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V.M. Kuts

SE State Scientific and Engineering Center for Control Systems and Emergency Response (SSEC CSER)
Ministry of Energy and Coal Industry of Ukraine, Kyiv

PERSPECTIVES OF BRANCH STANDARDIZATION IN NUCLEAR POWER INDUSTRY AND NUCLEAR INDUSTRIAL COMPLEX

The issues of regulatory documents of the nuclear-industrial complex belonging to the Ministry of Energy and Coal Industry of Ukraine are briefly described, and the prospects are discussed of standardization process in the industry.

Key words: regulatory document, standard, nuclear power complex.

Industry standardization can be quite a capacious concept, but of course this is first of all a set of regulatory documents of appropriate level of approval. This article is about the regulations that were approved by the Ministry of Energy of Ukraine (Minenergouglya), and institutions which legal successor is Minenergouglya today (Ukraine Ministry of Fuel and Energy, Ministry of Energy of Ukraine, Ukraine Derzhkomatom, etc.) and which are under the responsibility of the Department of Nuclear Energy and Nuclear Industrial complex. It does not address document issued to support the Ministry's regulatory activities as such. This article considers purely technical regulations of different types (standards, regulations, procedures, etc), so-called "industry standards".

The State Enterprise "State Scientific and Engineering Center for Control Systems and Emergency Response" (SE SSEC CSER) on behalf of the Ministry since 1999 has maintain the Fund of Regulations of nuclear industrial complex of Ukraine, carries out their registration, and examines draft regulations prior to their approval.

Industry regulations that inherited from the Soviet Union. During the Soviet times central executive authorities directly managed the country's economy and nuclear matters, and the nuclear power production industry was not an exceptions. Much attention was paid to standards and other regulations at all levels, as tools for implementing technology policy. It was the Ministry that approved significant number of technical regulations. In December 1991, after the collapse of the Soviet Union Ukrainian nuclear power companies were united into the concern Ukratomenergoprom, which in January 1993 was reorganized into the State Committee of Ukraine on Nuclear Energy - Derzhkomatom of Ukraine. The Fund of industry technical regulations of industry level of approval that were inherited from the Soviet Union, numbered at that time over 2000 documents. Of course, the legitimacy of those documents was constantly questioned.

State Standard of Ukraine in 1998 in its letter gave explanations that the documents approved by the executive authorities of the former Soviet Union may be used for information, in those parts that do not contradict the legislation of Ukraine. In case of need, appropriate Ukrainian documents were to replace the former USSR documents.

Resolution of the Cabinet of Ministers of Ukraine of January 3, 2002, No. 2 "On the procedure and terms of validity of industry standards and similar regulations of the former Soviet Union" regularized the use in Ukraine of industry standards and similar other regulations of the former USSR. That Resolution provided the conditions of extension of validity of the documents of the former USSR, and the deadline for their use was set as 5 years after entry into force of the Resolution. Central executive authorities were proposed to consolidate the industry standards and other equivalent regulations of the former USSR, making and approving lists of documents that would continue to be used. It was proposed to develop new national documents for replacement of the obsolete ones in 5 years.

The Ministry of Energy of Ukraine approved in the prescribed order and provided to the State Standard Committee under that Regulation the list of documents of the former USSR, which validity were planned to be extended. In fact, two separate lists were drawn up for nuclear power industry and for nuclear manufacturing industry, which numbered about 700 documents each. Nevertheless, the new documents of Ukraine to replace the Soviet Union documents were not developed because there was no funding allocated for that purpose. Just the same situation existed in other industries. The end of that story was put by the Resolution of the Cabinet of 20.08.2005 No. 788, which cancelled the Resolution of

January 3, 2002 No. 2.

Of course, after the time that passed, the most part of even technical provisions of those USSR regulations become obsolete. However, the issue of their validity still occurs. For example, back in 2014 SSEC CSER received a letter from the Energoatom with a request to provide information on the point. In the letter, the Energoatom, which is constantly working on updating its own list of regulations as the operating organization, requested Minenergouglya to clarify certain issues regarding industry regulations of the former USSR.

Ukrainian regulations of the nuclear industry complex. The Derzhcomatom and the Ministry of Energy had appropriate service departments for industry standardization, the leading and basic organizations were appointed to perform that work in certain areas of activity, and active development and adoption of industry regulations was initiated. For codification of those documents the State Standards Committee of Ukraine approved the use of code # 95, which belonged to the former USSR Ministry of Middle Engineering (in charge of nuclear industry, uranium mining and a number of other fields). The documents had codifications as GND 95, and KND 95. By their type those documents were mainly provisions of various services, guides, lists of indicators of specific industrial processes, and standards for products used in industrial activities.

However, the number of regulations to be approved by the Ministry began gradually decline. It was a reflection of the process of complete transfer of the management functions from the Ministry to businesses (SE NNEGC Energoatom, SC Nuclear Fuel, etc.). That is, instead of industry standards they began to develop mainly standards of enterprises. Exceptions were made in the cases when standards required coordination with other Ministries.

The last reorganization in the sector standardization was carried by the Ministry of Fuel and Energy of Ukraine in 2006. New procedures for standards development and approval by the Ministry of Energy standards were adopted:

SOU-N MBE 001: 2006 Regulations for drafting regulations;

SOU-N MBE 002: 2006 Rules of construction, presentation, design and content requirements of regulations;

SOU-N MBE 003: 2006 Regulations approval, registration and preparation for publishing.

SE SSEC CSER by order of the Ministry of Energy of 27.12.2006 No. 531 “On appointment of the Leading and Basic standardization organizations in the nuclear-industrial complex” was again appointed as the organization for reviewing, registration and maintenance of the Fund of regulations of nuclear-industrial complex.

The new Law of Ukraine “On Standardization” of 06.05.2014 No. 1315-VII, which was entered into force on 01.01.2015 significantly changed the situation in the industrial standardization. Under paragraph 3 of the “Final and Transitional Provisions” of the Law industry regulations “are applied until their replacement by technical regulations, national standards, codes of practice or their cancellation in Ukraine, but not more than 15 years since the enactment of this Law”. Further development of new industry standards was not provided for by that Law. For 15 years from the date of enactment of this Law central authorities have the right in their respective fields and within their authority to carry out activities of industry standardization, except for the development and approval of new industry regulations.

Pursuant to this Law Minenergouglya recognized its own standards on procedures for the development of regulations as being invalid:

SOU-N MBE 001: 2006, SOU-N MBE 002: 2006, SOU-N MBE 003: 2006.

From 01.01.2015 not any new industry regulations in the nuclear-industrial complex was approved.

However, by the order of the Minenergouglya of 20.05.2015 No. 305 “On organization of works on standardization in the fuel-and-energy sector” the leading organizations of industry standardization were re-appointed. SE SSEC CSER was again appointed as the leading organization for standardization in nuclear power industry. State Enterprise “Ukrainian Scientific Research and Designing Institute of Industrial Technology” (Zhelty Vody) was appointed as the leading organization for standardization in atomic-industrial complex. That order indicated the intention of the Ministry to continue work on industry standardization. The surprising point in that order was significant innovation - expanding the scope of the main organizations for standardization on the regulations (standards) that belong to businesses and organizations of the industry. That was a completely new approach to standardization, because up to that moment the institution or organization that has endorsed the standard, was further responsible for its reviewing, updating, and validity of their existing regulations.

As of October 2015, the Fund of regulations of nuclear-industrial complex had 141 normative documents (documents of the industry level of approval without limitation of their term of validity and for which there are no orders to cancel them). This does not include legislative documents and regulations on general issues of power-generating industry, on physical protection, occupational safety and fire safety, which are under the responsibility of relevant departments of Minenergouglya of Ukraine.

Conclusions

First, there is a need for critical analysis and review of the Fund of regulations of the nuclear-industrial complex. According to the new Law of Ukraine "On Standardization", industry regulations should be replaced with national standards, or standards of enterprises, or canceled. SE SSEC CSER offers to validate the existing regulations with the conclusions about each document, with sending materials to industry stakeholders and processing the feedback. Available materials can be submitted for collective discussion, for example, at the section of the Science and Technology Council of Minenergouglya, which will make a decision on each applicable regulation of the nuclear-industrial complex. In the future, these decisions can be approved by Minenergouglya. SE SSEC CSER proposal on such work was sent for putting into the Plan of Scientific and Engineering Works of the Minenergouglya in 2015

Secondly, in the light of the new Law of Ukraine On Standardization, the future of industrial standardization looks very uncertain. In 15 years no industry regulations should remain. This provision of the Law certainly indicates the intension at eliminating industry-level standardization. But the question of work in the transitional period remains open. During these 15 years central authorities need to make decisions on standardization work in their fields and within their authority. So we still have to wait that Minenergouglya will formulate a more detailed vision regarding the work on standardization in the industry. And it should be noted that this article considered only a relatively small part of all industry regulations. Even the nuclear-industrial complex, in addition to the documents discussed in this paper still applies regulations on general issues of power-generating industry, on physical protection, on occupational safety and fire safety. And then there are the thermal generation, renewable energy, hydroelectric power, oil-and-gas sector, the coal industry. So far, Minenergouglya order of 20.05.2015 No. 305 "On organization of works on standardization in the energy sector" generates more questions than answers regarding the continuation of works on standardization in the energy sector of Ukraine.

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V. Skalozubov [1]***, S. Vasilchenko*** [2]***, I. Kozlov*** [3]***, T. Gablaya*** [1]

[1]*Institute for Safety Problems of Nuclear Power Plants, NAS of Ukraine, Kiev*
[2]*NPP Operation Support Institute, Kiev*
[3]*Odessa National Polytechnic University, Odessa*

REASSESSMENT OF NPP SAFETY IN UKRAINE UNDER INFLUENCE OF TORNADOES

The article describes the main provisions regulating the danger of tornadoes for nuclear installations, as well as the analysis of known results of assessments of impact of tornadoes on the safety of Ukrainian NPPs obtained in the "before-" and "post-Fukusima" periods. The analysis reveals insufficient justification of the estimates of tornado frequencies, and exclusion from consideration of the emergency events with flooding of industrial sites under the impact of tornadoes at least of the 2nd class of intensity. The necessity is underlined of re-evaluation of the safety of Ukrainian NPPs with regard to the reasonably established characteristics of tornado-hazardous zones and lessons learned of the Fukushima accident.

Keywords: tornado hazardous areas, safety of nuclear power plants

A whirlwind (tornado) usually means a strong, small-scale atmospheric vortex of great destructive power, up to 1000 meters in diameter, where the air is rotated at a high speed (up to 100 m/s) [1].

The defining gas-dynamic characteristics of the power of tornadoes are:

- maximum horizontal speed of the rotational motion of the walls of the tornado V;
- forward speed of a tornado motion U;
- length L_k and width W_k of the path of the tornado;
- differential pressure between the periphery and the center of rotation of the vortex Δp.

The ranges of changes of the main gas-dynamic characteristics correspond to a particular class of the intensity of tornadoes on the F-scale Fujita-Pearson [2] (see. Table 1)

The maximum design wind pressure at the impact of tornado should be accounted as a vector sum of the maximum horizontal speed of the tornado wall rotary motion V and forward speed of a tornado motion U.

The difference in atmospheric pressure depending on the distance (radius) r from the center of the vortex of a tornado is defined by [1]

$$p_a \;\; r = \rho \frac{V_m^2}{2}\left(2 - \frac{r^2}{R_m^2}\right); \; 0 \le \mathbf{r} \le \mathbf{R_m}; \qquad (1)$$

$$p_a \;\; r = \rho \frac{V_m^2}{2} \cdot \frac{R_m^2}{r^2}; \; \mathbf{r} \ge \mathbf{R_m}, \qquad (2)$$

where V_m — maximum tangential wind speed; R_m — radius corresponding to the maximum speed of the air flow rotation; ρ — density of air.

For a nuclear facility (NF) a tornado hazardous event means passage through a NF site of a tornado capable of causing damage to the NF with possible radiological consequences; tornado hazardous district is an area in which the probability of a tornado passage through a fixed point exceeds the allowable limit of a tornado hazardous event; effective observation period means the time interval during which the frequency of tornadoes in a considered region is constant and equal to the frequency of occurrence of tornadoes for a period of regular observation.

Statistical data on tornadoes passage through the chosen site of NF location / construction determine a decision to set the acceptable limit of probability of tornado hazardous event P_0. Taking into account the recommendations of [1], it can be taken equal to 10^{-4}.

Tornado hazardousness of the territory of NF location is evaluated by establishing an annual probability P_S of occurrence of tornado hazardous events in the area of NF location / construction within the area surrounding the object of 1,000 km^2 [1].

If for the place of NF location within the surrounding area of 1,000 km^2, located at the territory with homogeneous physical and geographical conditions of the formation of tornadoes, the annual probability of a tornado is higher than P_0 ($P_S > P_0$), then the area is tornado-hazardous that requires determination of the main characteristics of tornadoes.

Table 1: Correspondence with the intensity class of tornadoes and the range of the main gas-dynamic characteristics

Intensity class k	Characteristics ranges				
	Maximum horizontal speed of the rotational movement of a tornado wall V, m/s	The translational speed of a tornado U, m/s	The length of the tornado path L, km	The width of the tornado path W, m	Pressure difference between the periphery and the center of the tornado vortex Δp, GPa
0	Up to 33	Up to 8	Up to 1,6	Up to 16	Up to 13
1	33 — 49	8 — 12	1,6 — 5,0	16 — 50	14 — 31
2	50 — 69	13 — 17	5,1 — 16,0	51 — 160	32 — 60
3	70 — 92	18 — 23	16,1 — 50,9	161 — 509	61 — 104
4	93 — 116	24 — 29	51 — 160	510 — 1600	105 — 166
5	117 - 140	30 — 35	161 — 507	1601 — 5070	167 — 249

In analyzing the parameters of the tornado hazardousness of NF site one should take into consideration, starting with the 3-class tornado intensity, the items which can be moved by a tornado, in accordance with the IAEA recommendations [2]:

- Vehicles weighing 1800 kg;
- 200 mm armor-piercing artillery shell weighing 125 kg;
- Solid steel sphere with a diameter of 2.5 cm.

The area of action of the load shall be equal to the cross sectional area of an object. The direction of movement of the object in the collision with a construction is taken as the most unfavorable, i.e., perpendicular to the outer surface of the structure. Location of a collision point can be arbitrary, i.e., at any point on the outer surface of structures.

The annual probability P_s of occurrence of a hazardous tornado event in the area of location and construction of a nuclear installation of 1000 km^2 surrounding the nuclear site, located at the territory of A km^2 with similar physical and geographical conditions of the formation of tornadoes is determined by the formula [1]:

$$P_S = \frac{S \cdot 10^3}{AT}, \tag{3}$$

where S - the total area of the zone of destruction caused by tornadoes in the area A; T - effective observation period.

To evaluate the effective observation period T in this region (zone) by analyzing the chronological chart for the recorded tornadoes one should select a maximum homogeneous period T_0 (by frequency of tornadoes), during which m_0 tornadoes were recorded. The value T should be determined from the condition of a constant frequency of tornadoes by the formula:

$$T = T_0 \frac{m}{m_0}, \tag{4}$$

where m — the total number of tornadoes recorded in the area.

The annual probability of passage of a tornado with the class of intensity k through an area of A, in which the nuclear site is located, should be determined based on the ratio:

$$\boldsymbol{P = P_S\,[1 - F(k)],} \tag{5}$$

where $F(k)$ — the probability of non-exceeding f class k among tornadoes recorded in the area.

The total number of tornadoes N, passed through the area under consideration, and the total area S of destruction should be determined using the expressions:

$$N = \sum_{k=0}^{m} n_k a(k), \quad S = \sum_{k=0}^{m} n_k a(k) L_k W_k, \tag{6}$$

where n_k — the number of registered tornadoes of class k; L_k - the length of the path of a tornado; W_k - the width of the path of the tornado.

The estimated class of intensity of a probable tornado at the territory of nuclear facility location should be determined taking into account the requirements of:

$$F(k_p) = 1 - \frac{P_0}{P_S} \tag{7}$$

using the expression

$$k_p = -\frac{1}{a}\left[\ln\left(1 - \frac{P_0 AT}{S \cdot 10^3}\right) + b\right] \tag{8}$$

The value of $F(k)$ is determined under the condition:

$$\boldsymbol{P_S > P_0} \tag{9}$$

If the condition (9) is not satisfied, determination of danger of tornadoes and tornado design characteristics is not conducted and the area of NF location is accepted as safe on the probable impact of tornadoes [1]

Analysis of tornado hazard assessments for nuclear power facilities in Ukraine. According to statistical data registered in the period from 1844 to 2001 the most part of NF in Ukraine are located in tornado-hazardous areas: calculated class of tornadoes intensity $k_p \geq 2$ with annual probability of occurrence P_S of a tornado-hazardous event 10^{-4} year^{-1} [1].

In the "before-Fukushima" period all assessments of tornado hazardousness for Ukrainian nuclear power plants have been implemented in the framework of Safety Analysis Reports for External Extreme Impacts (SAR EEI). The main results of these assessments for Zaporozhye NPP are presented in Table 2 (for example, see [3]).

Analysis of the impact of tornado hazardousness on the NPP safety in the SAR EEI is actually boiled down to assessments of the overall probability of tornadoes (frequency of passage of tornadoes) in the area of NPP location at conservative assumption that occurrence of tornadoes of different intensity class (including those of not less than the 2nd class) results in serious accidents. As a result, for Zaporozhye NPP it was determined that among all possible extreme events tornadoes give the greatest contribution in the core damage frequency (CDF), which is 14.3% of the base CDF for internal initiating events, and the most critical systems to the safety at tornadoes impact are normal power supply systems and industrial water supply of main consumers. However, on the basis of conservative estimates of a relatively small influence of tornadoes on the total CDF value the corresponding emergency response organizational and technical measures had low priority.

Similar results were obtained for other nuclear sites in Ukraine.

The 'stress test' analysis conducted in the "post-Fukushima" period for re-evaluation of nuclear power safety of Ukraine taking into account the lessons learned of Fukushima accident [4] did not reveal any additional (new) safety deficiencies with respect to extreme natural phenomena (including tornadoes).

With regard to the main results of tornadoes hazardousness estimates for Ukrainian NPPs in the "before- " and "after-Fukushima" analysis one should note, that:

1. Estimates of SAR EEI frequency of tornadoes contradict the established categories of tornado hazardous zones. As an example, for the area of Zaporozhye NPP (zone B of increased tornado hazardousness) the tornadoes intensity class is 3.58 at a frequency of passage $87{\cdot}10^{-4}$ year^{-1} [1]; but in SAR EEI it $8.19{\cdot}10^{-7}$ year^{-1}. It was this (and the similar ones) assessment of unreasonably low frequency in SAR EEI for tornado that eventually allowed to come to the conclusions about relatively small impact of tornadoes on the basic CDF even at excessively conservative assumptions on the conditional probability of core damage when exposed to tornadoes (assumed to be equal to unity).

2. One of the main lessons of the Fukushima accident is inadmissibility of exclusion from consideration (modeling, analysis, emergency measures) of relatively improbable extreme natural phenomena (including their combined effect). Therefore, adopted in SAR EEI low priority of consideration of the impact of tornadoes on the NPP safety is unjustified (and even more so in the areas of increased hazard of tornado).

3. Exclusion from consideration of the event of flooding of the NPP site, caused by the impact of a tornado (including the case of the combined action with other extreme natural events) was substantiated insufficiently.

Table 2. Data on the number and characteristics of tornadoes for the B zone of increased tornado hazard (region of Zaporozhye NPP location)

Tornado class	Number of registered tornadoes	Coefficient *a(k)*	Number of actual tornadoes	Length of tornado path, km	Width of tornado path, km	Area of tornado path, km^2	Area of tornado path for all tornadoes of class *k*, km^2	Tornado frequency
0	29	1,5	44	0,90	0,01	0,01	0,36	$1{,}13\cdot10^{-8}$
0,5	1	1,5	2	1,61	0,02	0,03	0,04	$1{,}23\cdot10^{-9}$
1	33	1,5	50	2,86	0,03	0,08	4,05	$1{,}28\cdot10^{-7}$
1,5	2	1	2	5,09	0,05	0,26	0,52	$1{,}64\cdot10^{-8}$
2	18	1	18	9,05	0,09	0,82	14,74	$4{,}66\cdot10^{-7}$
2,5	1	1	1	16,09	0,16	2,59	2,59	$8{,}19\cdot10^{-8}$
3	8	1	8	28,61	0,29	8,19	65,49	$2{,}07\cdot10^{-6}$
3,5	1	1	1	50,88	0,51	25,89	25,89	$8{,}19\cdot10^{-7}$
4	1	1	1	90,48	0,90	81,87	81,87	$2{,}59\cdot10^{-6}$
Total	94		126			119,73	195,54	$6{,}18\cdot10^{-6}$

For example, at the known event of 24.07.1991, in the region of the Matsesta river and the Bzugu river valleys (Krasnodar district) the tornado that came from the sea (intensity class 2) raised the water level up to 5 m, which led to great destruction of buildings and communications, as well as the loss of human life. Also on the territory of the former Soviet Union dozens of events were recorded with the tornadoes of at least 2nd class of intensity that were originated at water bodies and had disastrous consequences [1]. Thus, for the region of Zaporozhue NPP location for a class of tornado intensity more than 3 the specific (per unit area) lifting force is almost 2.5 times ([1]) greater than the corresponding value in the above examples. Therefore, an additional analysis of possible flooding of the industrial site of NPP in tornado-hazardous areas is necessary (including the combined effect with other extreme natural events).

Conclusions

It is necessary to re-evaluate the safety of all nuclear power plants in Ukraine, taking into account characteristics of the revealed tornado hazardous zones and lessons learned of the Fukushima accident as regards the low-probable extreme natural phenomena.

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Yu.A. Kutlahmedov, I.V. Matveeva

National Aviation University, Institute for Ecological Safety, Kiev, Ukraine

EXPERIENCE AND PROSPECTS OF THE "TURF CUTTER" SOIL DECONTAMINATION TECHNOLOGY

The study evaluates the effectiveness of decontamination technology for contaminated soil using skimming sod by "Turf Cutter" installation. It compares the results of decontamination of pastures by using "Turf Cutter" technology and by traditional methods of pastures cultivation (grass seeding, land reclamation, etc.). Comparison of the "harm-benefit" results for these two methods shows that the decontamination of the soil after the Chernobyl accident with the "Turf Cutter" technology is almost twice more efficient than traditional methods of improvement of pastures and meadows. It stressed also the need to develop this technology and to implement it in Ukraine.

Key words: ecosystems, radioactive contamination, decontamination of soil, skimming of sod on pastures.

Evaluation of the effectiveness of the "Turf Cutter" technology of soil decontamination. The researches that we initiated within the framework of the international project ECP-4 "Technologies and strategy of decontamination" [1], allowed us to develop and test a new technology of decontamination of contaminated soil by removing a thin layer of sod (2-5 cm) by vibrating knife of a special machine "Turf-Cutter" capable to respond to repeated irregularities of microrelief. Our experiments were conducted on radioactively contaminated soils in the 10-km zone of Chernobyl and other territories of Ukraine and Belarus in 1992-1998. The experiment used the US-made machine «Turf Cutter». The first test of the technology was conducted on the territory of the radio-ecological fully sodded site "Buryakivka" 4 km away from the Chernobyl industrial site which had contamination level of 100 Ci/km^2 on ^{137}Cs, 80 Ci/km^2 of ^{90}Sr, 7 Ci/km^2 of ^{239}Pu. The studies have shown that up to 95% of the radioactivity at the unploughed part of the site were concentrated in the upper layer of sod. As a result of testing at the selected part of the site we managed to achieve high efficiency of decontamination of soil with decontamination factor K_d = 25-40 [2].

The second test of the technology was conducted at the site "Chistogalovka" 3 km from the ChNPP site. That area was characterized by high levels of radionuclide contamination (150 Ci/km^2 on ^{137}Cs), poor sod on light sandy soil and unplain surface. Removing of sod on this site range has made it possible to perform sufficiently effective decontamination of soil, with K_d = 10-15.

Another testing of this technology was conducted in the Belorussian part of the Chernobyl Exclusion Zone. It has demonstrated the ability to selectively remove the sod in a spotted radionuclide contamination. Rapid assessment of spot contamination was carried by a field gamma spectrometer "Korad." According to the spot assessment of the contamination the selective removal of sod was performed at the site, which reduced the volume of the removed turf by 70%. In this case the K_d on 137Cs for the whole area was 5-7 units

Than we tested the technology in 1993 in Milyachi village (Rivne region, Dubrovitsky region, Ukraine) at the pasture "Stav" with drained peat soils, and contamination on ^{137}Cs of about 5 Ci/km^2, which was free from other protection measures. After removing the contaminated sod (K_d = 15-20), perennial forage grasses were sown at the site. The level of contamination of those herbs was 20 times lower than in the control plots. Comparison of milk contamination from experimental cows fed with grass from decontaminated plots and milk from control group of cows, fed with grass taken from neighboring non-decontaminated areas showed that the level of contamination of milk dropped 15-20 times [2].

These findings confirmed the high efficiency of the proposed soil decontamination technology through removal of sod by the "Turf Cutter" special machine. The obtained K_d values ranged from 7-15 on the silty sand and sandy soils with a loose sod to 20-40 on the drained peat soils with the most dense sod (turf). According to the field test, it was shown the dependence of efficiency of the decontamination technology on the nature of the soil, vegetation and landscape conditions and showed a high degree of environmental safety. This allowed, using the elements of GIS technologies, to perform the appropriate evaluation of redistribution of radionuclides and zoning of the contaminated area, concerning the possibility and efficiency of the "Turf Cutter" technology for decontamination of soil and highlight areas where the expected decontamination will be the most efficient [3].

Application of the "Turf Cutter" soil decontamination technologies on the territory of Milyachi village (Dubrovitsky district, Rivne region). An example of effective use of protective measures is the result of their implementation on the soil of Milyachi village, Dubrovytsky district, Rivne region (see Table 1).

It can be seen that the decrease in the collective dose for the population of Milyachi is about 380 man-rem. Expected collective dose to the population on their dietary pattern is about 1800 man-rem. Thus, a dose reduction to 380 man-rem from the use of protective measures in the private sector is not significant for the population. Reducing the collective dose in the collective sector by 1284 man-rem is a significant amount, but it is aimed mainly at reducing the export dose.

We analysed the possibility of using the method of removing the top layer of soil (turf) on the Milyachi territory. Data from this analysis are shown in Table. 2. The area in which it is possible to effectively use the installation "Turf Cutter" is estimated at 340 hectares of peat soils. This pastures have not been plowed since the Chernobyl accident.

It should be emphasized that the protective measures, widely used in agriculture, as a rule, do not change (does not degrade) the quality of agricultural ecosystems, and thereby do not reduce radiocapacity factor values (see. Table. 1). The exception is the use of the "Turf Cutter" machine for removing sod layer of 2-5 cm. In this case a part of the topsoil is lost, causing some reduction of the radiocapacity factor of soil (F = 0,9). But the mechanical removal of the fertile layer (10-15 cm) with a bulldozer and other heavy equipment is particularly dangerous for the ecosystem.

