;
- веществ, вступающих в экзотермическое взаимодействие с водой, сопровождающееся взрывом;
- веществ, содержащих или способных генерировать токсичные газы, пары или возгоны.

Содержание жидкости в упаковке РАО не должно превышать 3%.

8.7. Выбор конструкции контейнера и конструкционных материалов контейнера должен быть основан на:

- физических и химических характеристиках РАО;
- способах последующего обращения с упаковкой РАО;
- установленных для последующего обращения с РАО критериев качества.

8.8. Конструкция контейнера и конструкционные материалы контейнера должны обеспечивать сохранение его целостности и работоспособности, в том числе прочностных характеристик в период последующего обращения с упаковкой ЖРО.

8.9. Конструкционные материалы контейнера и использованные для покрытия его поверхностей материалы должны обеспечивать защиту от атмосферных воздействий и возможность проведения дезактивации.

8.10. При наличии в упаковке РАО коррозионно-активных веществ внутренние поверхности контейнера должны быть обработаны антикоррозионным покрытием.

8.11. Если проектом ядерной установки, радиационного источника и пункта хранения не установлены способ, место и конкретные сроки захоронения кондиционированных ЖРО, то используемый контейнер должен сохранять целостность в течение ожидаемого периода хранения до захоронения и предотвращать неприемлемое распространение радионуклидов из упаковки РАО.

Контейнер должен обеспечивать возможность:

- извлечения упаковки РАО из хранилища в конце периода хранения;
- размещения упаковки РАО в дополнительный контейнер при необходимости;
- транспортирования упаковки РАО на захоронение;
- обращения с упаковкой РАО при захоронении.

8.12. Если упаковка РАО не соответствует установленным критериям качества РАО для транспортирования, и (или) хранения, и (или) захоронения, то с целью исключения несоответствия должен быть использован дополнительный контейнер.

8.13. Контейнеры и упаковки РАО, предназначенные для долговременного хранения и (или) захоронения, подлежат обязательной сертификации.

8.14. Хранение кондиционированных ЖРО должно осуществляться в специально оборудованных хранилищах с системой барьеров, предотвращающей поступление радионуклидов в окружающую среду выше пределов, установленных федеральными нормами и правилами в области использования атомной энергии. Технические характеристики барьеров, сроки хранения кондиционированных ЖРО и их количество устанавливаются и обосновываются в проекте ядерной установки, радиационного источника и пункта хранения в соответствии с требованиями настоящего документа и других федеральных норм и правил в области использования атомной энергии.

6.0 Attachment C

Collecting, Reprocessing, Storage, and Conditioning Liquid Radioactive Waste – Safety Requirements Presentation by V. A. Starchenko

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Gosatomnadzor of Russia
Federal standards and regulations
in the field of use of nuclear power

Collecting, reprocessing, storage and conditioning of liquid radioactive waste. Safety requirements.

НП-019-2000

Put into effect
since January 1, 2001

Purpose and area of application

The present document establishes requirements for ensuring safety when collecting, reprocessing, storing and conditioning LRW at nuclear facilities, radiation sources, and nuclear materials, radioactive substances and RW storages.

The present document doesn't apply to:

- treatment of LRW formed at production and enrichment of radioactive substances ores and other minerals;
- handling of LRW accumulated in superficial reservoirs of nuclear fuel cycle objects.

General safety requirements for collecting, reprocessing, conditioning and storing of radioactive waste

When establishing LRW quality criteria main characteristics of LRW, a container and LRW packing should be taken into account.

LRW characteristics:

- **chemical composition and phase status;**
- **total activity;**
- **radionuclide composition, specific alpha and beta activity.**

Characteristics of solidified LRW:

Cement compound:

- **radionuclide composition, specific alpha and beta activity, equivalent dose rate;**
- **water resistance;**
- **mechanical strength;**
- **radiation stability;**
- **thermal stability.**

LRW container characteristics:

- corrosion stability, radiation stability, configuration – for a metal container;
- density, porosity, water permeability, gas permeability, resistance to cold, radiation stability, resistance to microbes, mold and fungi, resistance to fire – for a ferroconcrete container.

LRW packing characteristics:

- radionuclide composition, alpha and beta activity, equivalent dose rate;
- total activity;
- homogeneity;
- mechanical strength (static, dynamic, impact loads);
- resistance to heat loads and thermal cycles;
- radiation stability.