Prospects of soil decontamination by the "Turf Cutter" technologies in Ukraine [2]. We provide evaluation of the difference between the "benefit - harm" for this particular protective measure in Table 3.

Table 1. Evaluation of the protective measure implemented by private firms the Milyachi village in 1988-1993 years for milk (Bq/l)

Protective measure	Number of cattle	Amount	Cost of the protective measure (USD)	Content of ^{137}Cs in milk (Bq/l)		K_d on milk	Lowering of the collective dose, man-rem	Net benefit (USD)
				before 3M	after 3M			
Boluses	80	240 items	720	500	220	2,2	28	509
Humolit	250	45 tons	270	500	280	1,8	314	2143
Ferrocyn	50	7 kg	55	500	200	2,5	29	171
Turf Cutter	0,5 hectares	—	12	720	40	20	5	27
TOTAL							376	2850

Notes.

1. K_d – is determined defined as the ratio of milk contamination level before applying protective measures to its value after the use of protective measures.

2. The analysis of the "Harm-benefit" ratio. The net benefit is the difference between the cost of the countermeasures and the value of reductin of the collective dose, multiplied by the cost of man-rem - 40 USD.

Table 2. Expected decrease in the collective dose for the use of milk taken from private pastures decontaminated by the method of removing sod (K_d = 20)

Type of soil	Content ^{137}Cs Ci/km^2 (area, hectares)	Contamination of milk (Bq/l)		Lowering of the collective dose, man-rem
		before	after	
Podzolic	2-5(100)	150-200	10	242
	5-15(60)	400-600	25	435
Peat soil	2-5(110)	200-300	15	383
	5-15(70)	600-900	40	751
TOTAL				1810

Table 3. Evaluation of the expected benefits and reduction of the collective doses after the use of "Turf

Cutter” technology in the contaminated areas of Ukraine (1-15 Ci/km^2)

Region	Area, thousand hectares	Lowering od the collective dose thousand man-rem	Cost of “Turf Cutter” work, thousand $/ hectare	Benefit, thousand $	“Harm-benefit” тыс $
Level of contamination ^{137}Cs (1—5 Ci/km^2)					
Kiev	1	5	25	200	175
Zhitomir	11	55	275	1760	1485
Rivne	15	75	375	2400	2025
TOTAL	27	135	675	4360	3685
Level of contamination ^{137}Cs (1—5 Ci/km^2)					
Kiev	0,3	4	7,5	128	120
Zhitomir	2,5	30	62,5	960	897
Rivne	1,0	12	25,0	384	359
TOTAL	3,8	46	95	1472	1377

The cost per person-rem of 40 USD was taken by the laws of Ukraine (the lowest evaluation in comparison with Western countries). [3] Overall benefits of the widespread use of the “Turf Cutter” technology on the unploughed pastures and meadows in Ukraine is estimated at 5062 thousand USD [5].

Conclusion

The above analysis of the data on improvement of contaminated pastures in Ukraine showed that the evaluation of “harm-benefit”difference is about 1668 thousand USD. Comparison of the “harm-benefit” difference of the two methods shows that decontamination of soil after the Chernobyl accident with the “Turf Cutter” technology is almost twice more efficient than the traditional methods of improvement of pastures and meadows. Further, over time this difference will increase as meadows and pastures improvement should be repeated every 3 years and decontamination using the “Turf Cutter” technology is sufficient to be applied just once. These figures once again underline the high efficiency of the method of soil decontamination in the pastures and meadows by the “Turf Cutter” technology, as well as the need to develop this technology and use it in Ukraine.

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Yu.V. Bondar, S.V. Kuzenko
Institute for Environmental Geochemistry, National Academy of Sciences of Ukraine, Kiev

ADSORPTION OF STRONTIUM ONTO COMPOSITE FIBRES COATED WITH FERRIHYDRITE LAYER

New composite fibers were synthesized by in situ deposition of a layer of two-line ferrihydrite on the surface of modified polypropylene fibers. Composite fibers have demonstrated high chemical stability both in acid and alkaline solutions. Results of strontium ions adsorption onto synthesized fibers depending on the contact time, pH, initial strontium concentration and amount of calcium ions in model solutions are presented. It has been founded pH-dependent and two-stage strontium adsorption behavior as well as suppression of adsorption by calcium ions.

Key words: liquid radioactive waste, adsorption, strontium, composite fibers, polypropylene, ferrihydrite

Sustainable development of nuclear power depends on environmentally acceptable solution of the problem of liquid radioactive waste (LRW) management. A special problem is treatment of low-level liquid radioactive waste because of the large amount of accumulated waste, and due to their permanent formation.

In many cases, an effective way of LRW treatment is extraction of radionuclides by sorption to achieve radiation safety of the purified solutions.

Strontium-90, along with Cesium-137, causes more than 90% of the activity of low-level liquid radioactive waste. Despite the considerable range of organic and inorganic adsorbents synthesized for the selective binding of Sr-90, the search for new low-cost and efficient sorption materials is still an urgent task.

Field observations in the contaminated areas of Belarus have shown that accumulation of Sr-90 and Cs-137 in quantities 5-10 times exceeding the background levels, occurs at the amorphous and poorly crystallized iron and manganese hydroxides [1]. The study of forms of radionuclides in the samples revealed the connection of the bulk of Sr-90 with the oxide fraction (32-58%) at relatively high rates of exchange forms (16-48%) and water-soluble forms (1-9%), and in some places the increased share of organogenic (5-18%) forms [2]. Laboratory tests have confirmed the high sorption capacity of artificially produced amorphous hydroxides of iron and manganese with respect to Sr-90 and Cs-137 at pH=4.9 (close to nature values). Thus, the as-precipitated iron hydroxide ($Fe(OH)_3$) has demonstrated higher absorbent capacity relatively to radioactive strontium, while hydrated manganese oxide ($MnO(OH)_2$) had better sorption of radioactive cesium [3].

In other work core samples have been studied drilled in the rocks near the liquid radioactive waste storage pond (Lake Karachay, Russia) [4]. Radiogeochemical analysis of the core samples of pieces-monoliths taken from different depths, allowed the authors to find that the main activity is concentrated in the surface section (up to depth of 30-40 m). The highest specific activities are confined to different types of cracks, the surface of which is covered with iron hydroxides, clay material, leucoxenizided with sphene and leucoxene. The main beta- and alpha-activity is caused by Strontium-90 (3470 Bq / kg), Cesium-137 (56 Bq / kg), Uranium-238 (234) (24-27) Bq / kg), thorium-230 (22 Bq / kg), as well as Cobalt-60, and Plutonium-239 (240). The authors concluded that sorption is the main mechanism of retardation of radionuclides migrating through cracks of rocks. It is caused by such natural sorbents as iron hydroxide and clay minerals.

Among the 16 major mineral types of iron oxides and hydroxides the highest adsorption activity was demonstrated by ferrihydrite [5]. Ferrihydrite ($5Fe_2O_3 \cdot 9H_2O$) - ferric oxyhydroxide, which exists exclusively in the form of poorly crystallized (or “amorphous’) nanoscale crystals (2-6 nm) and their aggregates [5]. Ferrihydrite mineral as a new species was first described by Chukhrov in 1973 [6], and is still erroneously called as “amorphous iron oxide” or “hydrated iron oxide” [5].

Ferrihydrite (Fh) is formed as a result of the dissolution and subsequent oxidation of Fe-containing minerals and is a product of the geochemical changes in the supergene zone. Fh is thermodynamically unstable and undergoes a natural mineral transformation (aging) into the more stable goethite and hematite [5-7]. The high specific surface area, excellent sorption capacity, and low cost shows that ferrihydrite can be considered as a promising adsorbent material.

Ferrihydrite can be easily synthesized in laboratory conditions in the form of a colloidal solution or in the form of ultra fine particles after low-temperature drying [5, 7]. However, in this form Fh is unsuitable for practical use as an adsorbent because of its difficult separation from the purified solution [8].

A new technology has been proposed for synthesis of composite sorption materials by controlled in situ deposition of Fh onto/into a solid matrix, in which both inorganic and polymeric matrix were used [9, 10].

Composite adsorbent based on iron hydroxide-coated sand (IOCS) was used for isolation of Sr-90 from a model liquid radioactive waste, simulating the LRW from waste canisters at Hanford Nuclear site. The model LRW had alkaline pH, high salt background, and high concentrations of sodium - (5.5 mol / dm^3) and nitrate ions (3.7 mol / dm^3) [11]. The paper showed that the composite adsorbent based on IOCS could selectively remove Sr-90 even in the presence of competing ions of Ca^{2+}, Al^{3+} and Cr^{3+}.

Another study compared the adsorption capacity with respect to Sr-90 of natural montmorillonite from Cherkassy deposit and montmorillonite coated with iron hydroxide (III) [12]. It was shown that the presence of iron hydroxide on the surface of montmorillonite increased the Sr-90 adsorption over a wide pH range.

Unfortunately, in many studies on the synthesis and use of composite adsorbents with precipitated iron hydroxide phase, mineralogical diagnosis of the precipitate phase was not carried out. Meanwhile, this issue deserves attention due to the fact that ferrihydrite, as a metastable mineral can be transformed into a more stable goethite and / or hematite, and of great scientific interest are the issues related to further behavior of the deposited contaminant [5, 6].

The aim of this research was to perform synthesis of the composite adsorbent based on Ferrihydrite deposited on a surface modified with polypropylene fibers, and to study strontium sorption processes from model solutions onto the synthesized adsorbent.

Experimental Procedure

Synthesis of composite fibers. As the base polymer for the synthesis of composite adsorbents based on ferrihydrite we used modified polypropylene fibers (PPO) with grafted chains of polyacrylic acid (PAA) [13, 14]. PPO with grafted PAA chains were placed in $FeCl_3$ solution for 12 hours; then they were washed in distilled water and dipped in NaOH solution (pH=9.8) for 10 minutes to precipitate the ferric hydroxide colloid particles (iron oxides hydroxides). Fibers with the deposited layer of iron oxide hydroxides were washed thoroughly in double distilled water to remove sodium and chloride ions, and than dried at 60° C for one day.

Morphology of fibers before and after the synthesis was analyzed using a scanning electron microscope (SEM) JEOL JSM6490LV. X-ray studies were carried out on a diffractometer DRON-3 using Cu Kα-radiation. Recording was conducted at room temperature in a range of angles 2θ of 10 to 90 degrees in step scanning mode.

Stability of composite fibers in aqueous solutions with different pH values were determined by placing a sample of synthetic fibers (0.1 g) in a flask with 10 cm^3 of solution. After the desired contact time, the solution was filtered and the iron content was determined by atomic absorption spectrophotometer (Model AA-8500, Nippon Jarrell Ash Co Ltd., Japan)

The adsorption of strontium ions. Study of the process of adsorption of strontium ions on the synthesized composite fiber was performed in static conditions. Initial solutions were prepared using salts of strontium chloride ($SrCl_2 \cdot 6H_2O$, Duksan Pure Chemicals Co.). In a series of test tubes we placed fiber (0.1 g) with 15 cm^3 of poured solution with the initial concentration C_0 and wait the required time. After adsorption the solution was separated from fibers by filtration through a glass filter and measured concentration of Sr in it by atomic absorption spectrophotometer.

The amount of adsorbed Sr, A_{Sr} (mg/g) was calculated by the formula:

$$A_{Sr} = \frac{(C_0 - C_t) \cdot V}{W} \tag{1}$$

where C_0 and C_t — initial and final concentration of strontium in the test solution, mg/dm^3, V – volume of the solution, dm^3, and W - weight of the adsorbent, g.

The efficiency of extraction of strontium from the solution, E_{Sr} (%), was determined under equilibrium conditions by the formula:

$$E_{Sr} = \frac{C_0 - C_P \cdot}{C_0} \cdot 100 \qquad (2)$$

where Cp - the equilibrium concentration of strontium in the solution.

All reagents were of "chemically pure" or "high purity" grade. For preparation of the solutions, distilled water was used, and pH of the solutions was adjusted by adding several drops of HCl or NaOH.

In the experiments on desorption the fibers with known amount of adsorbed strontium were washed in the distilled water and placed in a flask with 20 cm^3 of desorption solution. Concentration of the desorbed Sr in the solution was determined by atomic absorption spectrophotometer.

Results and discussion

Both natural and synthetic forms of ferrihydrite are formed as nanocrystals with 2-6 nm size. By the number of peaks in the X-ray diffraction patterns two modifications of ferrihydrite can be identified – the 2-line one, and the 6-line one. On the X-ray diffraction pattern of the two-line ferrihydrite (2LFh) 2 broad peaks are registered, and in X-ray diffraction pattern of the six-line ferrihydrite (6LFh) we have 6 broad peaks. The broadened X-ray peaks indicate a very small particle size and / or low structural order ("amorphous"). Fh in transmission electron microscope can be observed as individual spherical particles with 2-4 nm sizes for 2LFh, and 5-6 nm sizes for 6LFh [5, 7]. It was found that the structures of the two modifications of ferrihydrite were almost identical and differ only in particle size [15].

Quick hydrolysis of the solution of Fe (III) salt at low pH and increased temperature (~80 ^{o}C) leads to the formation of 6LFh, while the rapid rise in pH of the solution of Fe (III) salt to 7 – 9 at room temperature leads to the formation of the two-line 2LFh [3, 5, 15].

In this paper as a basis for synthesis of composite adsorbent we chose polypropylene fibers with grafted chains of polyacrylic acid. Choosing this basis was caused by the fact that polypropylene fibers have excellent physical and chemical properties (low density - 0,91-0,92 g/cm^3, elasticity, resistance to the double bends, high resistance to acids, alkalis, organic solvents, and non-toxicity) . Graft polymerization allowed to fix covalently on the surface of the fibers polyacrylic chains with cation-exchange functional groups, which served as precursors and stabilizers of colloidal iron hydroxide particles deposited on the surface of the fibers.

Experiments on precipitation of colloidal particles of iron hydroxide on the surface of the fibers was carried out under conditions similar to the synthesis of 2-line ferrihydrite - at a pH of 8-9, followed by low-temperature drying of composite fibers [5,7].

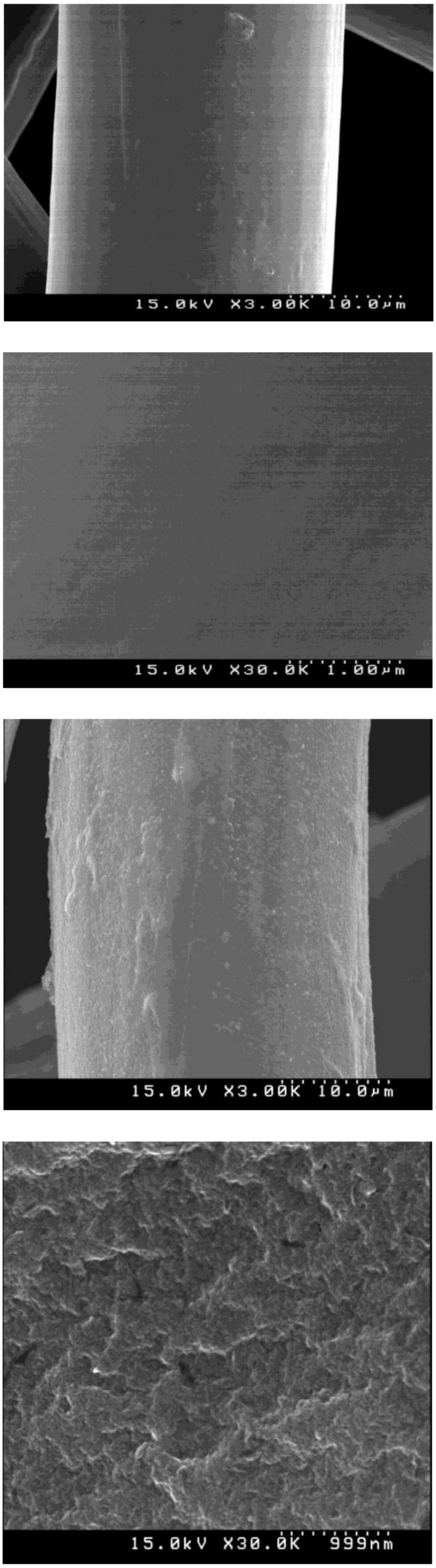
Initial polypropylene fibers and fibers after graft polymerization of acrylic acid had white color and a smooth texture (Fig. 1a, b). After the reaction of the grafted fibers in the Fe (III) salt solution and precipitation of iron oxides hydroxides their color changed to reddish-cinnamon. SEM images of composite fibers (Fig. 1, d) show that the iron hydroxide deposited on the surface of the fibers forms a uniform layer, consisting of a tightly abutting nanoaggregates (30-40 nm). Microanalysis of the layer deposited on the surface of the composite fibers, have shown that it contains the basic elements of iron hydroxide - Fe and O and the absence of elements of a polypropylene fiber matrix - C and H.

Fig. 2 shows the diffractogram of polypropylene (1) and the composite fiber layer deposited iron oxides hydroxides (2). The X-ray spectrum of the starting polypropylene fibers shows peaks at $2\theta = 14°$, 17^o, 18.6^o, 21.5^o, 22^o, which meet the main reflections of α-phase of the polypropylene - (110), (040), (130), (111), (041) , respectively. For synthesis of composite fibers we selected grafted fibers with average values of graft polymerization (150-200%). In [16] it was shown that the degree of crystallinity of the polypropylene matrix decreases with increasing of graft polymerization.

The diffractogram of the composite fibers with the deposited layer of iron oxides hydroxides (2) one can see the appearance of two new broad peaks centered at $2\theta \sim 35^o$ and ~62^o. The position of maxima on the diffractogram and the intensity ratio of these peaks are substantially identical to those published for natural and synthetic samples of 2-line ferrihydrite [5, 7, 15, and 17]. Thus, the obtained results allow us to state that the iron hydroxide deposited on the surface of the composite fibers, is a 2-line ferrihydrite.

We also investigated stability of the composite fibers with the deposited layer of ferrihydrite in aqueous solutions with different pH values. The results presented in Table 1 indicate that the composite fibers are stable in alkaline and acidic media for a long time.

Maximal dissolution of the deposited layer takes place in acidic environment. However, it happened that even at pH 2 after 42 days of exposure in the solution there was less than 2% of the iron deposited on the fibers in the form of ferrihydrite.



a *b* *c* *d*

Fig. 1. SEM - images (a, b) - polypropylene fibers grafted with PAA chains (degree of grafting 171%); (c, d) - the composite fiber with the layer of deposited iron hydroxide

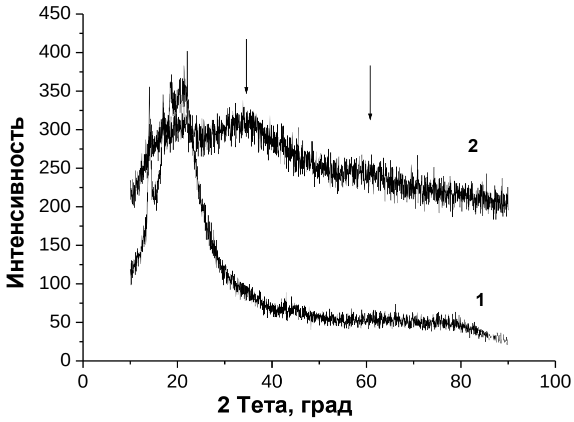


Fig. 2. The diffractograms of the polypropylene (1) and composite fibers with the precipitated iron hydroxide layer (2)

The synthesized fibers were tested as an adsorbent for the separation of strontium from model solutions.

The main functional groups that determine sorption activity of ferrihydrite are amphoteric hydroxyl groups. It is believed that adsorption from solution onto the surface of ferrihydrite is due to exchange of protons. Depending on the pH of the solution hydroxyl groups react either as an acid or a base, which is manifested in a pH-dependent sorption.

We have investigated sorption of Sr from the strontium chloride solutions at different pH values. The results presented in Figure 3 show the pH-dependent sorption - with increasing of pH from 3.0 to 9.2 there is an increase in adsorption of Sr onto composite fibers from 0.23 to 1.13 mmol/g, respectively. These results can be explained by the fact that below the pH of zero charge point (pH_{TH3}) which is for ferrihydrite is close to 8.0 [5], the surface is positively charged. Therefore, at low pH there is virtually no adsorption of positively charged strontium ions on ferrihydrite.

Table 1. Stability of composite fibers with deposited layer of ferrihydrite in aqueous solutions at different pH values

pH	Contact time (day)	Concentration of iron in solution, mg/dm^3
2,8	1	0,9
2,0	42	9,0
5,1	1	0,6
8,3	1	0,3
10,05	1	0,1
10,2	42	0,3

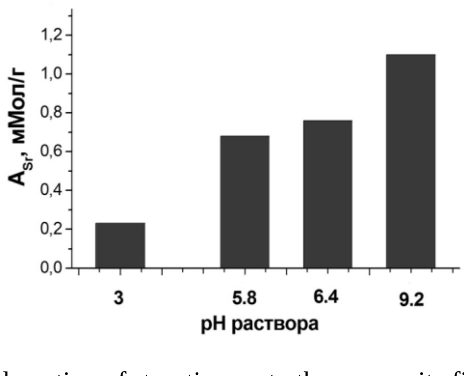


Fig. 3. The dependence of adsorption of strontium onto the composite fibers on the pH of the solution C_0= 8.2 mmol / dm^3; sorption time - 24 hours.

Therefore, at low pH there is virtually no adsorption of positively charged strontium ions on ferrihydrite. With increasing the pH, neutral or negatively charged groups appear on its surface, which leads to a gradual increase of adsorption. Above the pH_{TH3} we observe sharp increase in the adsorption of strontium by the synthesized fibers.

The results of research of the Sr sorption by the composite synthetic fibers with time are shown in Fig. 4. It can be seen that during the first 90 minutes about 90% Sr was adsorbed. However, the sorption process, albeit at a much slower rate continued in the next five days of the study (time to achieve equilibrium) and the maximum adsorption was 1.25 mmol/g.

The two-stage nature of the adsorption kinetics of Sr, Cd, Zn, Ni, Pb, Cu, as well as phosphate and arsenate ions was observed using freshly precipitated and freeze dried “hydrated ferric oxide’ (ferrihydrite) [18]. Due to the fact that nanoparticles of ‘hydrated iron oxide’ tend to form microaggregates with substantial internal porosity, it is believed that the stage of rapid adsorption occurs on aggregates surface and subsequent slow stage is associated with diffusion into the pores (intergranular channels) of the aggregates and adsorption of nanoparticles on the internal surfaces [18, 19]

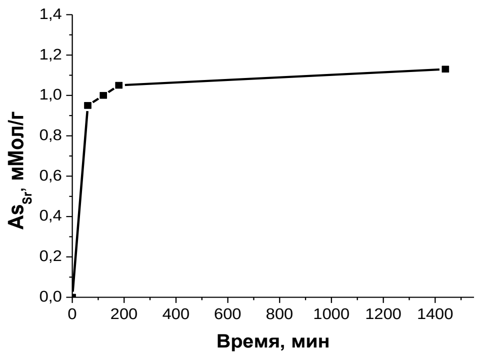


Fig. 4. Dependence of Strontium adsorption onto the composite fibers on time.
C_0= 8.2 mmol / dm^3; pH≈ 9

The results of study of Sr adsorption from alkaline solutions (pH ≈ 8.5 – 9.0) onto the composite fibers have shown that the adsorption of strontium depends on the initial concentration of Sr in the solution (Fig. 5). In the investigated range of initial concentrations (1.3 – 6.7 mmol/dm^3) A_{Sr} linearly increases from 0.2 to 0.97 mmol/g. Thus, with increasing of the initial concentration of the adsorption the efficiency decreases. E.g., when C_0= 1.3 mmol/dm^3 the adsorption efficiency is 99.9%; when C_0= 5.3 mmol/dm^3 - α = 98,1%, while for C_0= 6.7 mmol/dm^3 - α = 96,7%.

Fig. 5. Dependence of the adsorption on the initial concentration of strontium in the solution.
pH 8.5– 9.0; sorption time – 5 days; temperature 23^{0}C; with occasional shaking of the solution.

It was also found that at increase of the concentration of background electrolyte $NaNO_3$ from 0.01 to 0.5 mol/dm^3 (pH≈7.5 – 8.0) strontium adsorption remained virtually unchanged, indicating that chemical interaction between the adsorbent and Sr ions. The results also indicate that sodium ions are not competing with Sr sorption on ferrihydrite and do not affect the sorption process

Competing with sorption of strontium ions are apparently, calcium and magnesium ions, which have similar (geo) chemical properties. Our studies have shown that calcium inhibits strontium sorption (Figure 6): with increase in the calcium content in the solution A_{Sr} is rapidly decreased. With an equal ratio of Sr and Ca ions there is virtually no adsorption of strontium.

Similar results were obtained when studying the adsorption of strontium by natural montmorillonite coated with ferric hydroxide [12].

In [11] it was discussed the complexity of separation of Sr-90 from liquid radioactive waste in the presence of Mg and Ca ions, the concentration of which exceeded the concentration of Sr. Competing Ca and Mg significantly suppressed adsorption of strontium on clay, feldspathoids and silikotitanats.

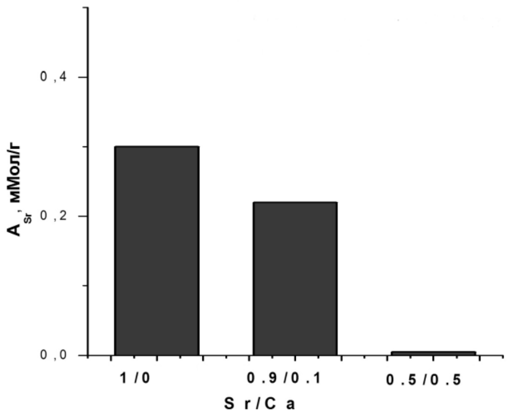


Fig. 6. Effect of calcium ion concentration on the adsorption of Sr in the composite fibers.
pH = 6.8 – 7.0; C_0 = 2.8 mmol/dm^3; sorption time – 50 hours.

However, sand with deposited iron hydroxide has shown the ability to selectively extract strontium in the presence of calcium ions, aluminum and chromium. Despite the fact that the adsorbent based on P-iron oxides hydroxides showed low capacity in relation to Sr, such advantages were marked as low cost, affordable and non-toxic materials for synthesis, the possibility of regeneration and reusing. In the same paper it was studied the effect of ethylenediaminetetraacetic acid (EDTA) on the sorption of Sr and Ca, as well as the stability of the sand-iron oxides hydroxides adsorbent in the presence of EDTA. Researchers have revealed that EDTA binds first of all calcium ions, leaving strontium ions in the solution, which can be considered as promising in the use of adsorbents based on iron hydroxide for the isolation of Sr-90 from LRW by preliminary binding calcium ions with EDTA.

For conducting the pre-concentration process of strontium ions and for repeated use of the composite fibers, the desorption of Sr was investigated using distilled water solutions of 0.5 mol/dm^3 of sodium acetatem and 0.1 mol/dm^3 of hydrochloric acid. As shown in Table 2, neither water nor sodium acetate desorbent are not effective, which indicates the formation of strong chemical bonds between the composite fiber layer of deposited ferrihydrite and strontium ions. An interesting fact which requires additional studies is that the desorption efficiency in sodium acetate solution decreases with increasing of Sr sorption time for the synthesized composite fibers. E.g., from fibers on which Sr sorption occurred within 1 hour in a solution of sodium acetate about 20% was desorbed, and from the fibers with sorption time 163 hours only 2% of adsorbed strontium was desorbed. At desorption with hydrochloric acid complete desorption of adsorbed Sr takes place. In this way, the hydrochloric acid solution allows us to fully desorb the strontium without destroying the adsorbent composite and subsequently use it repeatedly.

Table 2. Desorption Sr with distilled water, solutions of sodium acetate and hydrochloric acid

Composite fibers (Sorption time Sr, h)	Desorbent	Desorbtion, %
4	Distilled water	0
	Sodium acetate, mole /dm^3	
1	0,5	20
2	0,5	17
3	0,5	15
6	0,5	11
163	0,5	2

	HCl, mole/dm^3	
9	0,1	100
163	0,1	100

Conclusions

The obtained results allow us to make the following conclusions:

new composite fibers were synthesized with the layer of deposited ferrihydrite by in situ forming of a layer on the surface of modified polypropylene fibers. X-ray studies have diagnosed this deposited layer as 2-line ferrihydrite. Electron-microscopic investigations have shown that ferrihydrite forms a uniform layer, consisting of a tightly abutting nanoaggregates (30-40 nm) on the fiber surface;

composite fibers showed a high chemical stability in acidic and alkaline solutions;

composite fibers, tested as an adsorbent for strontium ions showed pH-dependent adsorption with the highest values in the alkaline range. The two-stage nature of Sr sorption suggests that the stage of rapid adsorption occurs at the nanoparticles surface, and the subsequent stage is connected with a slow diffusion into the pores (intergranular channels) of aggregates, and adsorption on the inner surfaces of the nanoparticles;

strontium adsorption increases with increasing of initial strontium concentration in the solution, and the presence of calcium ions in the solution suppresses Sr sorption onto the composite absorbent;

neither water nor sodium acetate solution are effective desorbents, thus indicating the formation of strong chemical bonds between the composite fiber and the layer of deposited ferrihydrite and strontium ions. The effectiveness of desorption in the sodium acetate solution is decreased with increasing of Sr sorption time for the synthesized composite fibers. Complete desorption of Sr occurs in 0.1 mol/dm^3 HCl, enabling to regenerate and re-use the adsorbent.

The positive characteristics of the synthesized composite adsorbent are low cost, accessible, and non-toxic materials for synthesis, the possibility of regeneration and reuse, the prospects of use for decontamination of large volumes of contaminated solutions.

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G.A. Sandul

SOME ASPECTS OF ORGANIZATION AND WORK OF SCIENTIFIC AND INDUSTRIAL STAFF OF NUCLEAR POWER ENTERPRISES

In this paper we discuss a number of issues of organization and work of scientific and industrial staff of nuclear power enterprises, by considering them in the following order: background of creation of an enterprise (organization) - construction of its organizational structure

Key words: nuclear power, safety, enterprise, organizational structure

Nuclear power industry, as an area of human activity (production of energy), should be considered in the socio-intellectual space, which coordinates are the following aspects: political, legal, economic, scientific, technological, organizational, environmental, and others.

A necessary condition for evolution of nuclear power is to find a certain balance in the system of the above-mentioned aspects. In principle, it can be applicable to all sectors of economic activity, and nuclear energy is not an exception.

In recent decades, starting from the 80-ies of the last century, in many areas of economic activity it became clear that there was certain imbalance between the individual aspects of the activity, particularly with regard to the issues of organization of the activity itself. Since any activity can be matched with the process (or processes) of the system, we are talking about the imbalance in the organization of process-oriented activities.

Specialists involved in management issues, have suggested that heavy accidents at nuclear power plants in different countries occurred as the consequences of such imbalance. These issues, in varying degrees, have long been discussed at almost all international conferences, “linking” them with the safety issues.

Since the activities in the field of nuclear energy is associated with a high risk to the environment, it must be based on the fundamental principles of safety and establishment of appropriate systems to ensure the specified level of safety at the nuclear power enterprises. The national nuclear legislation of Ukraine and a number of other legal documents are aimed at this, and this is the subject of many scientific papers, for example [1-3].

It is obvious that there are some fundamental “eternal” problems, which, in principle, cannot be solved once and for all. It is such problems that include a strategic safety issue of life in general, as well as a number of its particular components, such as ensuring the safety in the field of nuclear energy.

Safety of nuclear power, in principle, cannot be achieved without solving many fundamental interdisciplinary issues, such as:

• Development of technologies for creating a variety of process controlling systems (design and development of algorithms of actions, and corresponding mathematical apparatus) and the quality of products resulting from these processes;

• Development of decision-making mechanisms;

• Consideration of the role of human factors in ensuring the safety (see. e.g. [4]), etc.

In addition, all these actions should be supported by improvement of nuclear legislation.

Ensuring the safety of nuclear power, even in special cases, requires constant active discussion of its various aspects, in which new ideas are generated and technologies are developed of implementing these ideas into practice. In principle, the discussion of this rather complex and multifaceted subject does not have its final solution. It should be noted that, even an intensive discussion can only solve some aspects of this particular problem for only a limited time of their existence and relevance. But the very fact of “pointing” the problem and formulating it correctly is the first step to its solution.

In the field of nuclear power the safety philosophy should be reflected in the ideology of creation of enterprises (organizations) and in their organizational structures.

In this paper we attempt to briefly (schematically, at conceptual level) consider and organize some “space” to discuss a range of issues, by linking them in the following sequence: “prerequisites for establishment of an Organization - the construction of its organizational structure”, with application of our consideration to nuclear energy enterprises.

The relevance of this topic is largely caused by the “specificity of the moment”: today we can say that society has come to such a turning point when implementation of many innovative research and technical solutions is impossible without innovative solutions in the field of management, business and processes. In other words, it becomes increasingly difficult to ensure the safety of production and technological processes without new approaches to management, including the development of appropriate institutional structures of enterprises.

For clear understanding of the information below let's first consider some elements of conceptual and categorical apparatus.

Main concepts: Safety / hazard (of a system) **-** a philosophical category, a scientific concept that expresses one of the most common properties of a system or phenomenon, given to us in particular applications where it acquires its specific meaning [4, 5].

System - a certain wholeness consisting of interdependent parts, each of which contributes to characteristics of the whole [6].

Organization[1] as a process - a philosophical category, which implies coordination of planned activities of the group of people who, acting on some legal grounds, under the conditions of division of labor, as well as in compliance with the established hierarchy of management, strive to achieve a common goal or purpose of the group.

Organization[2] as a subject of activity - the most common name of any group of people involved in organized activities.

Process[3] (production process) **-** a set of sequential actions to achieve a certain result. [7]

Processes are conditionally divided into administrative (management) processes and the production ones [8].

Management processes are understood here as processes that mainly deal with the information.

Production processes are the processes mainly dealing with actions under objects.

Risk - the combination of the probability and consequences of adverse events.

Strategy - a long-term, consistent, constructive and rational, sustainable plan, which is accompanied by constant analysis and monitoring in the course of its implementation and directed to achieve success as the end result [9]

Most often, the term "strategy" covers the most common management principles.

Personnel - staff workers of the institutions, and enterprises [7].

Team - a group of persons united by a common work, common interests [7], or a group of people united by a common activity, common interests; group of people belonging to the same organizations, enterprises, institutions [10].

Program – a plan of certain activities containing the description and goals, as well as algorithms for solving relevant problems [7].

Plan - pre-planned system of activities, providing the order, sequence and timing of the work [7].

In other words, the **plan** is a list of what needs to be done; the program is a list of what needs to be done + description of how to do it (technology) under availability of the appropriate resources.

Comments: First, you should draft a plan, and then a program.

Some general principles of an Organization and designing of its organizational structure. The environment, in which an organization (enterprise) operates, is determined, for example, by the following coordinates:

• The size of an Organization;

• Type of an organization (institute, factory, shop, etc.);

• The nature of the Organization's activities (educational institution or scientific research institute, food store, etc.);

• Form of ownership of the Organization:

a) state-owned;

b) private;

• Scope of activities;

• The type of activity (Institute for researches in a field of physics or philology, etc.);

• Organization resources.

Defining the metrics of the "space" (or environment) in which any organization operates, is a fairly complex, challenging task and it requires the development of a specific methodology, i.e. it itself is a subject of systematic study of a system. The result will depend on many aspects of the activities of the Organization in dealing with specific problems.

First of all, let's look briefly at some general "structural" diagram of creation of different organizations, reflecting the core "business" principles of modern society. However, the point here is not so much in a

[1] generalized term
[2] generalized term
[3] In principle, it is also a philosophical category, as currently the term can be applied to many systems and phenomena

"modernity" of the society, but in its conservatism, determining appropriate "lag" of implementing the innovations.

Whatever the size of an organization is, and what their activities are aimed at, there are the most common organizational approaches to their creation. Consider this item in more details.

Before starting any activity, taking into account the relevant legal framework, it is necessary to conduct a thorough analysis of, for example, the following subjects:

• Relevance of the activities to the community;
• Needs of the society (or of a part of it) in the supposed products of such activities;
• How long could be the need for this activity (production).
• Short-term or long-term planning activities;
• Competition, etc.
• Alternative activities, etc.

Naturally, the dimensions of the "space" in which the organization operates, determines the choice of strategy to create its overall organizational structure, including particular organizational substructures, for example, safety and security systems (industrial, economic and others.).

In principle, the safety issues are relevant in any activities in any organization, even when operating a small kiosk; however, one should always take into account the shape-and-size issue, for example: NPP – a kiosk.

Following the analysis presented above and in the case of making a positive decision, and within a certain legal framework, it is necessary to consider a strategy of activity: the so-called "own demands" vs. "own capabilities", for example:

• What version of the "future" among the available options we want to choose;
• What we are willing to sacrifice to achieve this goal;
• What indicators should we use for the assessment of whether we are approaching the goal or not, and if "yes", then "how fast", etc.
• What our action in the event of failure should be, etc.

It is obvious that the choice of a clear strategy in conducting any activity - is just a necessary (but not a sufficient!) condition for achieving this goal. Certain other actions are also needed.

To choose the right business strategy it is necessary to go beyond its narrow "craft" framework, which requires a wide summarizing view taking into account the synergy: accounting of mechanisms of interaction of complex systems. In addition, it should be understood that mistakes in the choice of strategy are the most "expensive" ones.

Then, one need to be aware that even the most precise formulation of the strategy, as well as the associated planning, should have a certain dynamics, i.e., ability for changes and improvements.

Mechanisms of strategy selection and planning are closely related to the fundamental issues of forecasting, which in this paper are not the subject of our discussion.

In concentrated form any organization in real-time, can be highly conditionally presented as a time function

$$F(t) = F(R, OS, t) \quad (1)$$

where R - the resources available to the Organization, OS - an organizational structure that is able to effectively use the available resources; t - time of the proposed activities.

In this dependence R and OS are the sums of dimensionless parameters describing the resources and organizational structure, respectively. Presentation of specific performance indicators (what exactly they should be) depends on the specific activity of the Organization. This approach can be very useful in assessing the effectiveness of an Organization.

Presentation of F (t), using of R and OS as arguments is quite conditional, because in some cases the organizational structure (or a part of it) may be of a confidential nature ("trade secret") and then it will be treated as the "resource' of an Organization.

Without going into details, it should be noted: a simple analysis of this function shows that it is not symmetrical. This means that at the availability of resources and a poor organizational structure, it can be improved, but if there are no resources, than not any perfect structure will help.

Unfortunately, as a rule, in the real world we have to perform any activity in the framework of imperfect organizational structure and lack of resources.

To be more precise, the ratio (1), actually, is not a function, but a functional, which could be, for the sake of completeness, presented in the following way:

$$F(t) = F[R(t), OS(R_{const}, t,), t]. \quad (2)$$

Dependence, for example, OS (t) = f [R (t)] indicates that the functional is actually nonlinear: functional arguments are dependent on each other.

The ratio (1) where the arguments are not independent of each other is a function of a linear combination of the arguments that at the macro level within the acceptable simplifications quite well and clearly describes the behavior of the system.

Depending on the task (the “depth” of its consideration) we should consider the ratio of (1) or (2). In this case, we note that in the case of (2) we should have a lot of additional information, and obtaining it is a big problem.

However, in any case, at knowing the scenario (even an approximate one!), this approach allows the development of algorithms of action as a system of corrective and preventive actions. For example, we can make some corrections in the organizational structure, etc.

Then, when you create any type of Organization, some general scenario has to be written to be able to carry out any activity:

• A strategic problem to be solved by the activity and the corresponding strategic goal should be formulated;
• Tactical problems and objectives should be formulated, as well as their solutions;
• Specific problems and ways of their solutions should be formulated and put on the list.

A strategic problem is a kind of problem over which many organizations work in various countries. It is, for example nuclear energy and many other sectors of activity (medicine, industry etc.), the safe management of radioactive waste, etc. The strategic goal, following the above approach is, for example, the protection of human health and the environment against harmful impact of ionizing radiation.

It should be noted, that for any way of solution of a strategic problem (evolutionary or revolutionary way), in most cases, in practice it is not possible to resolve it by one Organization. In this regard, efforts should be focused on the solution (taking part in the solution) of tactical problems (tasks), such as design and construction of storage facilities for radioactive waste, the development of technologies that minimize the production of radioactive waste, and others.

It is possible that the efforts of one Organization can not completely solve even a tactical task; then this tactical task should be differentiated into a number of individual specific tasks which can be implemented (wholly or partially) by a particular Organization.

Thus, even in dealing with the “small-scale” task the solution of strategic problems should be kept in mind. This is the sense of the goal-oriented activities.

Clearly formulated ideas about the problems to be solved by the Organization are the basis for the design of its public and private institutional structures.

The organizational structure of the enterprise which operates in the field of nuclear energy includes, for example, the following components (particular organizational structures):

• Organizational structure of the general management of the Organization, for example, the Board of Directors, the Board of founders, managing direction, Scientific and Engineering Council, etc.;
• Organizational structure of the general safety system, which, in turn, consist of, for example, three sub-systems (systems of the lower level of the Organization) [3]:
a) organizational structure of industrial safety systems: mechanical, electrical, fire, etc.
b) organizational structure of the radiation safety system;
c) organizational structure of the system of physical protection;
• Organizational structure of the quality management system;
• Organizational structure of monitoring and / or control systems, such as radiation monitoring of the environment at NPPs, environmental objects, etc.
• Organizational structures (different) for managing a variety of projects, such as:
a) selection of sites for the construction of some objects, such as radioactive waste storage facilities;
b) designing radioactive waste storage facilities;
c) construction of waste storage facilities, etc.
• Organizational structure for managing the subsidiaries and / or separate subdivisions;
• Organizational structure for external relations:
a) supply;
b) procurement;
c) lobbying, etc.

Particular organizational structures providing effective management of the Organization and its operation can be based on different principles, such as: basic unit - its representatives (resident or non-resident) in other parts of the Organization, etc.

In so doing, the base unit, for example, providing overall safety could have its representatives in other units, where they organize (provide) safety on topical issues for the divisions: general industrial, radiation and others.

It is obvious that all the activities of these particular organizational structures - elements of the overall organizational structure should act as a “single mechanism” and to be process-oriented, i.e., be presented in the form of networks and relationships between them (“exit” of a process should match the “entrance” of the follow-up process, etc.). Thus, the integrated management system is reduced to the process management systems and links between them. All processes should be personalized, and safety must be an “inherent feature” of all processes.

Communication between the processes is a difficult issue, and it is appropriate to discuss it only when dealing with specific processes. In principle, there is a certain mathematical apparatus, based on the study of parametric series, which allows us to differentiate and characterize the individual elements of the processes [11].

Very often, it is difficult for top executives of the Organization, usually focused on the “current situation” management, to predict further development of the entire system of processes, sub-processes, etc. that make up the essence of the activities of the Organization. In this regard, it should be recommended to create a Board of consultants, which, like JASON Society[4], could advise the management on a number of specific issues related to the Organization's activity. For large organizations or their associations there already exist certain scientific and technical centers, but they have a completely different function: if a “Board of consultants” have advice functions only, the scientific and technical centers conduct researches and develop mechanisms for the introduction of innovative solutions.

It should also be noted that it is necessary to articulate and prioritize the objectives of the Organization (their hierarchy) to create the organizational structure of any organization:

- specified quality of the performance of its functions / operations;
- monopolization of certain activities;
- expanding the scope of activities in the specified direction;
- receiving the expected profit in the performance of certain works;
- participation in international projects, etc.

Formulated objectives should also be presented by sub-systems, i.e. each goal should be represented by “hierarchical target steps”. In addition, the classification for the purposes of the interim guidelines should be carried out: the current, short-term, long term, etc.

To implement these goals it is necessary to perform functionally specific actions, i.e., one should perform a number of functions, such as management and production. Depending on the content (forms and scope) of these functions a system of relevant units is developed. This structure is known as the Organization of functional-target matrix structure. It involves a single chain management structure and is, in certain sense, a “unit cell” to create a more complex and diverse (at macro level) organizational structures.

Thus, the functional-target matrix is a basic principle underlying the construction of the organizational structures of the various enterprises.

Organizational structures of various organizations at the macro level, depending on the specifics of the organization (type of activity, geographical location, conditions of performance of activities, educational qualification of employees, etc.) can be presented in various forms, such as brigades (teams) (for adjustment of equipment, service etc.), research laboratories, departments, etc. However, within the teams, laboratories, departments, etc. to effectively carry out the work its organizational structure should be based on a matrix of functional-target structure.

Similarly constructed are military units: The goal is - to win, but to achieve it you need to perform certain functions, which are differentiated, for example, by combat arms, etc., i.e., we have separation of functions.

Functional-target structure of the Organization of the communities being widespread even in the animal world (some are beaters, others are catchers), appeared in the prehistoric times, and there is nothing else neither humanity nor animal world is invented. It should be noted that this is the structural organization implemented in a “healthy family” (targets and functions) as the unit of a society[5].

Creation of effective organizational structures of organizations - the science, which is based on certain laws; a science in which has a lot of “fine details” (horizontal differentiation, vertical differentiation, spatial distribution, integration, formalization, centralization, communication, socio-technical environment, etc.) and “pitfalls” (uncertainty of the surrounding environment, including information).

This science is dynamic and evolving through ‘trial and error’, a science in which, according to Confucius,

[4] The society, consisting of reputable scientists who advise the US government on various issues of public activities

[5] A huge amount of publications is dedicated to this topic, in particular, the work of F. Engels ‘The Origin of Private Property, the Family and the State’

life experience and common sense of its creators – is a ‘lantern that hangs on the back and covers only the distance traveled,’ and each step we make in the dark, into the unknown.

Unfortunately, very often the organizational structure, as an abstract scheme, is something similar to a hieroglyph that is always someone once appends, and as a result, it finally loses all its meaning.

Requirements for the establishment of effective organizational structures of nuclear power plants, which are “completely responsible for safe operation,” are stated, for example, in [12]. According to the aforementioned document, the organizational structure of a nuclear power plant should not only ensure the implementation of the safety policy of the Organization (highest priority), but also to provide all of its management function in view of possible corrective and preventive actions.

We emphasize that the organizational structure of any organization should be well-documented.

In addition, the organizational structure of an enterprise should be designed so that it could provide the formation of traditions and the creation of certain “schools” of leadership on the basic directions of activity of the enterprise. This determines the final result of the Organization's activities and the degree of achievement of its objectives. In the field of nuclear energy the safety should be a muse for all traditions.

In general, almost always the main goal of any organization is to maximize profits. In principle, these profits need not be expressed in solely monetary terms. For example, it can be expressed in the increase of the Organization rating (charity), in the dissemination of a particular ideology (missionary activity), etc.

Organizations performing their activities in the field of nuclear energy, such as nuclear power plants, along with their developed infrastructure (storages of spent nuclear fuel and radioactive waste, etc.), as well as together with satellite Organizations (research centers, engineering and design bureaus and others) due to the extremely high risk of this activity, should, in addition to their basic function - production of electricity, be focused not so much on maximizing the profits, but rather on the priority of safety.

Of course, in any safety system, there is no small detail; however, there are dominant factors there. In this context, the safety system must necessarily be subject to regulation, which should facilitate making more informed decisions.

As we have already noted, the successful work of the Organization is directly dependent on its resources. The economic dimension of the Organization's activities, such as planning, accounting, cost control, etc., is largely determined by its organizational structure.

The issues of interdependence, defined by OS ratio (t) = f [R (t)], should be considered in a separate article.

Conclusion. This paper is an attempt to create some “intellectual territory” to discuss the issues of effective arrangement of activities of enterprises operating in the field of nuclear energy.

Of course, the matters discussed herein, are only a small part of the whole spectrum of possible problems and their possible combinations. Naturally, all these issues are relevant, in fact, for all organizations, no matter what they do. Apparently, a lot of effort is still needed to understand and reflect the relevant documents and actions, the specifics of the enterprise (organization) of the nuclear complex.

First of all, the task is to clearly define what the specifics are, and how it should affect, for example, on creation of organizational structures of such enterprises and their work. Whether we like it or not, we still have to solve this problem.

In principle, organizational structures, rather arbitrarily, can be divided into two categories:

1) “rigid” ones, with a high degree of administration, which are most effective in large-scale projects;

2) “soft” ones, having a low degree of administration, which are most effective in scientific or scientific and engineering teams, where the main role is played by the creative component of the activity.

The complexity of development of organizational structures of the enterprises working in the field of nuclear energy, is that a kind of hybrid form should be used, which also would include both categories for different groups of units.

In addition, successful implementation by the Organization of its activity essentially depends also on two interrelated factors: creation of perfect (optimal) organizational structure, and the system of work arrangement, such as the allocation of rights, duties and responsibilities in a team, as well as relations in a team between its members and managers.

Then, it should be noted that the very fact that the issue of the need to improve the organizational structures of the enterprises has been raised is already encouraging: only by understanding the problem, in principle, it is possible to solve it. Thus, naturally, there may be findings and heuristic solutions, but it is likely to expect here a new unknown combination of the known solutions. For the development of evolutionary processes in this area there is a need of “market of ideas”.

Further, it should be noted that the very fact that the issue of the need to improve organizational structures of enterprises has been raised, is already encouraging: only understanding the problem, in principle, it is possible to solve it. Thus, naturally, there may be findings heuristic solutions, but is likely to be expected here new unknown combination of known solutions. For the development of evolutionary processes in this area we need to organize a "marketplace of ideas".

The study of the scientific literature since 1980, demonstrates that, unfortunately, there is no systematic work on the topic in Ukraine, while in the US, Japan, China, the UK and Germany, much attention is paid to the establishment of efficient organizational structures and related management systems.

As for the experience of other countries, it is not possible now just adapt it to our conditions, due to significant differences in the organization of the society itself (the mentality, performance discipline, etc.), as well as in the organization of enterprises (their organizational structures).

Besides, one should also respond to some of the trends of development of socio-productive relations, “inspired” by the globalization of society, for example, integration of industrial production sphere and the sphere of provision of services, etc.

It is also necessary to understand that we now live in a system of completely unpredictable “post-liberal philosophy”, which replaced liberalism, set in the developed countries after the World War II.

The modern way of development of the society and its values is not quite clear: even now new concepts are entered into use, such as “hybrid warfare”,”homo sapiens” vs. “homo informaticus”, “erotic capital”[6] " as an instrument of self-praise of leaders and its influence on public life, “practical application” as a benefit to society and “commercial use” as an opportunity to earn for a certain number of persons, etc.

However, Ukraine has a huge opportunity to improve and further develop the legal aspects of the activities, including those in the field of nuclear energy, for example, based on the “comparative law”.

Given the above, while developing nuclear power, it is necessary to create specialized programs such as “The use of good practices: learning, adaptation, implementation” on a variety of issues. However, adaptation to more advanced technologies, such as performance management is a complex process, long and painful.

The core around which all activities in the field of nuclear energy[7] have to rotate is the safety, or rather, its numerical expression – “the risk” - a concept that can be represented by the following equation

$$R = W \cdot C \tag{3}$$

where W — the probability of a negative event; C — consequences of this negative event which may be expressed, for example, in terms of money, or it is also important for the nuclear power, in the collective irradiation dose of a nuclear facility personnel and the public (or a part of it), etc.

Despite the negligible probability of severe accidents in nuclear power, their negative consequences could be trans-boundary in nature and cover large areas. For example, the Chernobyl disaster affected many countries of the Eurasian continent, and it is impossible to count how many millions of people will eat contaminated seafood after the disaster at the Fukushima nuclear power plant in Japan.

Developing nuclear power in Ukraine, one must comply with the general scenario of economic growth in Europe – with focus on science and effective use of its achievements in solving specific engineering problems. Underestimating the role of science, we do not solve problems, but only increase their number. At the same time, it should be noted that unfortunately over the past twenty years science in Ukraine gradually loses its prestige and ceases to be an area of state and public interest.

Moreover, taking into consideration the “challenges of modern life”, it should be understood that the political life in Ukraine will be for a long time a zone of conflicts, and society will be in a condition of “milling the wind”. Society will have to learn to live in such a system and, overcoming social pessimism, to ensure the safety of nuclear facilities.

All this is very important also because, according to many analysts [13], it is the first half of the XXI century that has a point of singularity - the point of concentration of crises, such as those associated with the increase in population in the world ($dn / dt \sim n^2$, where n - amount of population), population aging, the crisis of food and drinking water, lack of energy resources, wars, etc. All this requires appropriate activities different in relation to the existing forms of social and cultural activities.

[6] This form of capital existed in ancient Rome and even earlier, but now it has taken a qualitatively different form
[7] Excluding the direct production of electricity.

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V.V. Ryazanov

Institute for Nuclear Researches of the National Academy of Sciences of Ukraine, Kiev

STATISTICS OF HIERARCHICAL SYSTEMS AND PROCESSES IN NUCLEAR REACTORS

Based on the theory of hierarchical structures it is found the correspondence between the dynamics of number of neutrons obtained from the theory of branching processes, neutron numbers of n-th generation, node numbers of the n-th hierarchy level, rate of change of the probability of a chain reaction, type of intensity and power of hierarchical link, degree of criticality of the reactor, and the trajectories of neutrons in the reactor. The connection between the probability of formation of a generation of neutrons and the probability of occurrence of a self-sustaining chain fission reaction is discovered. It is shown that Tsallis and Renyi distributions that describe these processes are connected by relationships of deformed algebra, and under certain conditions may be the escort with respect to each other.