When collecting, reprocessing, storing and conditioning of LRW the following potentialities should be excluded:

- **uncontrollable change of LRW state of aggregation, including formation of deposits and sediments;**
- **uncontrollable occurrence of exothermic reactions;**
- **uncontrollable formation of corrosive substances.**

At collecting, reprocessing, storing and conditioning of LRW containing nuclear hazardous fission materials the opportunity of a self-supporting chain reaction occurrence.

Collecting, reprocessing, storing and conditioning of LRW together with not radioactive waste are not authorized.

Safety requirements for liquid radioactive waste collecting

LRW collecting should be carried out separately depending on the following:

- **radionuclides half-life period (less than 15 days, more than 15 days);**
- **specific activity;**
- **concentration of alpha active radionuclides;**
- **chemical composition;**
- **phase composition;**
- **prospective reprocessing method.**

Organic dangerously explosive and fire hazardous LRW should be collected separately from other types of LRW.

Safety requirements to liquid radioactive waste reprocessing

LRW reprocessing should provide purification of LRW liquid phase and radionuclides concentrating in smaller volume.

Complete dehydration of high-salt aqueous LRW solutions is not allowed in the case of possible exothermic interaction of components of the LRW dry residue.

Safety requirements for liquid radioactive waste storing

Premises intended for accommodation of capacities for LRW storing should have not less than three-layer waterproofing and facing from corrosion-proof steel. Volume of the reveted premise should contain all amount of LRW taking place in capacities.

Control and observation chinks for earth waters sampling should be stipulated in the territory around of premises with capacities for LRW storing.

Premises in which there are capacities for LRW storing should be provided with:

- **leak alarm;**
- **leaks collection and recovery system;**
- **ventilation;**
- **radiation control;**
- **means for decontamination.**

- cement compound should have the following parameters of quality:

Quality parameter	Allowable values
Specific activity of compound	$<3{,}7\cdot10^{10}$ Bq/kg ($1\cdot10^{-3}$ Ci/g)
Beta activity	$<3{,}7\cdot10^{7}$ Bq/kg ($1\cdot10^{-6}$ Ci/g)
Water resistance (leaching rate for cesium-137 and strontium-90)	$<1\cdot10^{-3}$ g/cm^{2}day
Mechanical strength (compression strength)	≥ 50 kg-force/cm^2
Radiation stability	Mechanical strength not less than 50 kg-force/cm^2 after irradiation by the dose of 10^6 Gy (10^8 rad)
Resistance to thermal cycles	Mechanical strength not less than 50 kg-force/cm^2 after 30 cycles of freezing and defrosting (-40 - +40 ^{0}C)
Water resistance	Mechanical strength not less than 50 kg-force/cm^2 after 90-day immersion in water

Safety requirements for liquid radioactive waste solidification

The technological process of LRW solidification should provide reception of products with quality parameters established in the present document. Specific technical methods and means for LRW solidification are established and proved in nuclear facility, radiation source and storage designs.

The LRW solidification process should be fire- and explosion-proof and not be followed by formation of significant amount of secondary radioactive waste.

During LRW solidification using the method of cementation the following basic requirements should be carried out:

- the cementation facility should be in a separate premise supplied with ventilation system;
- the used inorganic binders (cement, Portland cement, slag Portland cement and others) should provide quality of a cement matrix according the requirements of the present document;
- LRW containing substances cooperating with cement with formation of toxic substances (for example, ammonium salts) can not be included in a cement matrix;

Safety requirements for liquid radioactive waste conditioning

LRW conditioning should provide LRW reduction into forms suitable for the subsequent transportation, storing and (or) burial.

Conditioned radioactive waste package should be free of:

- **strong oxidizers and chemically unstable substances;**
- **corrosive substances;**
- **poisonous, pathogenic and infectious substances;**
- **biologically active substances;**
- **highly inflammable, dangerously explosive and fire hazardous substances;**
- **substances entering in exothermic interaction with water followed by explosion;**
- **substances containing or capable to generate toxic gases, fumes or sublimates;**
- **liquid contents in the radioactive waste package should not exceed 3%.**

Radioactive waste containers and packages intended for long-term storing and (or) burial are subject for obligatory certification.