Keywords: percolation hierarchical subordination system, the risk of contagion, Tsallis and Renyi distribution

In [1], strict relations of the percolation theory on Bethe lattice describe the behavior of the neutron multiplication factor. The critical point of the reactor corresponds to the percolation threshold. The behavior of the percolation probability is interpreted as the probability of occurrence of a self-sustaining chain reaction, and the derivatives from this value. A clear manifestation of the complexity and non-equilibrity of chain processes in the nuclear reactor is their hierarchical structure. In this paper, statistics of hierarchical systems is applied to the more detailed study of complex fission chains.

Many networks have a block structure, in the presence of which one can choose a group of nodes which are strongly coupled to each other, but having weak links (or even unlinked) with nodes that do not belong to this group. It is caused by the fact that the phase space of a system in some point far from equilibrium, when ergodicity is lost, is divided into clusters, corresponding to structural levels, hierarchically subordinative to each other. Chain fission in nuclear reactors usually has such behavior. Hierarchically subordinative systems form ultrametric space [2-4, 11]. Its geometric image is Cayley tree (Figure 1).

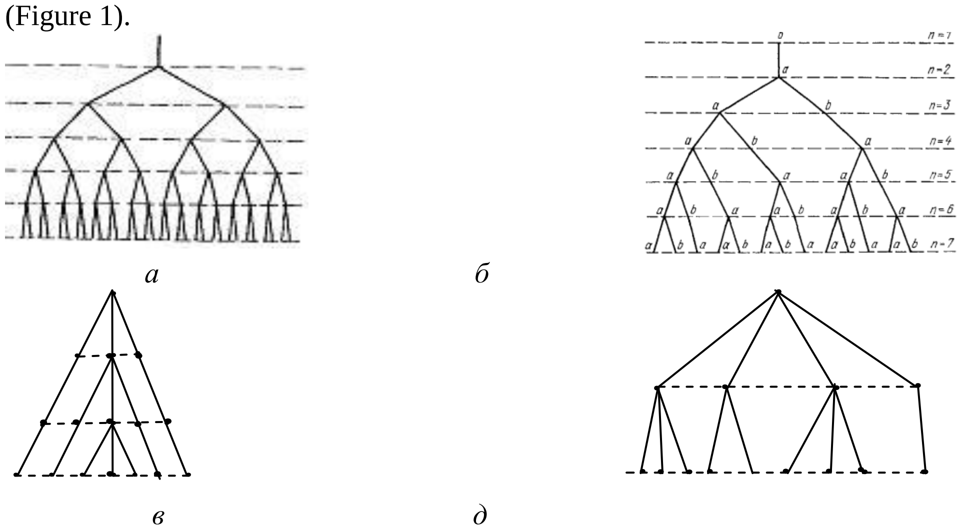

Fig.1. a). The simplest regular Cayley tree with branching s = 2;
b). Irregular Fibonacci tree with variable branching;
c). Degenerate tree with s = 3; d). Irregular tree for n = 2, a = 2

In this paper, some results of the theory of hierarchically subordinated systems beyond the equilibrium are used to describe the nuclear chain reaction in a nuclear reactor. In the second section, an association is found of percolation peculiarities of the behavior of neutron-nuclear processes in reactors, considered in [1], with the intensity of a hierarchical object at the level n, which for a stochastic system is reduced to the probability density, and with the degree of hierarchical links of objects w, corresponding to the nodes of the tree on predetermined level. For different hierarchical trees one can found an explicit form of these values, and set a correspondence with different modes of reactor operation and with trajectories of movement of neutrons in these modes. The third section considers the processes of anomalous diffusion in the ultrametric space. We have found stationary solutions in the form of Tsallis distribution [7], and have showed that the distribution in the particular case were the escort ones with respect to the distributions found in the previous section (which are the Renyi distributions) associated with percolation probabilities and the probabilities of occurrence of the chain fission reaction.

Relationship between the reactor modes and trajectories of neutrons. In [1] for a percolation probability $P(n, c)$ the following recurrence relation was used:

$$P(n+1,c)=c[1-(1-P(n,c))^s], \qquad P(0,c)=c \tag{1}$$

where $c=p=\lambda_f(\lambda_f+\lambda_c)^{-1}$ - the probability of fission of a nuclei by a neutron, the intensity of neutron death (absorption by medium or escape from the system) during the time $\Delta t\to 0$ designated as $\lambda_c\Delta t+0(\Delta t)$, and the intensity of fission of a nuclei by a neutron $\lambda_f\Delta t+0(\Delta t)$ ($\lambda_f=v\Sigma_f$, v — neutrons speed, Σ_f — macroscopic fission cross section), $s=\overline{\nu}$, where $\overline{\nu}$ — expectation value of secondary neutrons per fission. The effective neutron multiplication factor $k_{ef}=p\overline{\nu}$. The probability of $c=p$ from (1) is linked with the important value of the percolation threshold related with the critical point of the reactor. Equations (1), as shown in [1], allow us to determine the critical point.

In [8] the relationship of the form (1) was written in the form:

$$P_{n-1}= P_n+ N_n^{-1}w(P_n), \tag{2}$$

where P_n — intensity of a hierarchical object at the level n, which for a stochastic system can be reduced to the probability density, this is the joint probability of creation of an assembly of hierarchical levels, n-level of hierarchical structure, w — the degree of hierarchical communication of objects, corresponding to nodes of the tree at a given level, N_n – number of nodes at the level n. The degree of hierarchical links of w objects corresponding the tree nodes at a certain level is determined by the number n of steps to a common ancestor that sets the distance in the ultrametric space. The value n in our case corresponds to the number of generations of neutrons in the chain fission reaction. The value w corresponds to kinship in genealogy. Comparing the expressions (1) and (2), we find that

$$w(P_n)=N_n[1-P_n-(1-P_n/c)^{1/s}], \qquad s=\overline{\nu}\ . \tag{3}$$

In [9] and [1] the value $P(n,c)$ denotes the probability of percolation of the root node in the distance n. In [1] this value is mapped to the probability of a self-sustaining chain reaction. The value n, the number of neutron generations in a chain reaction, is proportional to the time with the proportionality coefficient which depends on the reactor type. For thermal neutron reactors the lifetime of one generation of neutrons is equal to 0.1 sec; for the fast reactors the lifetime of one generation of neutrons is by 3-7 orders of magnitude smaller. The value of N_n, - the number of nodes at the level n, corresponding to the number of neutrons of n-th generation for regular tree (Figure 1a) is

$$N_n=\overline{\nu}^{\,n}. \tag{4}$$

The main feature of hierarchical systems is their property of self-similarity [8]. Let's consider the degree of hierarchical relationship $w(P_n)$ (3) for small values of the argument. Expanding in a series the value (3) in the region of small values $P_n\to 0$, we find for $P_{n0}\to 0$, $P_{n0}< P_n$ the maximum term of the expansion, which is equal to

$$w(P_n)=N_nAP_n^{1/s}, \tag{5}$$

where $A=P_{n0}^{(s-1)/s}(1-P_{n0}/c)^{(1/s)-1}/c$, P_{n0} — a fixed value P_n, close to 0.

If we compare (5) with the expression obtained in [8] for the case n>>1, when P_{n-1}~P_n,

$$w(P)=WP^{\beta}, \qquad P\to 0, \tag{6}$$

where $W=w(1)$ — positive constant, $\beta=1-D$, $D\leq 1$ — self-similar fractal dimension of the object of rugged coastline type [10, 11], than we obtain, that $1/s=\beta$, $D=1-1/s=\ln\bar{\nu}/\ln q^{-1}$, $q<1$ — parameter of similarity, and $P_n \sim q^n$, communication function satisfies the homogeneity condition $w(qP)=q^{\beta}w(P)$. We find that $\ln q^{-1}=\ln\bar{\nu}/(1-1/\bar{\nu})\approx 1{,}500803549$, $q=(\bar{\nu})^{-\frac{1}{1-1/\bar{\nu}}}$. From comparison of (5) and (6), because of $W=w(1)=1-c$, we find also that $P_{n_0}=\dfrac{[(1-c)c]^{\frac{\bar{\nu}}{\bar{\nu}-1}}}{(N_{n_0})^{\frac{\bar{\nu}}{\bar{\nu}-1}}+\frac{1}{c}[(1-c)c]^{\frac{\bar{\nu}}{\bar{\nu}-1}}}$. Putting in equations (2), (4) that for arbitrary values of P_n scaling relation $P_n=x_nq^n=x_ns^{-n/D}$, is satisfied we come to the recurrence equation for the function x_n:

$$x_{n-1}=\Phi(x_n), \qquad \Phi(x)=q(x+Wx^{1-D}). \tag{7}$$

Mapping $\Phi(x)$ has two fixed points corresponding to the condition $x=\Phi(x)$: stable $x_s=0$ and critical

$$x_c=(W/(q^{-1}-1))^{1/D}, \qquad q=s^{-1/D}. \tag{8}$$

System behavior is presented by homogeneous functions

$$P_n=x_cs^{-n/D}; \qquad w_n=W^{1/D}(q^{-1}-1)^{-\Delta s-\Delta n}, \tag{9}$$

where $\Delta=(1-D)/D$ — decrement defining the scope of hierarchical communication in ultrametric space [8, 11], which takes into account vertexes of the hierarchical tree.

In [8] the continuum limit $n\to\infty$ is used, finite difference $P_n - P_{n-1}$ is replaced by the derivative dP_n/dn, equation of type (2) is written in a continuous form. Comparison of the exact numerical calculation and solutions of approximate analytical expressions shows their convergence with the growth of n, and the match already at n values of the order of 10-20. Consider the solutions separately for different types of hierarchical trees with different behavior of function N_n, the number of nodes at level n, corresponding to the number of neutrons of n-th generation. For small values of P, in the asymptotic (6) for a regular tree with N_n of the form (4), an explicit solution is obtained for the equation of the form

$$\begin{gathered}P=W^{-1/(1-D)}[(1-u)+ue^{\zeta-\zeta_0}]^{1/D},\\ u=DW^{1/(1-D)}/\ln s,\\ \zeta=(n_0-n)\ln s,\\ \zeta_0=n_0\ln s, \quad n\leq n_0,\\ w=[(1-u)+ue^{\zeta-\zeta_0}]^{\Delta},\\ \zeta\leq\zeta_0, \qquad w(\zeta_0)=1,\end{gathered} \tag{10}$$

where ζ — distance in ultrametric space, $n_0 >> 1$ — total number of hierarchical levels. The argument c of the (1) enters into (10), (13), (14) through w from (3) and (5), and $W=w(1)$. At given configuration of the hierarchical tree an important role plays a fractal dimension D, the value of which determines the strength of a hierarchical relationship $w(\zeta)$. In non-stationary systems, the similarity parameter q varies with time, and $D(q)$ is changed as well. For such complex systems, as the system of breeding neutrons in the reactor, the hierarchical relationship is multifractal by its character [12, 13]. The essential role plays the spectrum of values q on which the communication power of $w_q(\zeta)$ is distributed with a density $\rho(q)$. The full value of the coupling strength is defined by $w(\zeta)=\int_{-\infty}^{\infty}w_q(\zeta)\rho(q)dq$. As a core $w_q(\zeta)$ it is used an expression like (10) with a variable value of the fractal dimension $D(q)$. The behavior of this function for the reactor, obtained by calculation, is shown in Figure 1 in [13] and Figure 1 in [1]. These relations determine only the asymptotic behavior of the hierarchical system in the limit of $1<<\zeta\leq\zeta_0$. The resulting asymptotic behavior is the behavior of a qualitative nature of the hierarchical system. It is necessary to proceed from the finite-difference equations of the form (1), (2) to get accurate solutions, using, as in [1], numerical methods. Distribution of hierarchical levels that was studied in [8] and is reproduced below, shows that the steady-state probability distribution takes the form of Tsallis distribution Note that when using the distribution comprising the lifetime [14] it can be obtained more general distribution, in particular, superstatistics and their generalizations. Tsallis distributions represent only a special case of such superstatistics and their generalizations. The probability of formation of a self-similar network, in our case, the emergence of a self-sustaining chain reaction of nuclear fission - increases monotonically with a decrease in the n, up to a maximum value on the upper level $n = 0$ (initial neutron), corresponding to the entire system. A single initial neutron reactor has the highest probability of a chain reaction, although the real possibility for this is not such significant. Evolution of hierarchical structures is considered in [8] as a diffusion process on randomly branching trees, whose structure is determined by the diversity parameter as a measure of their complexity. The complexity of the system similar to the entropy describes the hierarchical relationship disorder [8]. But entropy characterizes the disorder in the distribution of atoms, while at determining the complexity their role goes to sub-ensembles to which the full statistical ensemble is subdivided.

Relation (10) is recorded for the number of nodes N_n at n level, corresponding to the number of neutrons of n-th generation of the form (4), $N_n=s^n$, where $s=\overline{\nu}$ - index of the tree branching. We now use the above-noted proportion of the number of neutrons generations to time. Let us compare the expression for the number of nodes N_n at the n level with time behavior of neutrons determined, for example, from the theory of branching processes [15, 16]. Expression (4) is written in [8] for the regular tree case, shown in Figure 1a and, since $n \sim t$, it corresponds to the time behavior for the number of neutrons in the exponential form $e^{-\alpha t}$, which is true outside the critical region [15]. In [16] it is shown that in the critical region the dependence is described by power law t^{α}, which coincides with the power approximation

$$N_n=(1+n)^a, \qquad (11)$$

used in [8] in the case of self-similar irregular tree Fig. 1d. This corresponds to a power-law dependence of the type obtained in [1] for numerically critical region boundaries. The behavior inherent in simple statistical systems, is observed in exponent index of hierarchical tree branching above the golden ratio $a_+=(5^{1/2}+1)/2\approx 1{,}61803$, and decrease of complexity with increasing hierarchical relationship dispersion which is characteristic to complex systems is manifested only at weak branching, limited by the interval $1< a <1{,}618$. For a degenerate tree Fig.1c

$$N_n=1+(s-1)n\approx sn\ , \qquad (12)$$

which is close to (11) at a=1 and corresponds to a linear function of time and time behavior at the critical point [15, 16]. It remains an open question whether the movement on the Fibonacci tree has some physical similarity in reactors 1b [8]. Nucleus under fission in this case should be such that the average number of secondary neutrons produced in their division, would equal to the golden section $\tau=(5^{1/2}+1)/2\approx 1{,}61803$

For a degenerate tree with the number of nodes (12), for the behavior at the critical point, we obtain instead of the exponential dependence in (10), a logarithmic dependence of the form

$$\begin{gathered}P=W^{-1/(1-D)}[1-u\ln(1+(s-1)(\zeta_0-\zeta)/\ln s)]^{1/D},\\ u=DW^{1/(1-D)}/(s-1),\\ \zeta=(n_0-n)\ln s,\ \ \zeta_0=n_0\ln s,\\ w=[1-u\ln(1+(s-1)(\zeta_0-\zeta)/\ln s)]^{\Delta}, \qquad \zeta\leq\zeta_0.\end{gathered} \qquad (13)$$

In the intermediate case of the irregular tree with a power-law growth (11) of the number of nodes (and neutrons), the intensity and power of hierarchical links also behave in a power-law way, depending on the distance ζ in ultrametric space proportional to the number of neutrons generations:

$$\begin{gathered}P=W^{-1/(1-D)}[1+u(1-\zeta/\zeta_0)^{-(a-1)}]^{1/D},\\ u=DW^{1/(1-D)}n_0^{-(a-1)}/(a-1),\\ \zeta=(n_0-n)\ln s,\ \ \zeta_0=n_0\ln s,\ \ n\leq n_0,\end{gathered} \qquad (14)$$

$$w=[1+u(1-\zeta/\zeta_0)^{-(a-1)}]^{\Delta}, \quad \zeta\leq \zeta_0.$$

The behavior of the probability of occurrence of a chain reaction is determined by the probabilities *c*, the degree of severity, the proximity to the critical point. Depending on this proximity we can separate three (or more precisely, four) basic modes of behavior: subcritical and supercritical modes (their laws of behavior (4) and (10) differ only in sign), critical mode (11), (14), and a critical point (12), (13). In the traditional theory of nuclear reactors only subcritical and supercritical regimes and the critical point are investigated, but in the general theory of phase transitions the critical region is necessarily presented. This is due to the fact that neutrons do not interact, and for them values of classical critical indices are true (for the self-consistent field) [12]. In the steady operating state of reactors there are a lot of neutrons, and their number can be considered infinitely large. The critical region in this case shrinks to a critical point. Note that the explicit form of expression $w(P_n)$ *(2)* is known, and the equation for *P* in the continuum limit can be solved exactly. But integrals are complicated, and it is difficult to clearly express the P function.

The critical region as such has a complex three-part structure. In [16] three modes of critical behavior of NR were identified, depending on the sign of the control actions and feedback, the boundaries of those modes were found, and it was shown that at the critical point of the area the time behavior is of power-law type. Time is proportional to the number of generations, and this behavior (11) is peculiar to self-similar irregular trees [8]. At the critical point the total number of neutrons is proportional to time (12) corresponding to the degenerate tree. Thus, the trajectories of neutrons vary depending on the probability *c*, and a multiplication factor. In subcritical (and supercritical) region the movement occurs on regular trees, in the critical region - on the self-similar irregular trees, at a critical point - on a degenerate tree. Above the critical point, but in the critical region - again on the self-similar irregular trees. In the supercritical region - again on a regular trees.

The probabilities of formation of hierarchical levels, the distribution on hierarchical levels and on generation of neutrons. Self-similar distribution described by a power law in the form (11)

$$p(k)\propto k^{-\gamma} \tag{15}$$

with power index $\gamma>0$, where *k* — the index of the vertice of the tree, which plays the role of scale in complex networks. Such kind of dependences are widely represented in most systems of different nature. The form of such a dependence is not changed with the variation of scale of the variable k that determines the distribution of the order of vertices of the hierarchical tree of a graph. Indeed, the change of variable k with value k/a, scaled by a positive constant, preserves the form of the distribution (15). Fission chains division lead to a hierarchical structure, which geometric representation is Cayley tree (see. Fig. 1).

In general case, the cluster structure of all levels determines the behavior of hierarchy; however, the self-similarity property allows us to restrict ourselves by specifying the structure of the minimum cluster and finding the number of hierarchical level. The hierarchical tree is a geometric representation of ultrametric space [17], and it is shown in [3] that the description of hierarchical structures is reduced to the consideration of the diffusion process in this space.

Evolution of complex hierarchical systems is anomalous diffusion on hierarchical levels, which leads to the stationary distribution in the form of Tsallis or Renyi distribution. According to [18], let's consider the probability density $p_{\mathrm{u}}=p_{\mathrm{u}}(t)$ of the system distribution on the coordinates in ultrametric space u in the moment of time t. This distribution satisfy the kinetic equation [19, 20]

$$\tau\dot{p}_{\mathrm{u}}=\sum_{\mathrm{u}'}\left(f_{\mathrm{uu}'}p_{\mathrm{u}'}-f_{\mathrm{u}'\mathrm{u}}p_{\mathrm{u}}\right), \tag{16}$$

where the upper point denotes differentiation with respect to time, τ - relaxation time, $f_{\mathrm{uu}'}$ is the frequency of transitions from u' into u. To determine the form of dependency from the ultrametric coordinates let's consider the regular hierarchical tree, which is characterized by a fixed index of branching $s>1$ and a variable number of hierarchical levels $n>>1$. In this case, the ultrametric coordinate u is *n*-digit number in the calculation system with base *s*: $u=u_0u_1...u_m...u_{n-1}u_n$, $u_m=0,1...,s-1$. The intensity of the transitions can be written as a power series $f_{\mathrm{uu}'}=\sum_{m=0}^{n}f(u_m-u_{m'})s^{n-m}$, where the first term ($m=0$) corresponds to the upper level of the hierarchy, which defines the behavior of the entire system - fission chain, while the last term with $m=n$ corresponds to the lowest level corresponding to the smallest clusters, the last branches of the chain. By definition, the distance between the points u and u' is equal to $0 \le l \le n$, if the conditions are met for $m=0,1,\dots,n-(l+1)$ and $u_m \neq u_{m'}$ for $m=n-l$, $n-l+1,\dots,n$ [11]. Thus, at a fixed distance l the first $n-l$ terms of the above series equal to zero by definition, whereas the latter, whose number is equal to l, contain s^{n-m} multiplier whose value when $s>1$ is much smaller than the multiplier s^l, which is the first of the remaining terms. As a result, only the term with $m=n-l$ and $f_{uu'} \sim s^l = s^{n-m}$ is significant in this series. Similarly, we can show that the probability density is estimated as $p_u \sim s^{n-l} = s^m$. For a random tree the branching index s becomes variable, and as a result, the frequency of transitions $f_{uu'} \to f_{n-m}$ and the probability density $p_u \to p_m$ take the form of Mellin transformation [20]

$$f_{n-m} \equiv \int_0^\infty f(s) s^{n-m} \mathrm{d}s, \quad p_m \equiv \int_0^\infty p(s) s^m \mathrm{d}s, \tag{17}$$

where $f(s)$ and $p(s)$ are weighting functions.

Thus, from the total coordinates $u=u_0u_1\dots u_m\dots u_{n-1}u_n$, $u_m=0,1\dots,s-1$ of the ultrametric space we move to the coordinates of the level number, and the number of neutron generations that have been used in the previous section.

As a result, basic kinetic equation for the probability of the formation of the n-th hierarchy level takes the form

$$\tau \dot{p}_n = \sum_{m>n} f_{m-n} p_n - \sum_{m<n} f_{n-m} p_m, \tag{18}$$

where in contrast to the expression (16) representing the continuum ultrametric space discrete representation is used, corresponding to the hierarchical tree of the type shown in Fig. 1. The first term on the right side of (18) takes into account the hierarchical link of the level n and the lower levels $m>n$, the second term with the upper $m<n$. Noteworthy is the fact that the right-hand part of equation (18) has a sign reverse with the one in conventional statistical systems [21]. This is due to the fact that inherent property of the autonomous systems is spontaneous establishing of hierarcy, but not its destruction [3].

Expanding the probability p_m in (18) in a series in powers of $n-m$ difference in the limit $n>>1$, we get

$$\tau \dot{p}_n = -D(\partial^2 p_n / \partial n^2) + D_n p_n, \tag{19}$$

where we have considered the lower moments $\sum_{m<n}(n-m) f_{n-m} = 0$ and $\sum_{m<n}(n-m)^2 f_{n-m} \equiv 2D$; the operator $D_n := \sum_{m>n} f_{m-n} - \sum_{m<n} f_{n-m}$ determines the difference between the intensities of transitions from this level to the lower and upper levels. If there is no hierarchy, so there are and no conditions $m>n$, $m<n$ from (18) and the operator $D_n = 0$. In hierarchical systems, the intensities of transitions depend strongly on whether they are made up and down the hierarchical tree. Using further the assumption about the form of the function D_n

$$D_n = -dqp_n^{q-1} \partial / \partial n, \tag{20}$$

where q, d — positive parameters. The formal basis for the assumption is that up to a factor $-d(q-1)$ the integral $\int_n^{qn} D_n p_n \mathrm{d}n$ reduces to the Jackson derivative

$$D_n p_n^q = \frac{p_{qn}^q - p_n^q}{q-1}, \tag{21}$$

representing the archetype of the self-similar hierarchical systems [4]. As a result, the control equation (19) takes the final form

$$\tau \dot{p}_n = -(\partial / \partial n)\ dp_n^q + D_n (\partial p_n / \partial n)\ . \tag{22}$$

The stationary solution of this equation can be written in the form Tsallis distribution [7]

$$p_n = \left(p_0^{-(q-1)} + \frac{q-1}{\Delta} n \right)^{-1/q-1} \quad p_0 \equiv \left(\frac{2-q}{\Delta} \right)^{\frac{1}{2-q}}, \quad \Delta \equiv D/d. \tag{23}$$

According to (23), with increase in the level number the probability of it formation p_n decreases on a power law from a maximum value p_0 corresponding to the upper level $n = 0$.

With the use of the deformed exponent $\exp_q(x) = 1 + (1-q)x_+^{\frac{1}{1-q}}$, $y_+ \equiv \max(y, 0)$ and effective energy $\varepsilon_n = \left(\frac{2-q}{\Delta} \right)^{\frac{q-1}{2-q}} n$ the probability (23) takes the canonic Tsallis form

$$p_n = p_0 \exp_q \left(-\frac{\varepsilon_n}{\Delta} \right). \tag{24}$$

According to [22], the effective temperature Δ satisfies the standard thermodynamic relations, provided that the distribution on the levels of the hierarchical self-similar set is determined by the probability of escort $\mathrm{P}_l := \frac{p_l^q}{\sum_l p_l^q}$, but not the original p_l. In [23] and [8] it is pointed out that if you put *q*`=1/*q*, then escort Tsallis distribution coincides with the Renyi distribution obtained by the application of the principle of maximum entropy for the Renyi entropy. In [24], it is shown that the Renyi entropy is a negative indicator of the degree of conformal transformation of the information divergence. The effective temperature Δ is related to the probability of a nuclear fission *c* from (1).

The probability of formation of a hierarchical level, and the associated with that level self-similar chain (chain reaction) increases monotonically with decreasing of *n*. Calculation [8] shows that the dispersion of growth $\Delta = D/d$, determined by the ratio of the diffusion coefficient *D* to the energy *d*, significantly expands the spread of stationary probabilities for the hierarchical levels. When $\Delta >> 1$ distribution (23) is slightly different from the exponential distribution at higher levels $n << \Delta^{1/(2-q)}$ but with the increase of *n* the power law tail begins to manifest more clearly.

Comparing the distribution (23) with the distribution (14) we see that one is an escort with respect to another one when *a*=0; these are Tsallis and Renyi distributions. As it was already noted, these distributions would be the same when replacing a physical deformation parameter *Q*=*2-q* by the 2-*q*`, *q*`=1/*q*. In this case the dispersion Δ of the distribution (23) and the parameter *W* in the expression (14) are connected by relation $\Delta = 1/W$. The fractal dimension of the ultrametric space in (14) is expressed in terms of deformation parameter *Q* as follows:

D=*q*`-1=*(Q*-1*)/(*2*-Q).*

In general case $a \neq 0$, and these relations are of particular nature. In [18] it is shown that the probability of the ensemble of hierarchical levels P_n and the probability of creation of each level p_n are related by expressions of so-called deformed (through index *q*) algebra, when

$$\ln_q P_n = \sum_{m=0}^{n} \ln_q p_n\ ,$$

$$\ln_q x = \frac{x^{1-q} - 1}{1-q},\ P_n = p_0 \otimes_q p_q \otimes_q \ldots \otimes_q p_n\ ,$$

$$x \otimes_q y = [x^{1-q} + y^{1-q} - 1]_+^{\frac{1}{1-q}}\ ,$$

$$P_n = \exp_q\left(\frac{\sum_{m=0}^{n} p_m{}^{1-q} - (n+1)}{1-q}\right) = (\sum_{m=0}^{n} p_m{}^{1-q} - n)_+^{\frac{1}{1-q}},$$

$$P_{n-1}^{1-q} - P_n^{1-q} = 1 - p_n{}^{1-q}.$$

Unsteady case is considered in [8] in the self-similar mode when the behavior of the system is determined by the power function $n_c(t)$) of the characteristic scale of the hierarchy (for example, the number of generations in which there is a percolation phase transition, the critical point of the reactor), and the probability distribution is presented by homogeneous function $p_n(t) = n_c{}^{\alpha}(t)\pi(n/n_c)$. Since in our case $n\sim t$, then depending on the type of $p_n(t)$ it is the self-similar mode that is critical.

Conclusion.

The article presents a new approach to the study of complex processes in a nuclear reactor, based on the synergistic methods associated with fractal and percolation ways to describe complex systems, and the theory of hierarchical subordination. New research methods make it possible to detect more detailed aspects of the behavior of reactor systems. Their comparison with the traditional methods of studying neutron-nuclear processes in the reactor will make it possible to find more fine aspects of the behavior of these processes, to consider them in due way, and to improve the safety of reactors.

Thus the complexity of hierarchical tree in [25] was characterized by $s_l=ln(M_l/M_{l-1})$, where M_l - the number of nodes on the level l. These expressions are given in (4), (11), (12). In the reactor, the ratio M_l/M_{l-1} between the number of neutrons of neighboring generations characterizes the neutron multiplication factor. For regular trees (4) $s_l = ln\,\overline{\nu}$. Directly in the critical point $M_l=M_{l-1}$, and $s_l = 0$, which corresponds to one neutron which is born in each generation and the picture of the degenerate hierarchical tree. For degenerate tree where $s = \overline{\nu}$

$$s_l = \ln[1 + \frac{(s-1)}{1+(s-1)(n-1)}] \approx \frac{(s-1)}{1+(s-1)(n-1)}.$$

This value tends to zero $\overline{\nu} = 1$ or when $n \to \infty$. For self-similar trees (11) $s_l = ln(1+1/l)^a \approx a/l$. This value tends to 0 at $l\to\infty$,, which was noted in [1]. In [8] it was demonstrated that a more adequate self-similar characteristic outline of the tree and the neutron multiplication factor corresponding to this value for multiplying the reactor system, is Jackson derivative (21).

We should also note that a powerful method for the study of such complex systems is the information geometry of probability distributions [26, 27]. The above distributions of Renyi and Tsallis can be obtained by applying the principle of maximum entropy for the Renyi entropy $H^{(\alpha)}{}_R(p) = \frac{1}{1-\alpha}\log(\int p^{\alpha}(x)d\mu(x))$ and the Tsallis one, $H^{(\alpha)}{}_T(p) = \frac{1}{1-\alpha}(\int p^{\alpha}(x)d\mu(x) - 1)$. This information entropy corresponds to information deviations (divergence) of Renyi

$D^{(\alpha)}{}_R(p|q) = \frac{1}{\alpha-1}\log(\int q(\frac{p}{q})^{\alpha} d\mu(x))^{1/\alpha}$ and Tsallis of the $D^{(\alpha)}{}_T(p|q) = \frac{1}{1-\alpha}(1 - \int p^{\alpha} q^{1-\alpha} d\mu(x))$ (p and q- distribution of probabilities). To these values the generalized Pythagorean theorem can be applied [26, 27], to which the expressions are bound for the maximum entropy and other important physical results, the use of which is essential for a detailed study of reactor systems.

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V.I. Skalozubov [1], I.L. Kozlov[2], S.V. Klevtsov[3], E.N. Pysmennyi[3]

[1] Institute for NPP Safety Problems, NAS of Ukraine, Kiev
[1]Odessa National Politechnic University, Odessa, Ukraine
[3]NTU "Kiev Polytechnic Institute", Kiev, Ukraine

METHODOLOGICAL BASES FOR IMPLEMENTATION OF PRINCIPLES OF ADEQUACY AND SUFFICIENCY FOR IDENTIFICATION OF WWER SEVERE ACCIDENTS TAKING INTO CONSIDERATION LESSONS LEARNT OF THE FUKUSHIMA ACCIDENT

This article considers methodological bases to implement principles of adequacy and sufficiency of symptom-informed approach for identification of severe accidents taking into account lessons learned of the Fukushima accident. The generalized factors of accident initiating events and the beginning of in-vessel stage of a severe accident are identified by example of accidents with leaks of reactor circuits that are dominant for VVER safety.

The obtained results can be used to develop effective strategies for severe accident management.

Key words: severe accident, initiating event (IE).

The necessary and determining initial stage of modeling, analysis and development of appropriate organizational and technical measures for severe accident management is investigation and identification of the list of initiating events and accident sequences of severe accidents (IE SA and AS SA). Identification of the list of IE SA for nuclear power plants with VVER units is determined by:

- the list of accident initiating events (AIE), which under certain beyond design basis scenarios of accident sequences can cause unacceptable damage to the reactor core;

- the list and the final state of beyond design basis accident sequences with possible failures of critical configurations of systems (CCS) ensuring the provision of safety functions (SF) to prevent the occurrence of severe accidents.

In general case the list of initiating events of severe accidents

$$\Pi(ИСТА) = \sum_i \Pi(ИСА_i) \sum_j АП_{ij} \tag{1}$$

where $\Pi(ИСА_i)$ — the list of primary *i*-th AIE; $АП_{ij}$ — the number of *j*-th beyond design basis accident sequence in the *i*-th group of AEI, leading to unacceptable fuel damage under the same initial conditions of severe accidents development and CCS states, ensuring implementation of the necessary safety functions.

The analysis of this issue in [1-3] in relation to the domestic VVER power units has identified the following major limitations and drawbacks of conventional current approaches.

1. Lack of validity and completeness of all the possible IE of SA. So far in the modeling and analysis of severe accidents on VVER - a large leak of the 1st circuit / blackout / loss of coolant during the final stages with fuel damage are usually considered. This approach requires additional studies of its conservatism, as in other accident IE (e.g. inter-circuit leaks, extreme impacts, etc.) worse conditions can occur for the emergencies and development of severe accidents.

2. Lack of validity of the used approach, which excludes relatively rare events from the list of IE SA. The reasons are as follows: the experience of severe accidents at TMI-2 nuclear power plant (USA), Chernobyl and Fukushima-Daiichi NPPs showed that those initiating events were also low-probable, but nevertheless had occurred, and significantly influenced the ecological safety and the possibility of further operation of nuclear power industry. At the same time insufficiently justified is the approach as such, on exclusion from consideration of relatively low-probable IE SA: exclusion of one IE can slightly (within the errors of evaluation) influence the reduction in the overall safety, but with the “cumulative” effect taking into account, the impact of such events can be significant.

3. Lack of effectiveness of implementation of symptom-oriented approaches in the identification of the AIE, the application of which is most important for the safety of the dominant groups of VVER reactor accidents with depressurization of the primary circuit (leakage of the 1st circuit and inter-circuit leaks). The sets of symptoms that are used for identification of AIE and accident sequences do not correspond with the principle of adequacy and minimum sufficiency, according to which the sets of symptoms at minimal, but sufficient quantity of them must identify each selected group of AIE individually. In particular, the consequence of failure to follow these principles is the fact that different groups of accidents can have the same symptoms, which is inadmissible.

Lessons learned of the Fukushima-Daiichi NPP accident have confirmed the relevance of the above-mentioned problems of identification of IE SA [2, 3].

Under identical initiating events with the complete loss of power supply the steam-gas explosions and destruction of the protective safety barriers occurred at different times (on March 12 at Unit 1, on March 14 at Unit 3 and on March 15 on Unit 2) and in different places (on Units 1 and 3 explosions occurred above the containment , and on Unit 2 - in the area of under-reactor heat exchanger), which is an indirect confirmation of dependence of initiating events and accident sequences from the history of accident processes.

In accordance with the results of the in-depth safety analysis of the VVER-1000 units, performed by the Ukrainian operating organization NNEGC Energoatom, the probability of occurrence of the primary accident initiating event with complete loss of power supply of in-house needs (similar to AIE at Fukushima-Daiichi nuclear power plant) is of the order of 10^{-7} per a reactor-year that is a relatively improbable event. This provision confirms the inadmissibility of exceptions from consideration of the low-probable initiating events of a severe accident.

The lack of sufficiently effective symptom-oriented approach of severe accident management and control was one of the reasons for lack of training the staff of the Fukushima-Daiichi nuclear power plant.

Thus, the issues of identification of initiating events of a severe accident are relevant for nuclear power plants of Ukraine with regard to the lessons learned of the Fukushima-Daiichi nuclear accident.

The main provisions of the proposed approach to identification of initiating events of a severe accident are as follows:

1. Sufficient list of initiating events of a severe accident is determined by the list of all groups of AIE and beyond design accident sequences, which final state is inadmissible damage of nuclear fuel.

2. Each *i*-th group of AIE corresponds to definite set and sequence of indicators $\overline{C_i}$ satisfying the principles of identity and minimum sufficiency.

$$\Pi(ИСА_i) = \Pi_i\left[\bar{C}_i(\Delta t_{mi})\right] \quad (2)$$

where the set and sequence of symptoms of the *i*-th group of AIE

$$\bar{C}_i = \text{col}\ \ C_{1i}(\Delta t_{1i}), C_{2i}(\Delta t_{2i}), \ldots, C_{ni}(\Delta t_{ni})$$

is realized at definite time intervals from the beginning of development of an accident process Δt_{mi}.

As the indicators of AIE we can take deviations from normal operating conditions (neutron, thermal hydraulic and chemical parameters) that determine automatic activation of technological protection / blocking, as well as operator's actions, characterizing the fact and conditions of a particular AIE. The set and sequence $\bar{C}_i$ are determined on the basis of deterministic modeling of accident sequences of each group of $ИСА_i$.

3. The set of beyond design basis accident sequences and indicators of the beginning of severe accidents *CT* is determined by the controlled fact of failure of CCS, ensuring performance of necessary safety functions (SF)

$$АП_{ij} = АП_{ij}\left[CT_{ij}(\text{отказ ККС/ФБ})\right] \quad (3)$$

Lists of safety functions and systems that support their implementation, for a serial VVER-1000 unit are provided in Table 1. CCS ensuring fulfillment of the necessary SF are determined on the basis of results of simulation of the beyond design basis accident sequences.

Thus, a sufficient list of initiating events of severe accident (IE SA) is determined by the formula (1) taking into consideration formula (2) and (3):

$$\Pi(ИСТА) = \sum_i \Pi_i(\bar{C}_i)\sum_j АП_{ij}(CT_{ij}) \quad (4)$$

In-depth analysis of the VVER safety shows that the dominant contributors to overall safety indicators are accident initiating events (AIE) with depressurization of the reactor circuit (leakage from the primary circuit into containment and inter-circuit leaks). For these groups of accidents the following critical configurations of systems CCS providing performance of critical safety functions (CSF) are defined.

1. For the accident initiating event S1 "Big leaks of the primary circuit in the containment boundaries" the critical safety function is "Providing the reserve of coolant" with the critical configurations as follows:

1/2 and 1/2 Hydro Accumulator of Safety Injection System HASI) + 1/3 Low Pressure Safety Injection System

(LPSI) or 2/3 High Pressure Safety Injection System (HPSI).

2. For the accident initiating event S2 ("average-size non-isolated leaks of the primary circuit with equivalent diameter of 50 to 200 mm) the critical safety function is control of reactivity and providing the reserve of coolant at CCS

or protection system (RPS) or 1/3 HPSI + (1/2 + 1/2) HASI + 1/3 LPSI

3. For identification of size ranges (and respective CCS) of small leaks of the primary circuit, not compensated by the feed-and bleed system (IE S3), two substantiating design simulations were made on:

Evaluation of the upper limit of the leak diameter, for which CCS is acceptable

1/3 HPSI + 1/4 SDV-A+ 1/2 Auxiliary MDFWP or 2/3 Accident MDFWP

Evaluation of maximum permissible time for the beginning of cool-down on the secondary circuit with the rate of 60°C/h at limit configuration of systems

1/3 HPSI (TQ14) + 2/4 HASI + 1/3 LPSI + 1/4 SDV-A + 1/2 Auxiliary MDFWP + 1/3 Accident MDFWP (1/3

Table 1. List of the design safety functions of the Safety Injection System (ECCS-1000/B-320)

Code	Safety functions /sub-functions	Required systems and equipment	Operational name
SF-1 Control of reactivity			
A_1	Emergency Reactor Shutdown	Reactor Control and Protection System - Emergency Protection System (RCPS-EPS) (СУЗ — АЗ)	AZ
B_1	Boron injection into the primary circuit	Feed and bleed and boron injection (chemical and volume control) systems	TK + TB10
B_2	Boron injection into the primary circuit	HPSI	TQ13,23,33
B_3	Boron injection into the primary circuit	HPSI	TQ14,24,34
B_4	Boron injection into the primary circuit	HA SI	YT
C	MCP cut-off of accidental loop (uncontrolled steam takeoff)	Main Circulation Pump (MCP)	YD
SF-2 Providing of coolant reserve in the primary circuit			
D_1	Providing of coolant reserve in the primary circuit by seed-and-bleed system	Feed-and bleed and boron control systems	TK + TB10
D_2	Providing of coolant reserve in the primary circuit HPSI	HPSI	TQ13,23,33
D_3	Providing of coolant reserve in the primary circuit HA SI	HA SI	YT
D_4	Providing of coolant reserve in the primary circuit LPSI	LPSI in the mode of operation through the reactor sump	TQ12,22,32
SF-3 Heat removal on the secondary circuit			
E_1	Feeding water to SG	Auxiliary Feedwater System AFWS	RL
		Accident Feedwater System	TX10,20,30
E_2	Maintaining pressure in the secondary circuit	SDV-A	TX
		SDV-C	RC
	Maintaining pressure in the secondary circuit (overpressure protection of the 2-nd circuit)	Safety valve of SG	TX
E_3	Cool-down on the secondary circuit	SDV-A	TX
		SDV-C	RC
SF-4 Heat removal on the primary circuit			
F_1	Cool-down on the primary circuit and residual heat removal	LPSI in the controlled cool-down mode	TQ12,22,32
F_2		LPSI in the mode of operation from HA-210 tank	TQ12,22,32
F_3		HPSI in the mode of operation from HA-210 tank	TQ13,23,33
SF-5 Pressure control on the primary circuit (ФБ-5)			
G_1	Pressure control on the primary circuit	System of dumping pressure of the primary circuit (injection into pressurizer from MCP)	YP
		System of dumping pressure of the primary circuit	TK

		(injection into the pressuriser from the feed and bleed system)	
G_1	Pressure control on the primary circuit	Emergency Gas Removal System	YR
G_2	Primary circuit overpressure protection	Primary Circuit Overpressure Protection Pressuriser System	YP
P_1	SG isolation on steam	Main Steam Isolation Valve (MSIV)	TX
P_2	SG isolation on feedwater	Controllers and gates of the main and auxiliary feedwater	RL + TX
SF-7 Provision of electric power supply			
R	Emergency power supply	Reliable power supply system	DG

NOTES. In the Table and in the text the abbreviations are used as follows: СУЗ — Reactor Control and Protection System – RCPS; АЗ (“AZ”) — Emergency Reactor Protection ERP; САОЗ — Emergency Core Cooling System – ECCS; САОЗ ВД— high pressure safety injection system HPSI; САОЗ НД — low pressure safety injection LPSI; ГЕ — Hydro-Accumulating tank HA; МСР — main circulation pump; SG — steam generator; ВПЭН — Auxiliary Motor Driven Feedwater Pump – MDFWP; АПЭН — Accident Motor Driven Feedwater Pump; БРУ-А — Steam Dump Relief Valve Discharged to Atmosphere – SDV-A; БРУ-К — main condenser steam dump valve SDV-C; ПК — Safety Relief Valve – SRV; КД — pressurizer; БЗОК —main steam isolation valve MSIV.

The key issue at controlling of the beyond design basis accident initiating event S3 is actuation of HPSI system (TQ13) for controlling reactivity and providing reserve of coolant: in the case of response of CCS of HPSI (TQ13) removal of heat on the secondary system is performed in the mode of maintaining constant pressure, and in the case of failure of CCS HPSI (TQ13) — in the mode of cool-down on the secondary system. In the first case CCS of the IE S3 is actually

ERP or ATWS + 1/3 HPSI (TQ13) + 1/2 Auxiliary MDFWP

or 2/3 Accident MDFWP + 1/4 SDV-A or 1/4 SDV-C (P = const).

In the second case (complete failure of HPSI (TQ13) CCS IE S3:

ERP or ATWS + 1/2 Auxiliary MDFWP or 2/3 Accident MDFWP + 1/4 SDV-A or 1/4 SDV-C

(in the cool-down mode) + 2/3 TQ14 + YR + 2/4 HA SI + 1/3 LPSI.

4. To identify the upper limit of small leaks of the primary circuit compensated by the makeup system of pressure channel (accident initiating event S4) and related CCS, in fact only two computational modeling were carried out to evaluate the upper limit of the size of a leak and the maximum allowable start time of cooling down at a rate of 60 °C/h at the secondary circuit.

To control this group of beyond design basis accidents the key issue is triggering of at least one feeding channel of the makeup system. In this case if the boron concentrate pumps TB10 are connected successfully operation ability of the make-up system (pressure channel) is not more than 25 hours. However, for accident initiating events S4 failures can be duplicated by HPSI cooldown pumps (1/3 TQ13 or 2/4 TQ14). In the case of performing the functions “Control of reactivity” and “Ensuring of coolant supply by high-pressure systems” heat removal on the secondary circuit is carried out in a mode of maintaining constant pressure. Otherwise, the reactor transfer to a safe state is carried out in a cooldown mode.

These results of the design simulations determine two main CCS for the accident initiating event S4:

At activation of CCS of high-pressure systems

1/3 Pressure channel + TB10 or 1/3 TQ13 or 2/3 TQ14 + 1/2 Auxiliary MDFWP ВПЭН or 1/3 Accident MDFWP + 1/4 SDV-A or 1/4 SDV-C (in the pressure maintaining mode) + LSS;

At complete failure of CCS of high pressure systems

АЗ or ATWS + 1/2 Auxiliary MDFWP or 2/3 Accident MDFWP + 1/4 SDV-A or 1/4 SDV-C +

+ YR + 2/4 HA SI + 1/3 LP SI (in the cooldown mode) + LSS,

where LSS — localizing safety system.

5. Design substantiation of accident sequences of the small primary-to-secondary circuit leak (accident initiating event T41) and average-size leak (accident initiating event T42) are performed mainly for evaluation of maximum permissible time for the beginning of cooldown through the secondary circuit with the rate of 60 °C/h and for shifting of Low-Pressure Safety Injection System to controlled cooldown at complete opening and locking of SDV-A at the accident SG.

CCS providing Critical Safety Functions for the group of accident initiating events T41:

1/3 Pressure Channel + (TB10) or 1/3 TQ13 or 2/3 TQ14 + AZ or ATWS + 1/2 Auxiliary MDFWP

or 2/3 Accident MDFWP + 1/4 SDV-A or SDV-C (in the cooldown mode) +

+ 1/1 БЗОК + isolation RL + closure SDV-A (after opening) + YR

Or injection into pressurizer from the pressure channel (“TK”) or closure after opening of HPSI +

+ 2/4 HA SI + 1/3 LP SI on controlled cooldown line.

The basis of accident management strategy with a small leak from the primary to secondary circuit are personnel actions on the pressure reduction in the primary and secondary circuits (accident SG) below 70 kg/m^2 to prevent loss of coolant through atmospheric steam discharge valves in the event of failure of its closure, and contamination of the environment.

6. CCS, providing critical safety functions for a group of accident initiating events T42:

AZ or ATWS + 1/3 TQ13 or 2/3 TQ14 + 1/1 MSIV + isolation RL

(on feedwater) + 1/1 SDV-A (closure after opening) +

+ 1/2 Auxiliary MDFWP or 2/3 Accident MDFWP + 1/4 SDV-A or 1/4 SDV-C (in the cooldown mode) + YR or closure of HP SI after opening +

+ 2/4 HA SI + 1/3 LP SI on the controlled cooldown line.

The symptoms of accident initiating events defined on the basis of the presented approach and results of computational modeling are presented in Table 2, and configurations of systems for the beginning of the in-vessel stage of a severe accident – in Table 3. In Table 2 and 3: accident initiating events (AIE) S1, S2, S4 — are big, average and small uncompensated leaks of the primary circuit respectively; accident initiating events T41, T42 — small and average inter-circuit leaks. Sufficient list of initiating events of severe accident (IE SA) is given in Table 4.

Table 2. Generalized indicators of loss of coolant accidents, in accordance with the principles of minimality, sufficiency and adequacy

C	Indicators of accident development	Group of accident initiating events (AIE)					
		S1	S2	S3	S4	T41	T42
C_0	Lowering of the level and pressure in the primary circuit, stable opening of the pressure tube controllers (beginning of an accident initiating event)						
	Formation of ERP automatic tripping signal (“AZ”):						
C_1	Not more than 2s from the beginning of an accident initiating event on setpoints У_1 and/or У_2 and/or У_3						
C_2	2—50 s from the beginning of an accident initiating event (AIE) on setpoints У_1 and/or У_2 and/or У_3						
C_3	More than 50 s from the beginning of an accident initiating event on the setpoints У_1 and/or У_2 and/or У_3 or without automated actuation of AZ						
	Formation of the signal of ECCS actuation:						
C_4	Not more than 10 s from the beginning of the accident initiating event on the setpoints У_2 and У_4						
C_5	10—200 s from the beginning of the accident initiating event on the setpoints У_2 and У_4						
C_6	10—200 s from the beginning of the accident initiating event on setpoint У_2						
C_7	200—4000 s from the beginning of the accident initiating event on setpoints У_2 and У_5 or У_6						
C_8	200—4000 s from the beginning of the accident initiating event on setpoint У_2						
	Formation of the signal on closure of the isolation valve of ECCS:						
C_9	Not more than 1 s from the beginning of the accident initiating event on setpoint У_7						
C_{10}	10—100 s from the beginning of the accident initiating event on the setpointУ_7						
C_{11}	More than 100 s from the beginning of the accident initiating event on the setpointУ_7						
	Formation of the conditions of the actuation of HA SI:						
C_{12}	1—200 s from the beginning of the accident initiating event on the setpointУ_8						
C_{13}	More than 200 s from the beginning of the accident initiating event on the setpointУ_8						
	Formation of the conditions of disconnection of the main circulation pump:						
C_{14}	Not more than 30 s from the beginning of the accident initiating event on the setpointУ_9						
C_{15}	30—300 s from the beginning of the accident initiating event on the setpoint У_9						
C_{16}	More than 300 s from the beginning of the accident initiating event on the						

	setpointУ$_9$						
C_{17}	Increase of pressure and activity under the containment						
C_{18}	Increase of activity on the output of turbine ejectors, in steam lines and blowdown water of SG						

NOTES:

У1 —setpoint of actuation RPS “Pressure above the core less than 148 kgs/cm^2 at temperature in hot legs more than 260 °C and reactor capacity more than 75 % of nominal power“;

У2 — setpoint of actuation of RPS “Decrease of difference of temperature coolant saturation and temperature of coolant in the hot leg of any loop less than 10 °C“;

У3 — setpoint of actuation RA3 “Pressure above the core less than 140 kgs/cm^2 at coolant temperature in hot legs more than 260 °C reactor capacity less than 75 % from the nominal“;

У4 — setpoint of actuation ECCS “Pressure under containment more than 0.3 kgs/cm^2“;

У5 — alarm signal “Pressure under containment more than 0.2 kgs/cm^2 “;

У6 — alarm signal “Pressure under containment more than 0.003 kgs/cm^2 “;

У7 — setpoint of closure of the stop valve of the turbine generator “Lowering of pressure before the main steam gate less than 52 kgs/cm^2 or cutoff of two turbine pumps“;

У8 — setpoint of actuation of HA SI “Pressure in the 1st circuit is lower than 60 kgs/cm^2“;

У9 — setpoint of MCP cutoff “lowering of oil pressure less than 0.6 kgs/cm^2”.

Table 3. Configurations of systems failure for the beginning of the in-vessel stage of a severe accident for the beyond design basis accident with leaked reactor circuit

CT	Configurations of systems failures for the beginning of a severe accident	Group of accident initiating events (AIE)					
		S1	S2	S3	S4	T41	T42
CT1	Failures (2 + 2) of HA SI						
CT2	Failures of all channels of HP SI and LP SI						
CT3	AZ failure ⇒ ATWS						
CT31	Failures of 3 safety valves of the pressurizer + AZ						
CT32	Failures of heat removal on the 2nd circuit: (2 Auxiliary MDFWP +3 Accident MDFWP +4 SDV-A +4 SDV-C)+ AZ						
CT33	Failures on reactivity control of boron solution: (TQ13 + TQ14) + AZ						
CT4	Failures of all HP SI channels						
CT5	Failures of all LP SI channels						
CT6	Failures of heat removal systems on the 2nd circuit: (Auxiliary DFWP + Accident MDFWP + БРУ-А + БРУ-К)						
CT7	Failures of 1st circuit pressure control systems						
CT8	Failures: (TK + TB10) + all channels of HP SI + AZ + ATWS						
CT9	Failures: MSIV + isolation RL						

Table 4. List of identified initiating events of severe accident (IE SA) on VVER-1000/V-320

Identificator of severe accident initiating events (SA IE)	**Conditions of emerging of initiating events of severe accident (IE SA)**	**Indicators of accident initiating events (AIE)**	**Indicators of the beginning of the fuel damage**
ИСТА-S1-CT1	AIE S1 “Big leaks from the 1st circuit into containment” at failure of the passive part of ECCS	C_0, C_1, C_4, C_9, C_{12}, C_{17}	CT1
ИСТА-S1-CT2	AIE S1 “Big leaks from the 1st circuit into containment” at failure of the active part of ECCS	C_0, C_1, C_4, C_9, C_{12}, C_{17}	CT2
ИСТА-S2-CT31	AIE S2 “Average leaks from the 1st circuit into containment” at failure of AZ and the pressure control function	C_0, C_2, C_5, C_9, C_{13}, C_{17}	CT31
ИСТА-S2-CT32	AIE S2 “Average leaks from the 1st circuit into the containment” C of heat removal on 2nd circuit	C_0, C_2, C_5, C_9, C_{13}, C_{17}	CT32
ИСТА-S2-CT33	AIE S2 “Average leaks from the 1st circuit into the containment” at failure of AZ and the function of reactivity control	C_0, C_2, C_5, C_9, C_{13}, C_{17}	CT33
ИСТА-S2-CT2	AIE S2 “Average leaks from the 1st circuit into the containment” at failure of HPSI or active part of ECCS	C_0, C_2, C_5, C_9, C_{13}, C_{17}	CT4 or CT2
ИСТА-S3-CT31	AIE S3 “Small leaks, uncompensated by pressure channel, from the 1st circuit into containment” at failure of AZ and the function of pressure control.	C_0, C_2, C_7, C_9, C_{13}, C_{17}	CT31
ИСТА-S3-CT32	AIE S3 “Small leaks uncompensated by pressure	C_0, C_2, C_7, C_9,	CT32

	channel, from the 1st circuit into containment" at failure of AZ and function of heat removal o the 2nd circuit.	C_{13}, C_{17}	
ИСТА-S3-СТ33	AIE S3 "Small leaks uncompensated by pressure channel, from the 1st circuit into containment" at failure of AZ and the function of reactivity control	C_0, C_2, C_7, C_9, C_{13}, C_{17}	CT33
ИСТА-S3-СТ5	AIE S3 "Small leaks uncompensated by pressure channel, from the 1st circuit into contain" at failure of LP SI or active part of ECCS	C_0, C_2, C_7, C_9, C_{13}, C_{17}	CT5 or CT2
ИСТА-S3-СТ6	AIE S3 "Small leaks uncompensated by pressure channel from the primary circuit to containment" at failure of the function of heat removal on 2nd circuit	C_0, C_2, C_7, C_9, C_{13}, C_{17}	CT6
ИСТА-S4-СТ8	AIE S4 " Small leaks compensated by pressure channel from the primary circuit to containment" at failure of "TK", HPSI, AZ and the functions supporting ATWS	C_0, C_3, C_7, C_{11}, C_{13}, C_{17}	CT8
ИСТА-S4-СТ5	AIE S4 " Small leaks compensated by pressure channel from the 1st circuit into containment " at failure of LP SI or active part of ECCS	C_0, C_3, C_7, C_{11}, C_{13}, C_{17}	CT5 or CT2
ИСТА-T41-СТ8	AIE T41 "Small leaks from the primary to secondary circuit" at failure of pressure channel, HP SI, AZ and functions supporting ATWS	C_0, C_3, C_8, C_{11}, C_{13}, C_{18}	CT8
ИСТА-T41-СТ5	AIE T41 "Small leaks from the primary to secondary circuit" at failure of LP SI or active part of ECCS	C_0, C_3, C_8, C_{11}, C_{13}, C_{18}	CT5 or CT2
ИСТА-T42-СТ31	AIE T42 "Average leaks from the primary to secondary circuit" at failure of AZ and the pressure control function	C_0, C_2, C_6, C_{10}, C_{13}, C_{18}	CT31
ИСТА-T42-СТ32	AIE T42 "Average leaks from the primary to secondary circuit" at failure of AZ and function of heat removal on the 2nd circuit	C_0, C_2, C_6, C_{10}, C_{13}, C_{18}	CT32
ИСТА-T42-СТ33	AIE T42 "Average leaks from the primary to secondary circuit" at failure of AZ and function of reactivity control	C_0, C_2, C_6, C_{10}, C_{13}, C_{18}	CT33
ИСТА-T42-СТ4	AIE T42 "Average leaks from the primary to secondary circuit" at failure of HPSI or active part of ECCS	C_0, C_2, C_6, C_{10}, C_{13}, C_{18}	CT4 or CT2
ИСТА-T42-СТ6	AIE T42 "Average leaks from the primary to secondary circuit" at failure of the function of heat removal on 2nd circuit	C_0, C_2, C_6, C_{10}, C_{13}, C_{18}	CT6
ИСТА-T42-СТ7	AIE T42 "Average leaks from the primary to secondary circuit" at failure of the function of pressure control of the 1st circuit	C_0, C_2, C_6, C_{10}, C_{13}, C_{18}	CT8

Conclusions

1. We considered methodical bases of realization of the principles of adequacy and sufficiency of symptom-oriented approach of identification of severe accidents, taking into account lessons of the Fukushima accident.
2. With the use of examples which are dominant for VVER accidents with leaks of reactor circuit we identified generalized indicators of accident initiating events and the beginning of the stage of a severe accident.
3. These results can be used in the development of effective strategies for managing severe accidents.

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A.D. Berezovskiy[1], V.N. Vashchenko[2], T.V. Gablaya[3], I.L. Kozlov[1], S.I. Kosenko[1],Zh.I. Patlashenko[2], V.I. Skalozubov[3]

[1]*Odessa National Politechnic University, Odessa, Ukraine*
[2]*State Ecological Academy of Post-graduate Education and Management, Kiev*
[3]*Institute for NPP Safety Problems, NAS of Ukraine, Kiev*

ADAPTATION OF ACCIDENT MODELLING OF "NON-DESIGN" WWER NUCLEAR FUEL BASED ON CRITERION METHOD

The paper considers problems of adaptation of results of accident modeling and safety analysis for VVERs when using the "non-design" nuclear fuel. The obtained results of the analysis allow us to draw up the conclusions that permissible concentration of plutonium should be not higher than 8% in the range of relatively low temperatures of nuclear fuel that is characteristic for the shutdown reactor and cooling pools of the spent nuclear fuel, and not higher than 3% for higher temperatures. When concentration of plutonium is higher, the additional analysis of nuclear safety concerning the possibility of use of MOX-fuel is required.

Based on the presented method, the conditions of applicability of 'non-design' VVER nuclear fuel with the increased heat conductivity are determined.

Key words: accident modeling, water-moderated power reactor, VVER, non-design nuclear fuel, spent nuclear fuel (SNF).

Purposeful or forced use of "non-design" nuclear fuel causes in general the need to address the following key issues in analyzing the safety of reactor units (RU) and the SNF cooling ponds:

1. Revaluation of safety limits or safety criteria taking into account the differences between physical, chemical and / or structural and technical parameters of 'non-design' nuclear fuel and the design fuel.

2. Simulation of accidents at reactors and spent fuel cooling pools with the 'non-design' nuclear fuel for all possible initial conditions and events.

Now the main established regulatory limit for nuclear safety of VVER units is the allowable temperature of the zirconium cladding (1200°C), at which the irreversible destruction of the cladding begins (as one of the protective safety barriers); and in the simulation of accidents the safety limit for the maximum permissible local temperature of the fuel matrix (about 2800°C) is not actually analyzed. This approach is based on the assumption that the relationship between the temperatures of the fuel rod cladding $T_{об}$ and nuclear fuel $T_{я.т.}$ are determined only by thermal properties of nuclear fuel, and unacceptable destruction of the cladding in the development of any accidents occurs before the beginning of the destruction (melting) of the fuel matrix. The latest developments [1, 2] revealed that the limit on the maximum permissible temperature $T_{об}$ is generally insufficient for the analysis of nuclear safety: because of the relatively low thermal conductivity of the ceramic oxide uranium fuel (UO_2-fuel) of VVER design, significant radial temperature gradients (over 106°C/m) emerge in the fuel matrix, and large central part of the fuel matrix (approx. 50%) will be melted before the cladding reaches the temperature of 1200° C.

Thus, in the simulation of accidents and analysis of the nuclear safety of reactors and spent fuel pools with the 'non-design' nuclear fuel one should also take into account the limit of nuclear safety at the local maximum temperature of the fuel matrix.

It also seems important to determine acceptable adaptation conditions for reactor units and fuel pools with the 'non-design' nuclear fuel of the well-known results of accident simulation and safety analysis of RU with the design nuclear fuel. A promising approach for a-priori determination of acceptable conditions of adaptation of 'non-design' nuclear fuel can be the criterial method of safety analysis, according to which we write safety limits of the fuel element in the criterial form [2]:

$$\tilde{T}_{я.т.} = \frac{T_{я.т.}}{T_{пл}} = f\,(K_\lambda, K_N) < 1$$

$$\tilde{T}_{об} = \frac{T_{об}}{T_{пл}}\ \tilde{T}_{я.т.} - \frac{\mathrm{Nu}}{\mathrm{Nu}+1}\,\tilde{T}_{тн} < \frac{T_{o.Zr}}{T_{пл}}\,,$$

where $T_{я.т.}$ — temperature of fuel in the central part of the fuel matrix, $T_{пл}$ — nuclear fuel melting temperature, $T_{об}$ — temperature of the fuel cladding, $T_{o.Zr}$ — maximum permissible temperature of the fuel clading, K_λ —dimensionless criterion of reduced heat flow from a fuel element, K_N — dimensionless criterion of reduced power of heat releases of the fuel, Nu — Nussolt criterion, $\tilde{T}_{тн}$ — coolant temperature.

Analysis of the safety-critical criteria K_λ, K_N, Nu included in these relationships allows us to formulate the following conservative adaptation conditions for reactor installations and cooling pools with the "non-design" nuclear fuel:

according to the criterion of the internal heat transfer - $K_{A\lambda}$

$$K_{A\lambda} = \frac{K_\lambda(\text{НЯТ})}{K_\lambda(\text{ПЯТ})} \geq 1 \qquad (1)$$

according to the criterion of the heat release capacity — K_{AN}

$$K_{AN} = \frac{K_N(\text{НЯТ})}{K_N(\text{ПЯТ})} \leq 1 \qquad (2)$$

according to the criterion of the external heat transfer

$$\text{Nu}(\text{НЯТ}) \geq 1 \qquad (3)$$

where НЯТ, ПЯТ — "non-design" and design nuclear fuel, respectively.

If these conditions are satisfied for the full range of possible temperature changes of nuclear fuel, than adaptation for reactor installations and cooling pools with the "non-design" nuclear fuel of the results of safety analysis made for the design nuclear fuel is justified. In the event that at least one criterial conditions is not satisfied, than there is a need for additional simulation of accidents and safety analysis for reactors and cooling pools with the "non-design" nuclear fuel.

Consider the application of the presented adaptation criteria for two characteristic examples:

- "non-design" nuclear fuel with high thermal conductivity of the fuel matrix of the fuel element (purposeful modernization);
- forced replacement of the VVER design nuclear fuel with the MOX fuel

The analysis of adaptation criteria in the first example of modernization of the nuclear fuel in order to improve thermal conductivity of the fuel matrix of a fuel rod allows us to conclude the following:

1. Adaptation by K_λ and K_N criteria are satisfied at slight change in the neutron-physical properties of "non-design" nuclear fuel in relation to the VVER design fuel (formally: $Q(\text{НЯТ})/Q(\text{ПЯТ}) \approx 1$) and at necessary conditions of "external" heat exchange (Nu >> 1).

2. From the minimum value Nu = 1 it follows the restriction on the "upper" limit of thermal conductivity values

$$\lambda_{я.т.} \leq \alpha_{min}\, r \qquad (4)$$

where α_{min} — the minimum possible value of the heat transfer coefficient on the surface of the fuel rod, r - radius of the fuel rod

At higher values of $\lambda_{я.т}$ under conditions of heat transfer with with α_{min} $T_{об} \approx T_{я.т}$ and the safety limit for allowable temperature cladding can be exceeded (or reached more quickly) during an accident in relation to the same conditions with the VVER design nuclear fuel.

When replacing the VVER design UO_2 fuel with the "non-design" MOX fuel (second example), critical adaptation condition is K_{AN} criterion that actually reflects the power ratio of internal heat generation Q of "non-design" and design nuclear fuel. Power of internal heat release of nuclear fuel is determined by conditions of the occurring neutron-physical processes, and depends on the neutron flux density of j-th nuclide in the composition of nuclear fuel Φ_j,, capture cross section σ_j and concentration of j-th nuclide γ_j. MOX fuel is characterized by a high plutonium content in relation to the VVER design UO_2-fuel (less than 1% Pu). Neutron-physical characteristics of plutonium and uranium can be essentially different in certain ranges of nuclear fuel temperatures. For example, in [3] it was found that the temperature dependence of the capture cross section was significantly different both qualitatively and quantitatively in the nuclear fuel temperatures over 500°C

Fig. 1 shows the results of the computation analysis of the temperature dependence of K_{AN} adjustment criteria for "non-design" MOX fuel at different concentrations of plutonium γ(Pu), taking into account data on the temperature dependence of the capture cross section for uranium and plutonium. To simplify the analysis, neutron-physical characteristics of the fuel were modeled in quasi-stationary approximation, and the main characteristics of the discrepancies of the "non-design" and design fuel were determined by the cross section capture parameters σ. Also we conservatively neglected the production of Pu in the design UO_2 fuel.

These analysis results suggest that in relatively low fuel temperatures, typical for the shutdown reactor and fuel pools, the allowable concentration of plutonium is not higher than 8%, and for higher temperatures it is up to 3%.

In adapting the "non-design" nuclear fuel one should consider a significant relationship of the determining neutron and thermal parameters from the fuel temperature. Under certain circumstances and conditions of nuclear fuel such substantial dependence may cause aperiodic (spontaneous change of state) or periodic (oscillatory process) instability. According to the general theory of instability, any system can be subject to random (fluctuating) disturbances (effects and / or changes in defining the parameters of the system state). Depending on the current state of the system, these perturbations in time can either be "attenuated" (the system is stable), or lead to a spontaneous change in the critical parameters (system is aperiodically instable) or oscillatory processes (periodic instability). Energy "source" of such unstable processes is convertion of internal energy of the system.

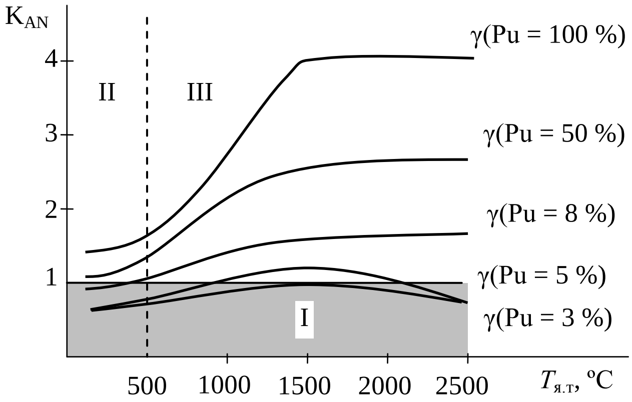

Fig. 1. The limits of adaptation of nuclear safety of MOX-fuel on the plutonium concentrations γ(Pu):
I - Region of permissible adaptation of MOX fuel;
II - Nuclear fuel temperature ranges in the shutdown reactor and / or fuel pools;
III - Nuclear fuel temperature ranges in routine and emergency modes of reactor operation

The main parameter determining the nuclear fuel behaviour is its temperature $T_{я.т}$. Fluctuational perturbation of fuel temperature $\delta T_{я.т}$ depending on the current state can lead to thermal instability of aperiodic or periodic nature under certain conditions.

The negative consequences of nuclear fuel thermal instability for the nuclear safety can be:

- exceeding the safety limits for nuclear fuel cladding temperature;
- instability of neutron-physical processes, which under certain conditions results in loss of control of controlled nuclear reactor power (unregulated "acceleration" or shutdown);
- spontaneous nuclear fuel temperature rise in the shutdown reactor or spent fuel storage pools, etc.

Let's consider the conditions for origination of thermal instability of the nuclear fuel in the linear approximation ($\delta T_{я.т} << T_{я.т}$). The equation of the heat balance of nuclear fuel in the fuel temperature in case of fluctuation disturbances $\delta T_{я.т}$ has the form

$$\frac{\mathrm{d}\delta\tilde{\mathrm{T}}_{я.т.}}{\mathrm{d}t} = \mathrm{K_H}\,\delta\tilde{\mathrm{T}}_{я.т.} \qquad (5)$$

where

$$\mathrm{K_H} = \frac{\mathrm{dK_N}}{\mathrm{d}\tilde{\mathrm{T}}_{я.т.}} - \mathrm{K}_\lambda - \frac{\mathrm{dK}_\lambda}{\mathrm{d}\lambda_{я.т.}}\frac{\mathrm{d}\lambda_{я.т.}}{\mathrm{d}\tilde{\mathrm{T}}_{я.т.}} \qquad (6)$$

Solution (5) has the form

$$\delta\tilde{T}_{я.т.} \approx \exp(K_H t) \quad (7)$$

where t — current time.

It follows from the above, that when K_H <0 the temperature perturbations will “fade” over time (the system is stable), while at K_H > 0 - (aperiodically unstable system) they will grow spontaneously.

Thus, the stability criterion of thermal processes in nuclear fuel

$$K_H < 0 \quad (8)$$

The interpretation of the mechanism of nuclear fuel thermal instability is as follows:

Fluctuation of temperature increase in fuel $\delta T_{я.т}$ determines corresponding increase in power of internal heat release $\delta Q_{я.т}$ and further nuclear fuel temperature rise, with other conditions remaining the same. However, the increase of $T_{я.т}$ determines also the increase in the temperature difference between the nuclear fuel and the outside ambient, increase in the density of heat removal from nuclear fuel $q_{я.т}$ and corresponding decrease in $T_{я.т}$. If the impact of the effect of $Q_{я.т}$ growth under the influence of the disturbance $\delta T_{я.т}$ prevails over the effect of increasing the density of the heat removal from the fuel, the thermal instability of the nuclear fuel appears. Otherwise the growth of $T_{я.т}$ which is determined by increase of power of internal heat releases of nuclear fuel, is “compensated” by increase in heat removal from nuclear fuel $q_{я.т}$ - the system remains stable with respect to perturbations of fluctuation $\delta T_{я.т}$

Conclusions

The analysis of adaptation of the accident simulations of “non-design” nuclear fuel for VVER reactors and fuel pools based on the criteria of internal and external heat transfer, heat generation and thermal stability leads to the following main conclusions:

1. Purposeful increase of thermal conductivity of “non-design” nuclear fuel contributes to increase of nuclear safety in relation to the design ceramic UO_2-fuel, and adaptation of the accident simulations is quite justified.

However, the “upper” values of thermal conductivity of the fuel matrix are limited by the conditions of the minimum intensity of the external heat transfer, in which the fuel rod cladding temperature reaches the fuel temperature. In this case, the safety limit for allowable temperature of the cladding can be exceeded or reached for the “non-design” nuclear fuel earlier in the accidental conditions than for the design nuclear fuel.

2. Adaptation of the accident simulation of the “non-design” MOX fuel for VVER in the operational and accident conditions (fuel temperature is over 500°C) for plutonium-239 concentrations greater than 3% is unjustified.

3. Adaptation of the accident simulation of “non-design” MOX fuel for VVER at temperatures less than 500°C, typical for the conditions of a shutdown reactor or spent fuel pool, is justified at concentrations of plutonium-239 less than 8%.

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V.P. Kravchenko, V.A Dubrovsky, E.V Kravchenko

Odessa National Polytechnic University, Odessa, Ukraine

TECHNICAL AND ECONOMIC EVALUATION OF POSSIBILITY OF USING SOLAR POWER STATIONS AS RELIABLE POWER SOURCE AT NUCLEAR POWER PLANT BLACKOUT

As one of the three channels of reliable power supply of vital services of a reactor building it is suggested to use semiconductor solar power (SSP) modules with batteries instead of a diesel generator. The necessary amount of energy for residual heat removal from a nuclear reactor for three days, the area of solar modules, and the capacity of batteries were determined. The calculations were made of technical and economic parameters of SSP modules. The possibility was considered of reducing the pumps capacity with the decrease of residual heat energy. The cost of electricity was calculated as well.

Key words: residual heat, semiconductor solar power, pumps capacity.

Lessons learned of the accident at the Fukushima-1 nuclear power plant (NPP) in March 2011 showed the need for more reliable power supply of equipment of the reactor building at de-energizing (blackout) of a nuclear power unit [1]. The cause of the accident was the lack of power supply of the vital services as a result of damage of the NPP diesel generators (DG) by tsunami. Based on the analysis of the situation at Ukrainian NPPs, it can be stated that lack of autonomous power supply at NPP blackout will lead to similar consequences. [2]

Analysis of the problem status. According to [3], an integral part of the emergency power supply for the safety systems is a standby diesel generator (in transliteration - RDES). It is designed for use as a stand-alone emergency and back-up power supply of vital services of VVER-1000 nuclear power plants. Diesel generators (DG) in reliable power supply systems do not work in normal modes of nuclear power plant operation and cannot be used for any purpose other than the emergency power supply. There are separate independent diesel power stations for each safety system: one NPP unit has three automated diesel generators of SDA-5600 type, designed to work without permanent presence of staff [4]. Diesel generators capacity is chosen on the basis of providing each safety channel with reliable power required for operation of mechanisms involved in the emergency cooling of the reactor in any kind of accident. Total power of services, which should be provided by RDES is equals to 5107 kW, hence the rated power of a DG is 5.6 MW

If we go further in the design requirements, we should mention the need for different principles of operation of each of the safety channels.

In 1986, the experimental 5 MW Solar Power Plant (SPP) of tower type was built in Shchelkino town just as a backup power source for the Crimean nuclear power plant planned there. During its operation the staff encountered many difficulties; in particular, the positioning system of reflectors virtually completely (95%) consumed the power generated by the Plant. Also, there were problems with cleaning of mirrors. Soon the plant ceased to exist [5]. In this paper, we propose to use a semiconductor solar power station (SSPS) with batteries as a reliable source of power supply of a safety channel. The main element of such station is a panel of photovoltaic (PV) cells (solar cells). At first sight, such a proposal should have an advantage: whereas RDES is idle during the entire life of the unit, SSPP will generate electricity, and thus will pay for itself and increase the probabilistic indicators of safety. Cost of maintenance and repair of a worked-out SDA-5600 is comparable to the purchase of new equipment. It should be noted that the equipment in the standby mode is also subject to maintenance and repair.

Procured diesel fuel should be stored for a certain time, and then disposed and replaced with a new one, which also complicates operation and raises its price.

Emergency cooling of the reactor also implies the use of jet pumps [6], fire engine pumps, connection of power units with special cable connectors and other methods [7].

The aim of this work is to determine technical and economic characteristics of the respective solar power station, and the feasibility of further elaboration of the proposed method of reliable power supply.

Determination of the required power of SSPS. The result of the analysis of occurrence of an accident similar to the one at Fukushima NPP (with the situation of failure of a standby diesel generator) for RDES of VVER was the conclusion that in case of emergency, leading to the complete NPP blackout and inoperable RDES, the reactor units transition to a safe mode is impossible. The used safety systems necessary for liquidation of emergency situations and design basis accidents and mitigation of their consequences to prevent escalation into beyond design basis accidents, will not perform their functions due to lack of power. In this regard, the required minimum set of system that require power supply during blackout has been determined as follows:

TX system (emergency supply of feedwater to steam generators) - 800 kW;

TQ system (emergency supply of boron in the reactor core) - 800 kW;

Other systems involved in the residual heat (RH) removal from coolant - 150 kW.

With accounting of the connected loads the mobile power diesel generating station capacity should be no less than 2 MW.

Thus, the minimum required power capacity of autonomous energy source, in our case a SSPS, is 2 MW.

Determination of characteristics of the photovoltaic cells. Solar modules are classified according to the peak power in watts - watt-peak (Wp). A watt-peak is a technical characteristics, which points out, what the solar station capacity is under certain conditions, i.e. when the solar radiation of 1 kW/m^2 falls on the element at a temperature of 25^{o}C. Such intensity is achieved, if the sun is at its zenith under good weather conditions. In order to produce 1 Wp, we need a PV transformer of 10 × 10 cm size. However, the sun rarely reaches the illumination of 1 kW/m^2. Moreover, the sun heats up the module significantly higher than the nominal temperature. Both of these factors reduce the efficiency of the module. At 1Wp capacity solar cells performance is about 6 W·h of electricity per day, or 2000 W·h per year [8]. Modern photovoltaic cells have efficiency output of 15.4% [9]. E.g. the battery of 1.681 m^2 has a peak power of 260 W at a wholesale cost of $ 304.2. The daily output of 1 m^2 solar cell will be

$$q_{mod}=6\ \text{W·h}\cdot 260\ \text{W}/1.681\ \text{m}^2=928\ \text{W·h/(day·m}^2) \qquad (1)$$

Determining the required capacity and the number of batteries. Duration of a diesel generator's work according to the operation rules is ten days [4]. Duration of providing electricity from the SSPS is taken to be three days. The possibility of low insulation at the time of an accident is compensated by installation of batteries. It is advisable to use alkaline batteries with long service life, and allowing a large number of discharge and charge cycles. The use of rechargeable batteries with high capacity reduces their unit costs. In this paper, the nominal capacity of the battery is taken as 200 A·h. At voltage of 12V the battery capacity is equal to 2400 W·h. Wholesale price is $ 455 / unit. [11].

According to the condition of permissible interruptions in the power supply, all consumers (services) of electricity for own needs are divided into 3 groups:

- The first group does not allow a break in the power supply for more than a split of second;
- The second group allows for a break in the power supply up to tens of seconds, but requires mandatory power supply after scram;
- The third group admits a break in the power supply and there are no specific requirements to it.

The services of the first group are the system of instrumentation and automation; technological equipment of the reactor and its systems. For these services the batteries are used at de-energizing. In addition to the instruments of control room these batteries provide power to some radiation monitoring systems; electric high-speed channels and shut-off equipment, providing entry into the work of systems of dampening and localization of the accident, as well as part of the emergency lighting; operational control circuits, protection and alarm systems; emergency oil pumps of the turbine generator and the generator shaft seal. The batteries should provide their operation for half an hour.

The services of the second group include mechanisms for reactor cooling down. This paper considers power supply of this group of services.

Consider the function of changes of the residual heat (RH) from time. Table 1 shows the RH values defined under Wigner and Wei law and Wintermayer-Wells formula [10] and those taken from the album of neutron-physical characteristics of the VVER NPP. Calculations were made for the following conditions: four-year fuel campaign; at a power of 3000 MW reactor is operated 286 days, and then on the reactivity effect at 2500 MW capacity it is operated for 24 days, the duration of outage is 55 days. Approximation of data from the album of neutron-physical characteristics (CNF) with power function gave the following equation (the value of reliability R^2= 0.9943):

$$Q_{\beta\gamma} = 413{,}08 \cdot t^{-0{,}279}\text{, MW} \qquad (2)$$

, where t — time after the reactor shutdown, s.

To determine the amount of energy that must be taken away from the reactor core for three days, integrate the function (2). This result coincides with the result of numerical integration: 1272.46 MW·h. The power supply system of vital services of diesel generator must take over the load after 15 s. Power supply from the batteries can be almost instantaneous, taken as 15 seconds. In 15 seconds after the shutdown of the reactor residual heat capacity is 194.05 MW (Table. 1). At this time, pump capacity is 1.75 MW (800 + 800 + 150 kW). Over time, the pump capacity will decrease with the decrease in the released energy of the residual heat. The required amount of energy for driving the pumps is determined from the proportion

194,05 MW —1272,46 MW·h

1,75 MW — E_{3days}

From which the required batteries capacity is E_{3days}=11,475 MW·h. For accumulating such amount of energy we need 11475/2,4=4782 batteries.

Table 1. Function of changing the residual heat with time, MW

Time after shutdown	Wei-Wigner formula	Wintermayer-Wells formula	CNF album	Calculated by (1)
15	90,7	123,35		194,05
30	78,47	111,56		159,93
60	67,81	98,9	117	131,80
100	60,86	89,67		114,30
300	48,09	71,39	85	84,12
600	41,37	61,35	73	69,33
1000	36,98	54,69		60,12
1800	32,45	47,79	55	51,03
3600	27,75	40,6	45	42,06
7200	23,66	34,32		34,66
10800	21,52	31,03	31	30,95
21600	18,23	25,98	26	25,51
36000	16,09	22,68		22,12
43200	15,38	21,58	21	21,02
86400	12,88	17,75	17	17,33
172800	10,71	14,41		14,28
259200	9,56	12,65		12,75
432000	8,24	10,63	10,5	11,06

Finding of characteristics of the field of solar collectors. Let us assume that the required solar cells will provide the daily requirements of pumps power. Similar to the previous section, the amount of energy released in the reactor during the day, will be determined by integrating the expression (2) in the time interval from 15s to 86400 s (1 day) and will be equal to 575.68 MW·h. The amount of electric power to drive the pumps required to remove this amount of residual heat, is determined from the proportion^

194,05 MW – 575.68 MW·h

1,75 MW — Q_{day}

So, the amount of energy that SSPS should generate a day equals to Q_{day}=5.191 MW·h. For production of such amount of energy the total area of photovoltaic transmitters should be:

F_{PVT}= Q_{day} ·1,2/q_{mod}=5191·1.2/0.928=6712.5 m^2,

Where 1,2 — a coefficient for accounting losses.

This corresponds to peak capacity of SPS 6712.5 m^2·260 W/1.681 m^2=1038221 W=1,038 MW.

To assess the area of photovoltaic transmitter just note that this is a square with 81.93 m side (without accounting of the passes between panels).

Determination of technical and economic parameters of SSPS. Table 2 shows the results of calculation of the estimated value of SSPS cost:

Specific investments are 4,013·10^6/1038=3866 $/kW.

Annual electricity generation:

0,928 kW·h /(m^2·day) · 5191 m^2 · 365 days

=2,273·10^6 kW·h.

Electricity supply:

2,273·10^6–11475=2,262·10^6 kW·h.

With these data, the utilization of nuclear power installed capacity will be equal to:

Load factor =2,273·10^6/(0,1546 kW/m^2·6712,5 m^2·8760 h/year)=0,25.

Table 2. Estimated cost of SSPS

Name of equipment or work	Amount	Cost of a unit, $	Total cost, $	Cost, %
Photo-voltaic converter (PV)	6712,5 m^2	180,96 $\$/m^2$	$1,214 \cdot 10^6$	30,26
Batteries	4782 item.	455 $/item.	$2,176 \cdot 10^6$	58,77
Inverter	1,038 MW p	0,1 $/W p	$0,104 \cdot 10^6$	2,80
Control means	1,038 MW p	0,2 $/W p	$0,208 \cdot 10^6$	5,61
building and installation	1,038 MW	300 $/kW	$0,311 \cdot 10^6$	7,76
		TOTAL :	$4.013 \cdot 10^6$	100,00

The cost of electricity production is derived from the following items (including additional costs for batteries and inverter):

- depreciation expenses (15% of the SSPS): 0,612 · 106 $;
- salary for 10 people, $ 4400 / year: $ 44,000;
- general plant costs (13% of the amount of depreciation expenses and payroll): $0.084 \cdot 10^6$ $;

Total $0.730 \cdot 10^6$ $.

So, the net cost of electricity is equal to $0.730 \cdot 10^6 / 2,262 \cdot 10^6 = 0.32$ $ / kWh.

Green tariff for electricity generated by solar power plants since 01.01.2015 is 1.84 UAH / kW · h = 0.084 $ / kW·h, i.e. it does not cover the costs of production.

Thus, it is unprofitable, for the existing prices the use of SSPS is not covered by income i.e. price of production does not cover the net cost.

The choice of battery type and their price is not a simple question. The solar stations ready for installing come without batteries. Without accounting the price of batteries the economic indicators will be as follows:

- SSPS cost 1.837 m $;
- specific capital investments 1770 $/kW;
- cost 0.17 $/kW·h.

That is, the results of the calculation allow us to make a conclusion that the SSPS under the assumed price for the equipment and the electricity tariff is unprofitable. It should also be noted a steady tendency to reduce the price and increase of the efficiency of solar modules, which can significantly improve the economic performance of SSPS.

To make a final conclusion, we compare the performance of using SSPS and DG as a reliable source of power. The cost of the ASD-5600 is estimated at $ 950,000. Fuel consumption per hour is 1340 kg. This is at the heat of combustion of diesel fuel corresponds to the electrical efficiency of 35.8%. Fuel is stored in two tanks: consumption one of 10 m^3 and the intermediate one for 100 m^3. The cost of this fuel is 110 $m^3 \cdot 828$ $ / m^3 = $ 91100. Comparison of economic indicators is shown in Table. 3

Table 3. Calculation of economic indicators for the use of SSPS and diesel=generators as a reliable power supply source, $

Name of the indicator	SSPS	ASD-5600
Equipment cost	$4,013 \cdot 10^6$	$0,95 \cdot 10^6$
Fuel cost	-	91100
Annual expenses (including): - depreciation expenses - labor cost - general plan expenses	$0,730 \cdot 10^6$ $0,602 \cdot 10^6$ 44000 $0,084 \cdot 10^6$	180913 142500 17600 20813
Annual profit	$0,190 \cdot 10^6$	-
Expenses for 30 years	$16,2 \cdot 10^6$	$5,518 \cdot 10^6$

Conclusions

1. We considered the economic feasibility of the use of a semiconductor solar power station, which provides charging and maintenance of the batteries, as one of the reliable sources of power to vital services of the reactor building. Power supply of pumps involved in the reactor cooldown is carried out at blackout by the batteries.

2. Technical characteristics of the main equipment of the semiconductor solar power station, providing emergency operation of the pumps with 1.75 MW capacity with frequency regulation, were determined as follows:

• area of solar modules with an efficiency of 15.4% should be equal to 6712.5 m^2, SSPS peak power is 1,038 MW,

• capacity of the batteries for the three-day operation to provide operation of necessary pumps must be equal to 11.475 MW·h = $0.956 \cdot A \cdot 10^6$hours (at a voltage of 12 V).

3. From the economic point of view at the green tariff of $ 0.084 / kW·h the use of SSPS is unprofitable. Moreover, the maintenance and service cost of RDES is almost three times cheaper than the provision of solar power, performing the same functions.

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***S. I. Azarov*[1]*, O. V. Popovich*[2]*, V.L. Sydorenko*[3]**

[1]*Institute for Nuclear Researches, National Academy of Sciences of Ukraine, Kyiv*
[2]*Ministry of Environment and Natural Resources, Kyiv*
[3]*Institute of Public Administration for Civil Protection, m. Kyiv*

CONCEPTUAL DIRECTIONS OF IMPLEMENTATION OF RADIATION SAFETY CULTURE

The principles of radiation safety culture were analysed as regards minimization of risks in case of operation of radiation-hazardous objects. The basic principles and components of improving radiation safety culture and a system for synthesis of radiation safety are considered. Problems to be solved are identified for the process of implementation of radiation safety culture. Attention is paid to the basic principles of radiation safety culture at safe operation of radiation hazardous objects and the role of human factors in ensuring radiation technological safety

Key words: radiation safety, radiation-hazardous objects, radiation safety culture

One of the priorities of national safety in Ukraine is ensuring radiation safe conditions of life of individuals and the society, preservation of the environment and rational use of natural resources [1-4].

Formation of a new culture of safety, which is based on increasing of the degree of development of an individual and the society, becommes possible by transformation of consciousness of all strata of the population. Development of new mentality should be preventive, allowing the public to switch the priority from protection in emergency situations (ES) to prevention of these situations, addressing the causes of radiation-hazardous threats and safety of human life. Radiation safety (RS) is a set of administrative, technical and sanitary measures that limit radiation exposure of personnel and population and contamination of the environment; it is also a scientific and practical discipline that predicts the radiation situation in the course of operation of radiation hazardous objects (RHO), investigates cases of radiation incidents and make recommendations to bring the RHO into a state of compliance with the established standards (sanitary and methodical and legislative documents and standards) [5, 6].

Radiation hazardous object – such object in which hazardous radioactive substances, ionizing radiation sources (IRS) and radioactive waste (RW) are used, processed, stored or transported, which according to [1] represent a real radiation threat in case of man-caused emergencies.

By analogy with the structure of the IAEA standards, we can offer the following RHO classification and activities [7, 8]

• nuclear power plants (NPPs) in operation;
• NPP that have been decomissioned;
• the “Shelter”;
• enterprises of the nuclear fuel cycle;
• research nuclear installations;
• sources of ionizing radiation;
• transport of radioactive materials;
• radioactive waste;
• ensuring physical security of nuclear facilities and nuclear materials;
• emergency preparedness and response;
• radiological protection.

There are rather a lot of potentially radiation-hazardous objects in Ukraine, and so there are a number of regions of intense radiational and even crisis condition of the environment. Today in Ukraine there are 4 nuclear power plants (Zaporizhzhya, South-Ukrainian, Rivne, Khmelnytsky) with 15 power units in operation (13 - VVER-1000; 2 - VVER-440), which in number and capacity lead Ukraine to the eighth place in the world and the fifth - in Europe; there is 1 research nuclear reactor WWR-M (in Kyiv); a critical assembly (in Kharkiv), and more than 8,000 companies and organizations that use more than 100 thousand IRS (only in Kyiv there are about 400 of them) [9, 10].

At NPPs in operation there are radioactive waste storage facilities and cooling pools for spent nuclear fuel - fuel rods with radioactive materials.

At the shutdown Chernobyl power units 1-3 of RBMK-1000 type decommissioning works are being carried out, and the “Shelter’ is in urgent need of isolation from the environment by constructing the Confinement [11].

In Ukraine, radioactive materials (radioactive substances, radioactive waste and radiation sources) that were used at RHO are stored in the vaults of six state-owned interregional specialized plants “Radon”, five storage shelters of military-industrial complex, in the storage vaults of the "Vector" and in the disposal points for decontamination waste located in the 30-kilometer exclusion zone of Chernobyl nuclear power plant, etc. [12]. Safe handling of radioactive waste in these facilities is subject to state regulation and control to ensure compliance with national and international requirements to protect human health and the environment [13]. Special attention is required by potentially hazardous objects such as interregional specialized plants for processing and storage of radioactive waste of the State production association "Radon" (Kyiv, Lviv, Donetsk, Dnipropetrovsk, Odessa and Kharkiv), 5 enterprises of mining and processing of uranium ore, which are located in Dnipropetrovsk, Mykolaiv and Kirovograd regions.

Current strategy of safe radioactive waste management is aimed at maintaining and isolation of radioactive waste from the biosphere. To implement this strategy, there carried out collecting, processing (conditioning) and storage of radioactive waste until their possible release from further regulatory control or final disposal which is the ultimate goal of radioactive waste management.

Until recently, the disposal of radioactive waste and spent IRS of non-nuclear industry enterprises have been carried out at the sites of state interregional specialized plants UkrDO "Radon" without proper sorting of radwaste with no assessment of impact of the burried radioactive waste on the humans. In addition to the mentioned flows of radioactive waste in Ukraine there are plenty of radioactive waste with specific characteristics arising from the ChNPP accident. Currently radwaste of different levels of activity are localized in the Exclusion zone: at the "Shelter", in the points of temporary location or disposal, and in particular, in the current repository "Buriakivka".

According to [13] and [10] the construction of RAW-processing plants and storage facilities for long-term storage of radioactive waste and used IRS, and facilities for disposal of all types and categories of radioactive waste are provided for at the Vector site in the Exclusion zone. Almost all radwaste from Ukrainian enterprices, including the Radon, the operating nuclear power plants, Chernobyl NPP, the Shelter and the Exclusion zone are planned to be tranported to the Vector facilities.

Uranium mining and processing industry have been accumulated around 65 million tonnes of solid low-level radioactive waste.

According to the Ministry of Health [14] of Ukraine, there are about five thousand enterprises and organizations using different radiation sources. Health care facilities operate 10280 X-ray diagnostics units, 254 X-Ray therapy, and 118 gamma-therapy units, 6 isotope medical devices. Industrial plants use 550 gamma-ray flaw detectors and radioisotope devices, installations for irradiation with Co-60 sources, thermoelectric installations with Sr-90, and mobile units for geological survey purposes with Am/Be sources, etc. Average annual total activity of radioactive waste and radioactive sources used in industrial and medical applications, can reach about 10^{16} Bq, and so there is a probability of radiation accidents involving radioactive contamination of the environment and exposure of personnel and general public.

The goal of radiation safety is to protect personnel, the public and the environment against harmful radiation effects and consequences through creation and maintaining at RHO of effective methods and means of protection against radiation. To achieve this goal, the following principles [5] are established:

- Any activity, accompanied by irradiation of people should not be carried out unless it is more beneficial for exposure to individuals or society as a whole compared with the damage which it has (principle of justification);
- The levels of exposure from all significant types of practice should not exceed the established dose limits (principle of non-exceeding);
- The level of individual doses and / or the number of exposed individuals from each individual source of radiation should be as low as possible, taking into consideration economic and social reasons (principle of optimization);
- An independent body that regulates the RHO activities should be established;
- Responsibility for radiation safety relies on the operating organization;
- Analysis and evaluation of all hazards that affect the level of radiation safety should be conducted;
- "Human factor" and its influence on radiation safety should be taken into consideration at all stages of RHO operation.

Radiation safety principles are combined into three groups:

1) radiation control (RC) and management (development of methods of decision making and implementation of quality and safety culture);

2) radiation protection (application of protective barriers on the way of ionizing radiation and radioactive substances, etc.);

3) Improved technical and organizational measures (industrial site selection, installation of sanitary protection zone (SPZ), etc.).

Radiation safety is an important tool in the National Safety and Security System, because every year radiation incidents and accidents occur at RHO, which result in people’s injury and death, and radioactive contamination with radionuclides, causing damage to the environment. But the potential for radiation safety culture is not fully implemented in RHO, primarily due to the absence of this concept in the legislative and regulatory framework, and lack of new tools for its implementation, as well as due to lack of political will for implementation of radiation safety measures in compliance with the recommendations of the European Union (EU)

The aim of radiation monitoring at RHO is to get using instrumental methods necessary information about the radiation situation and levels of exposure to personnel, population and environment. Analysis of this information makes it possible to monitor compliance with the established standards to identify and promptly remove the sources of increased danger, to consider various factors of radiation impact on the environment and undertake necessary protective measures to reduce this impact to the lowest possible values.

On the other hand, to identify approaches to improving the regulatory framework in terms of safety management and radiation safety culture during RHO operation in Ukraine itt is necessary to initiate work on comparative analysis of modern radiation safety standards existing in the EU, and to identify ways of their practical application

It is known that the concept of ‘safety culture’ emerged during in the work of the International Nuclear Safety Advisory Group - INSAG), and was included in the scientific and technical terminology after publication of [15]. In modern philosophical, sociological, psychological, cultural belief, there are many definitions of safety culture. Most of them can be reduced to the following definition: “Culture is, first of all, the system of values adopted in the society’. Safety culture is a part of human culture and cultural part of the national culture, it is directly related to the culture of the company, the man, and his personality. The system of human values is a kind of ‘core identity’, which has a significant impact on other personal characteristics and qualities, including the professionally important ones.

On our opinion, the radiation safety culture - is a part of the overall safety culture, which includes shared values, attitudes and behaviors with specific characteristics. Radiation safety culture is a catalyst for improving the RHO safety by providing scientifically accurate objective data to managers to develop and implement strategic initiatives that mobilize staff to minimize radiation and man-made risks. In scientific publications [16, 17] radiation safety culture is defined as the level of the individual and society, characterized by the importance of maintaining the safety of life in the system of personal and social values of safe behavior in everyday life and in a dangerous events and emergency situations, the level of protection against radiological threats and hazards in all spheres of life.

Accordingly, the radiation safety culture is based on the following components:

• Individual level - worldview, norms of behavior, values and individual preparedness of human in the life safety;

• At the collective level - corporate values, professional ethics and morality, preparedness of personnel in radiation safety;

• At the societal level - the tradition of safe behavior, social values, preparedness of the total population in the life safety.

Radiation safety culture belongs to the general concept of personal responsibility and commitment of those persons that are engaged in any activity that affects the safe operation of a RHO. The reasons of establishing radiation safety limits and the consequences of their violation are especially highlighted.

Conditions of safe operation of RHO is knowledge and complete understanding of potential radiological hazards associated with the RHO operation.

Safe RHO existence and ensuring their efficient operation requires proper regulatory framework and new approaches to improve radiation safety. The main components of improving operational safety of RHO are improvement of RHO design, improvement of operational procedures, and the radiation safety culture.

An integrated approach to accident-free operation of RHO involves continuous improvement and development of new regulatory acts, organizational and administrative documents, in which radiation safety culture should occupy the central place.

Fundamental principles of radiation safety culture at the highest level of safe RHO operation are the following (Fig. 1):

1) the leading role of the state in maintaining national safety and security at a high level;

2) establishing of the priority of nuclear, radiation and technical safety over the general economic and industrial goals;

3) personal responsibility of the executive body for nuclear, radiation and general technical safety;

4) scientific and technical justification of RHO safe operation, taking into account the human factor;

5) optimization of protective equipment;

6) restrictions initiating events that lead to emergency situations;
7) early revealing of signs of pre-accident situations and prevention of severe accidents and their consequences;
8) emergency preparedness and response to emergencies of anthropogenic and natural character;
9) protection of present and future generations against consequences of nuclear, radiation and industrial accidents

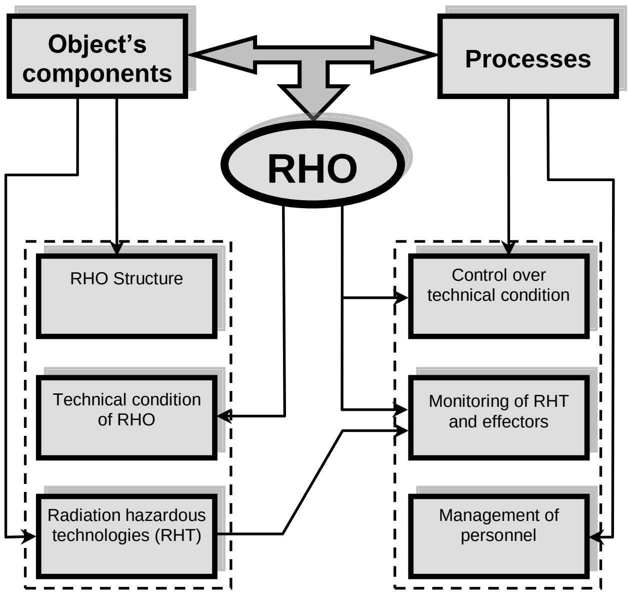

Fig. 1. Integrated scheme of the system of radiation safety culture at operation of RHO

Basic principles of radiation safety culture are connected with the following issues [18]:
1) RHO interacts not only and not so much with the staff (the operator), but with some human community (population);
2) RHO 'do not know' laws of natural and man-made hazards and emergency situations;
3) RHO is always absolutely rigid deterministic (causal) potentially dangerous system;
4) Rules and regulations of safe operation of RHO are always relative;
5) Decision-making levels in the hierarchical structure should take into account the mutual relations (RHO – human factor – the environment);
6) Feedback from each level of the hierarchical structure of a managing group to a lower level should be sufficiently complete and objective for the possibility to correct influence, passing through the management level;
7) There should be and over-structure controlling and correcting independent body that severely repress any uncorrected signals.

Commitment to radiation safety culture is achieved by:

• priority of nuclear, radiation and general technical safety over economic and industrial goals;
• recruitment, training and professional development of managers and staff of the operating organizations, regulatory bodies, equipment manufacturers and service providers;
• strict discipline with a clear division of authority and personal responsibility of managers and direct executors;
• compliance with operational instructions and technical rules of safe operation, their continuous improvement on the basis of accumulated experience and scientific and technical research;
• establishment by leaders at all management levels of trust and teamwork approaches that promote a positive attitude to radiation safety;
• understanding by each employee of influence of their activities on radiation safety and the consequences of poor performance or failure to comply with regulations, production rules, job descriptions, and technological requirements of RHO safe operation;
• self-management by employees of their activity that affects radiation safety;
• understanding by each employee of inadmissibility of concealing errors, and the need to identify and eliminate their causes, continuous improvement, learning and implementing best practices, including foreign

experiemce;

• establishing of such a system of rewards and penalties based on the results of production activity that encourages openness and prevents actions of concealment of errors in work

Radiation safety culture in the operation of RHO is determined by the degree of completeness of regulatory, technical, organizational and administrative, technical, operational and accounting records of radiation safety control procedures of potential radiation hazardous processes at the level of any and all hierarchical management systems of RHO and at the individual level (operator). The process of measuring the radiation safety culture is based on qualitative and quantitative indicators, leadership and management tools and direct communication. During the implementation of the process it is necessary to solve the following problems [19]:

- Describe the radiation safety culture, which has been formed on RHO, using quantitative indicators and identify trends in overall safety culture;
- Integrate the principles of radiation safety culture in the working processes;
- Form commitment of workers to principles of RS culture and active use of the experience of the staff of the existing RHO in the world;
- Provide centralized planning for the development of actions on development of RS culture, self-assessment of operational RS safety of RHO and arrangement of Days of Safety;
- Evaluate the effectiveness of the implemented measures to support high radiation safety culture

Evaluation of radiation safety culture is a catalyst for changes of radiation hazardous processes by providing data to managers for developing and implementing strategic initiatives that mobilize people to move to new directions of work. In analyzing the state of radiation safety culture of RNO there are three main aspects:

• the ability to identify hidden weaknesses and unsolved problems of RS;
• the ability to identify the significance of events and safety issues and to adequately respond to them;
• the ability to take into account previous experience and eliminate problems of RS.

Radiation safety culture is largely related to the problem of the human factor, which plays a crucial role in the operation of RHO and provides high qualification and psychological training of personnel at a facility. Public authorities in the regulation of radiation safety should adhere to the principles of radiation safety culture in their supervisory and licensing activities.

Ministry of Health at re-assessment of basic documents on radiation safety - Radiation Safety Standards of Ukraine (NRBU-97) [5] and the Basic Sanitary Rules (OSPU 2005) [7] - should take into account the recommendations of the IAEA, WHO, the The International Commission on Radiological Protection (ICRP) [20] etc., with the aim to introduce modern principles of the safe use of radiation sources.

The main components of radiation safety culture are: commitment to general technical safety culture, using established procedures, conservative approach to decision-making, the enterprise documenting system, system of changing unsafe actions and conditions, system of training, information sharing etc.

Additional components of RS are: operation management, operation practice, resources, operating experience, independent assessment and self-assessment, corrective action program, willingness ‘to raise questions about safety’, sagety policy, responsibility, organizational change management, continuous learning environment.

Safety management is used to promote high radiation safety culture by:

• Providing common understanding of key aspects of general technical safety culture within the organization;
• Providing employees and teams with additional means for the safe and successful execution of the tasks with taking into consideration the interaction of workers, radiation technology and organization activities;
• Encouraging constructive and critical positions at all levels of the organization;
• Providing the means by which the organization may strive to continuously develop and improve its safety culture.

RS Management System should be used, constantly evaluated and improved, be consistent with the objectives of the organization and contribute to their achievement. The main goal of management system is to achieve and improve radiation safety by:

- Consistent consolidation and integration of all the requirements of the organization management;
- Description of planned and systematically carried out actions necessary to provide adequate confidence that all these requirements are met;
- Providing due support of requirements related to issues of occupational safety and health, environmental protection, nuclear and general technical safety, physical security, quality and economy jointly with the safety requirements in order to eliminate any possibility of radiation impact on living conditions of citizens.

Safety actions are of paramount importance in the management and priority over all other needs. An important element of the radiation safety management system is documentation management system that includes:

• statements about the policy;
• Description of management system;
• Description of the organization structure;
• Description of functional responsibilities, accountability conditions, levels of authority and interaction between employees who manage the course work, perform and evaluate it;
• Description of processes and supporting information explaining the preparation, review, execution, registration, evaluation and improvement of the quality of work.

In the RS management important aspects are the responsibility and commitment of executives and manangers to safety culture. Management staff at all levels should demonstrate commitment to creation, implementation, evaluation and continuous improvement of management system and allocate adequate resources for conducting this activity. Management at all levels should be made available to workers how important are personal values, institutional values and standards of behavior and fulfillment of the requirements of management system. To effectively achieve this goal, management should encourage participation of employees in continuous improvement of management system. It is important to note that the top-level management has primary responsibility for the management system and ensures its creation, application, evaluation and continuous improvement.

In the framework of radiation safety system the information and knowledge available in the organization should be considered as one of the types of resources. In so doing, top-level management defines, creates, supports and re-evaluates the infrastructure and working conditions required for the safe performance of work and compliance with all legal and regulatory requirements. The processes that exist in the organization should be identified and described. For process management a responsible for the RB person should be appointed who is vested with certain powers and responsibility for

• developing and documenting the process and record-keeping of all necessary supporting documentation;
• ensuring effective cooperation between related processes;
• ensuring that the documentation associated with the process, any existing documents;
• maintaining a documentation process associated with records that are necessary for confirmation of achievement of a process;
• monitoring of processes implementing and reporting;
• promoting the improvement of processes;
• ensuring that a process (including all subsequent changes) complies with objectives, strategies, and plans of the organization.

For each process the information should be provided on the activities of inspection, testing, checks and certifications, eligibility criteria and responsibilities for conducting such activities, and the need of such activity and the designated employees or groups with the aim to confirm the ability within the certain processes to carry out planned activities, and to search capabilities to improve, monitor, measure, self-assess and independently assess the safety management system. In the event of deficiencies in the management of radiation safety one should determine the causes of nonconformities and undertake actions to remove them and avoid their repetition.

Common methods of evaluation of actions to improve the level of radiation safety culture is an environmental audit and radiological examination, reports and continuous monitoring of key performance indicators of radiation hazard, observations and surveys of staff and the public. The staff working in the production, use, storage, transportation and processing of radioactive substances and materials in the production of radioactive fuel for nuclear power and in the field of radioactive waste management is challenged to provide such a radiation safety, which could be perceived by society as complete. Effects of radiation safety management system is based on the process approach (Fig. 2).

Problem of human factor can be solved by professional selection system, special training and certification of personnel in continuing their training and psychological preparation.

Formation and monitoring of radiation safety culture at RHO is performed by:
- Preparing training and methodology materials and conducting self-assessment of operational safety;
- Strengthening the role of monitoring of the various types of operational activity;
- Development of additional guidelines and training materials aimed at increasing awareness of the importance of the personal aspects of industrial and radiation safety;
- Improving the training of operational and maintenance personnel

It should be noted that the radiation safety culture is a set of moral values and measures aimed to prevent unforeseen emergencies at different RHO. Radiation safety culture means understanding by each person of thgeir personal responsibility for the consequences that may affect personal and general safety. Radiation safety culture, as it has been already noted, should include a set of characteristics and features of the behavior of organizations and individuals which establishes that safety issues are concerned according to their significance. These primarily include:

- Preventing complacency during normal operation;
- Understanding by personnel of potential importance to the safety of all deviations from the operation rules;
- Recognition of the priority of safety in making decisions;
- A sense of responsibility for their actions and measures;
- Systematic approach at all stages, not neglecting small things.

Radiation safety culture requires from personnel not only to obey the previously developed guidelines and procedures, but also to constantly search ways for further improvements, to reduce the risk to a minimum level of almost impossible event. This focus on radiation safety and on ways to improve it, in the people minds in relation to other values should correlate with the level of theoretical and practical training, a clear knowledge by workers of tyheir professional requirements and job descriptions as well as necessary for professional work personal and psychophysiological qualities of workers.

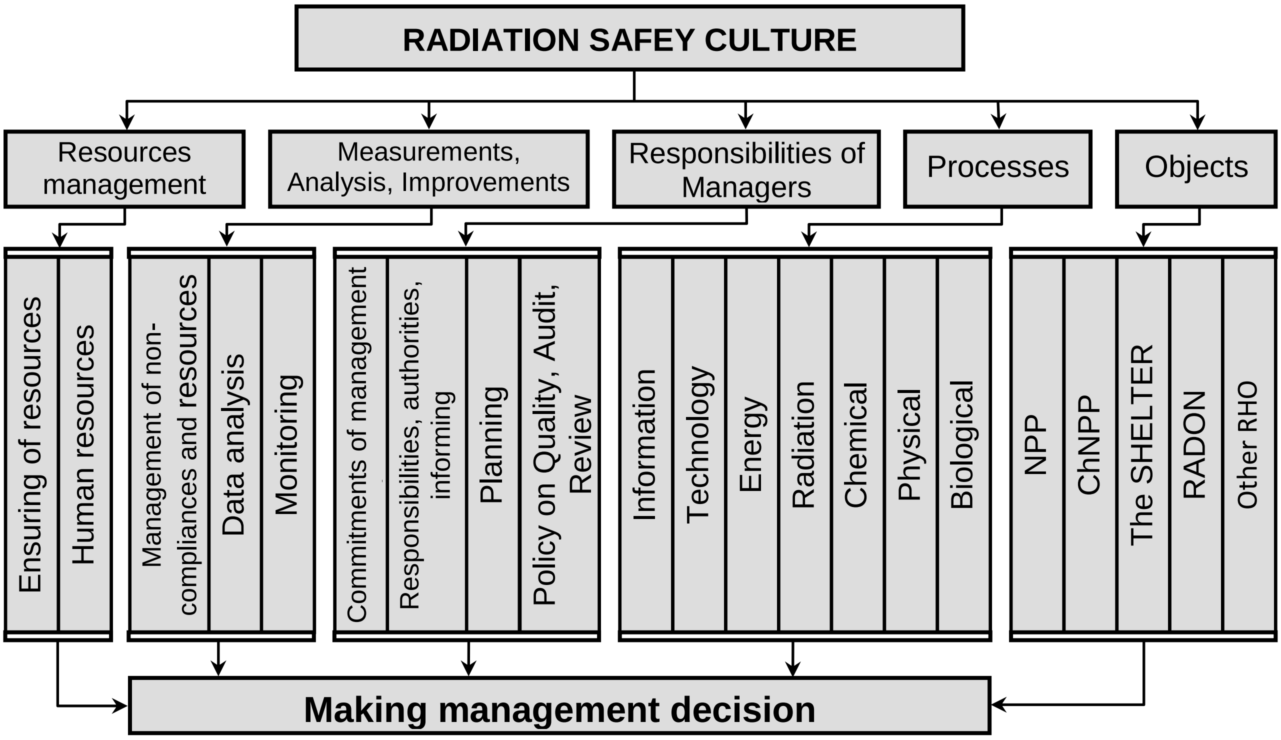

Fig. 2. Hierarchyof interactive system of radiation safety management

Conclusions

Thus, the radiation safety culture is a set of rules, regulations, standards, guidelines and operating characteristics of industrial organizations and staff of RHO, which states that the problems of radiation safety as having the highest priority according to their significance.

Given the importance of radiation safety culture, in many countries it has become a subject of attention of the top executives, and the state agencies on regulation of safety in various areas recognized it as one of the priority strategic areas of their activity.

Given the critical situation in Ukraine in creating the radiation safety culture, on one hand, and the imperfection of organizational and management system - on the other, it seems necessary to review the state policy in this area.

In Ukraine, national RHO began systematically implement the concept of radiation safety culture, but it further distribution was hampered on the following reasons:

- Actual absence of non-departmental coordination and methodological center that would facilitate implementation of the general organizational approaches to improve the situation of radiation safety;
- At the level of central and local executive authorities there is lack of attention to implementation into practice of the basic principles of the radiation safety culture as an effective tool to create and continuously maintain safe working conditions and living standards of citizens;
- The existing regulatory and procedural documents for implementation of the basic principles of radiation safety culture in practice in enterprises, organizations and institutions that use technologically dangerous technology are imperfect, and monitoring of their observance is inadequate;
- Executive authorities pay insufficient attention to this issue at development and implementation of state policy in the sphere of human life at work and prevention of home traumatism;
- Both professional and public educational institutions pay insufficient attention to the issues of radiation safety culture.

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A.V. Korolev, O.P. Ischenko

Odessa National Polytechnic University, Odessa, Ukraine

POTENTIAL HEAT CAPACITY OF SPENT NUCLEAR FUEL

A computational model of a spent nuclear fuel container with a system of pipes is proposed. Based on the analysis of the simulation results the theoretical possibility is demonstrated of implementation of this system with evaluation of heat that can be obtained using spent nuclear fuel.

Key words: spent nuclear fuel, container, mathematical model, a fuel assembly, thermal energy

Spent nuclear fuel is a fuel which is irradiated during one fuel cycle in the nuclear reactor. The spent fuel contains both uranium-235 underburned during reactor operation, as well as its isotopes and other transuranic elements, and activated structural materials. Therefore, nuclear transformations continue to occur in fuel assemblies (FA). They are accompanied by increased radiation and permanent release of heat (slowly decreasing over time).

Methods of spent nuclear fuel and radioactive waste management might be different by strategic approaches and by technologies used, but they are united by the general principles related to safety issues. These principles are developed and distributed by international organizations, in particular the IAEA, through appropriate rules and guidelines, as well as through the international treaties, in particular the "Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management" [1]

Storage of spent fuel assemblies (SFA) in the cooling pond of a nuclear power plant (NPP) is a mandatory element of the technological chain, which is determined by high activity of the fresh-irradiated fuel and the need to remove large amount of residual heat.

As a result of analysis of the causes of the Fukushima accident, studies were conducted on evaluation of power of SFA residual energy in cooling ponds, calculation of the time margins at loss of cooling, and measures to prevent boiling of water cooling the SFA.

This work clearly shows how much residual heat can be released by the stored spent fuel assemblies, which are in storage in pools and are ready for withdrawing and further sending for disposal or recycling.

Model description.

Convincing model used in this article to demonstrate the magnitude of residual energy that could be released by SFA is implemented as a heating system. It is an example clear for each citizen and one of the most disturbing factors of the public, due to significant increase in tariffs.

To implement this model, the computer code Melcor was used [2]. With this code, a container model was developed in which 3 SFA-W of 347WG type were placed with burn-up of 55 MW d/kgU [3]. These SFA have the highest value of residual energy, and therefore have the greatest potential.

The model was made up of four control volumes and 3 medium flows.

Test volumes:

SV001 - Container. The height of liquid in the container is 6 m, diameter 1.2 m, the volume of liquid in the container is 6.44 m^3 taking into account the volume subtracted by the placed SFA.

CV002 - the pipe of conditional diameter D_y= 32 mm – a standard heating pipe. Pipeline length of 70 m was selected.

CV003 - is a control volume, simulating the environment with a temperature of 30°C and atmospheric pressure, 10 m high and a volume of 1000 m^3.

CV004 - is the volume inside the container 1 m in height, intended to produce steam “cushion” in it for compensating thermal expansions of liquid and damping pressure pulsations in the container.

FL001, FL002, FL003 - function describing the flow of the medium between the control volumes.

The wall of the pipe in the model is divided into seven layers for better modeling of the heat transfer from the piping to the environment

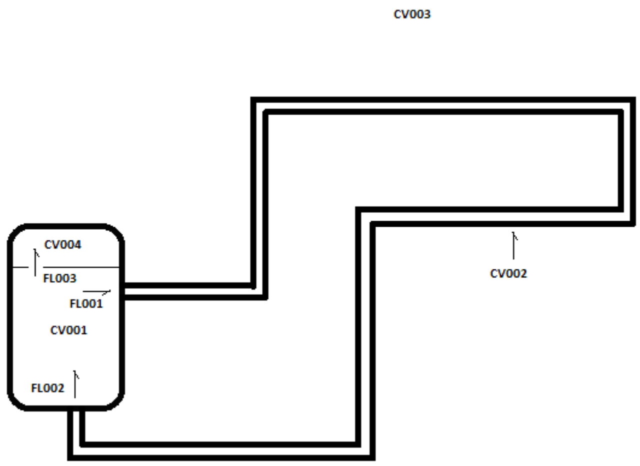


Fig. 1. Model of water heating system by residual energy releases from SFA

Energy release capacity is taken for the SFA, which is cooled 3 years in the storage pool and can potentially be withdrawn [4], as well as for the SFA with capacity of residual energy 10 and 20 years after the end of their operation in the reactor core of VVER-1000.

Analysis of the results. The key parameters that were considered in the analysis of simulation results are as follows:

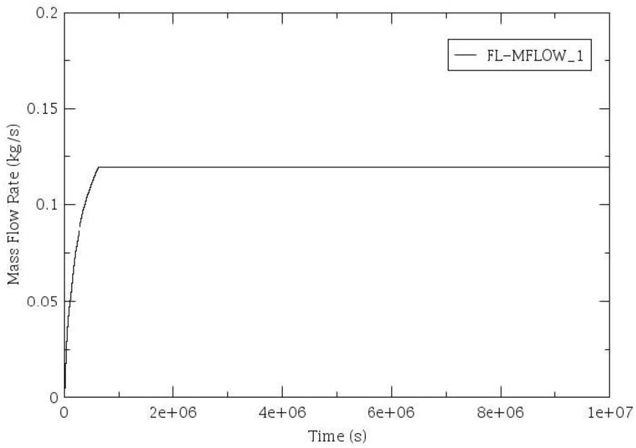
- The rate of flow of the medium between the control volumes;
- The pressure in the control volume SV001 and SV002;
- Temperature in the control volumes SV001 and SV002.

The results of calculations of the model of a water heating system by residual heat from the spent fuel assemblies shown in Fig. 1 are presented in Table 1. It demonstrates changes of heat capacity ($N_{отв}$) of 3 SFA, temperature (t) and pressure (P) in the container (CV001) and pipe (CV002), as well as water flow (F) under conditions of natural circulation.

The calculation results for SFA 3 years after the end of their operation in the reactor core are shown in Figure 2

Table 1. Results of calculations for SFA for 3, 10 and 20 years after the end of their operation in VVER-1000 reactor.

T, years	$N_{отв}$, kW (3 SFA)	Volume	t, ºC	P, kgf/cm^2	F, m^3/h
3	8,31	CV001 CV002	162 152	10,16 10,04	0.43
10	3,21	CV001 CV002	95 89	2.44 2.24	0.27
20	2,46	CV001 CV002	79 69	1.83 1.63	0.21



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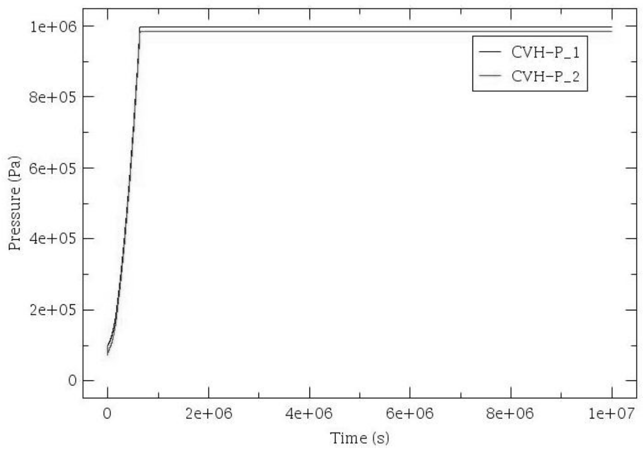

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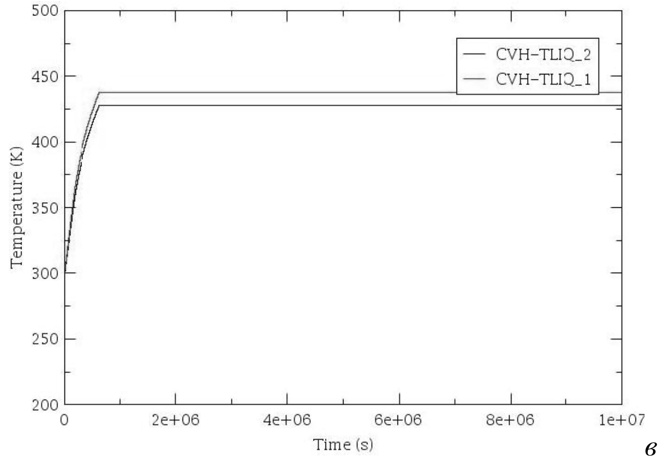

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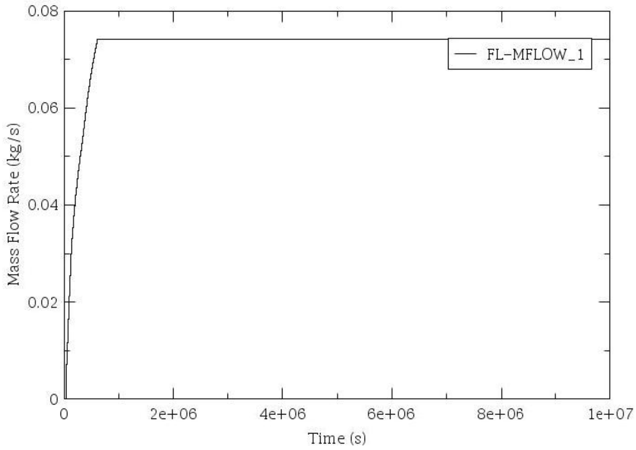
Fig. 2. Time dependence of changes for SFA in 3 years after the end of their operation in the reactor core:
a) medium flow speed between the control volumes;
b) pressure in the control volume SV001 and SV002;
c) temperature inside the control volumes SV001 and SV002.

The calculation results for SFA in 10 years after the end of their operation in the reactor core are shown in Figure 3.

The calculation results for SFA in 20 years after the end of their operation in the reactor core are shown in Figure 4



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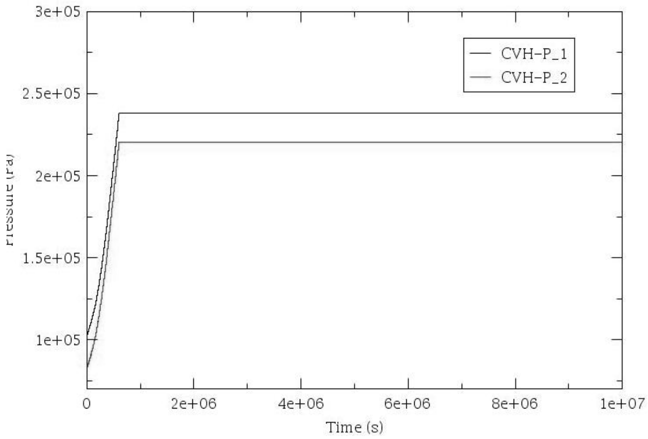

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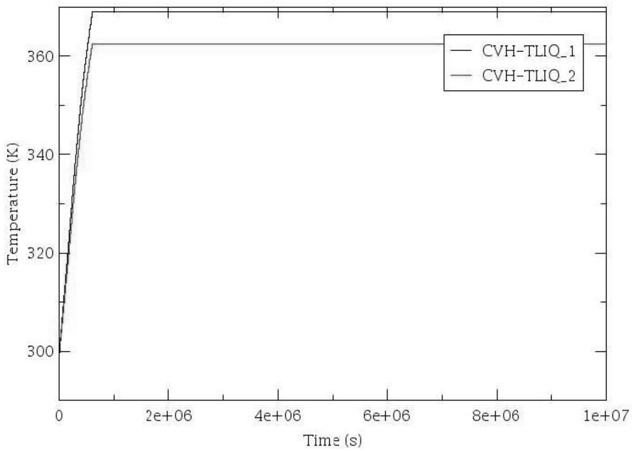

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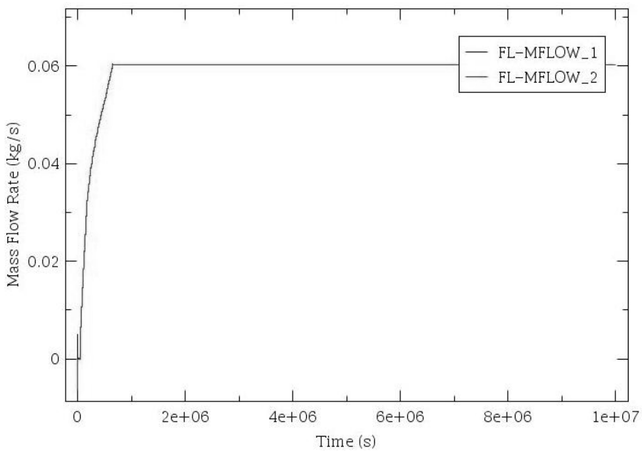
Fig. 3. SFA changes in 10 years after the end of their operation in time:
a) flow speed of the medium between the control volumes;
b) pressure in the control volume SV001 and SV002;
c) temperatures in the control volumes SV001 and SV002.



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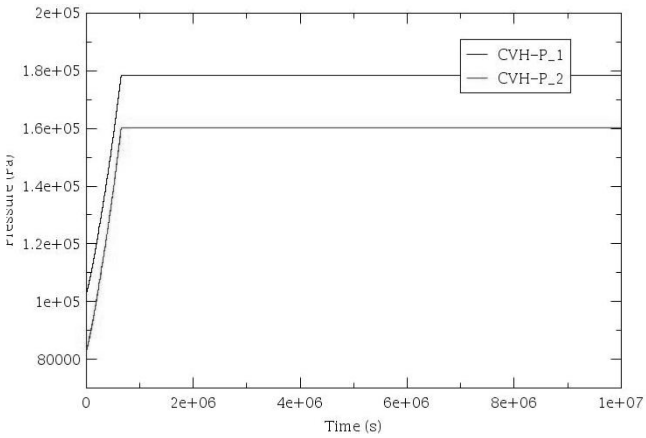

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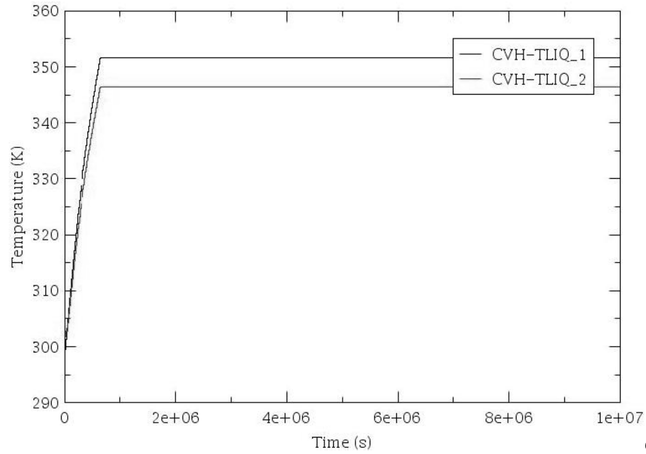

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Fig. 4. Changes to the SFA in 20 years after the end of their operation in time:
a) flow speed of the medium between the control volumes;
b) pressure in the control volume SV001 and SV002;
c) temperatures in the control volumes SV001 and SV002.

Analysis of the obtained data allows us to make the following conclusions:

1. Starting with 19,000 seconds (5 hours 17 minutes) of the process stable natural circulation of coolant appears between the control volumes, which rises with heating of the coolant in the container and became stable at the time of temperature stabilization in volumes.

2. After 11.5 days after the start of the process the pressure in the control volume become stable.

3. Stabilization of the temperature leads to stabilization of the pressure and flow rate of natural circulation, the process becomes stable through 11.5 days after the start of the process. The difference in temperature and, accordingly, the pressure can be explained by presence of heating of the coolant from the spent fuel assemblies in the control volume CV001 and the cooling of the coolant in CV002 control volume.

Conclusions

The results of calculations show that after 20 years 3 SFA will heat water in CV002 control volume, which simulates the heating system piping, up to 79ºC. An important parameter, which we managed to obtain, was the presence of a medium flow between the control volumes, which indicates the presence of natural circulation of the coolant. The presence of a stable low pressure in the system shows a properly chosen container construction.

In the cooling pond of VVER-1000 reactor there are 563 SFA cells, which total residual energy is about 5280 kWh [5]. Therefore, at loss of coolant accident of the storage pool there is a danger of temperature increase up to the boiling point.

The computational model shows how significant is the energy potential of the fuel, which may have been unloaded from the pool and is ready for further use.

For implementation of such a system one need to solve many technical issues. However, such studies allow us to determine the direction of future work in the use of spent nuclear fuel and allow evaluating its potential for understanding whether to spend time and money on the development of this direction.

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Yu.O. Olkhovik

SI "Institute for Environmental Geochemistry NAS of Ukraine", Kiev

ON PROTECTIVE PROPERTIES OF AERATION ZONE FOR THE VECTOR FACILITY SITE

The article gives description of the geologic structure of the aeration zone around the “Vector” facility’s site, intended for near surface radioactive waste disposal. It is demonstrated that the aeration zone is mainly composed of sandy soils characterized by high content of fine powder fractions with clay minerals. The analysis was performed of sandy soil sorption properties, which were found experimentally, and the values of distribution factors of Sr, Cs, Pu radionuclides were calculated for the strata of the zone of aeration in the whole. Estimations were made for the time of transition of water dissolved nuclides through the strata of the zone of aeration. It is noted that sandy soils of unsaturated zone in “Vector” site area are quite efficient natural barriers preventing radionuclides transit from the near surface repositories to the ground water.

Key words: aeration zone, sandy soil, radionuclides, sorption, protective properties

Almost all low and intermediate radioactive waste (RW) of Ukraine are planned to be disposed at the Vector site, including radioactive wastes of Chernobyl nuclear power plant (ChNPP), enterprises of Exclusion Zone (EZ), operating NPPs with VVER reactors, and specialized State enterprises of UkrDO “Radon”. According to the approved design solutions, all storage facilities on the Vector site are of the near-surface type.

Safety of storage and disposal of RW in these storage facilities is provided by a system of engineering and natural barriers. The main components of engineering barriers are:

- matrix that contains radionuclides;
- reinforced concrete container and monolithic concrete sections of the storage facility;
- multilayer anti-filtering geomembrane.

Natural barriers, which should prevent potential radiological impact of radionuclides in case of their release from the storage complex Vector are geological and hydrogeological conditions of the site. Protective properties of natural barriers should provide minimization of the flow of radionuclides in natural objects and thus prevent subsequent development of dose loads on the public. Considering the possibility of situations that lead to destruction of engineering barriers, it is necessary to evaluate features of descending migration of radionuclides in rocks of the aeration zone, as a constituent part of the natural barriers on the way to the spread of radionuclides from the surface storage. In this case, unsaturated aeration zone is the only protective barrier between the damaged storage facility and soil aquifer.

Radionuclides migration in the aeration zone is a complicated natural process, the intensity of which depends on many factors, among which one should mention the form of radionuclides, mineral grain size and species. A characteristic feature of the migration process in the unsaturated aeration zone is the presence of various forms of mobility of the water phase. It is distinguished a more mobile gravitational water which is not bound with the surface of particles and can move under the force of gravity, and less mobile capillary water contained in capillary pores of rocks and in cracks whose width is less than 0.25 mm and the diameter of the pores is less than 1 mm. In the case of descending migration of radionuclides in unsaturated aeration zone there is a diffusion of ions from gravitational water to capillary water, which together with the sorption results in reducing the speed of movement of radionuclides. The higher is the radionuclides inhibition during the migration process through the unsaturated aeration zone, the more radionuclides will decay and the greater is the protective potential of a geological system.

Geological structure of aeration zone of the Vector site. The geological structure of the aeration zone involves mid-pleistocene deposits, which are created by fluvioglacial, lacustrine fluvioglacial and moraine deposits. In general these rocks are of glacial complex of the Dnieper glaciation. Fluvioglacial deposits are sandy and clay soils. Sands are light gray, yellowish-gray to brown, and medium small- and middle grain with interbedded silty soils. In the body of the sand at different depths there are lenses of clay soils. Clay soils are yellowish-brown sandy loam and loam. They occur in the form of lenses and layers, their power is mostly 2-6 m. In the area of the Vector one can find a fill-up soil of 0,5-1,9 m thick. In further consideration of the protective properties the soil and vegetation layer and fill-up soil is beyond consideration. According to data of the Kyiv Institute for engineering researches Energoprojekt the mid-pleistocene deposits are divided into several lithological varieties, whose characteristics are shown in Table 1. According to the State Specialized Enterprise “Chernobyl Specialized Complex” the depth of groundwater level on the Vector is 13.25 - 19, 62 m.

Table 1. Generalized characteristics of lithographic types

Soil	Density of dry soil, g/cm^3	Natural humidity, volume fraction	Porosity	Content of fine fracture,% *0,25-0,1 mm* / *0,01-0,005 mm*
Fine sand, dense, non-uniform by particle size, light gray, yellowish gray	1,72	0,06 — 0,20	0,35	55 / 13,1
Silty sand, dense, non-uniform by particle size, light gray, yellowish gray	1,73	0,09 — 0,19	0,35	49,3 / 33,5
Loamy sand, sandy, light brown, yellowish-brown, sometimes with layers of sand	1,78	0,13 — 0,17	0,33	31,6 / 13 *(<0,005-9,2)*
Loam light sandy, gray, brown, reddish-brown, , tight- and soft plasticity	1,77	0,17	0,34	35 / 12,1 *(<0,005-10,9)*
Sand medium size, dense, of homogeneous particle size, yellowish-gray	1,73	0,04 — 0,18	0,35	6,2

Available data make it possible to estimate the time of the water migration through the aeration zone, i.e. the time of descending migration $T_{за}$ during which atmospheric precipitation from the surface soil will reach the groundwater level. This value can be calculated by comparing the volume of water in the aeration zone with the value of infiltration supply typical of the conditions of the Exclusion zone. We assume that thickness of unsaturated aeration zone in the area near the the "Vector" is M = 14 meters. As a result of statistical processing of more than 60 natural values of volume humidity (W) of fine sand that prevail in the geological sections near the "Vector" site we obtained median values equal to W = 0,127. Regarding the value of infiltration supply of aquifer in the mentioned area the authors of [1] note that this issue remains almost unknown. According to the proposed distribution of infiltration supply (ε) in the territory of the zone near Chernobyl [1, Fig. 12.9] this parameter for the area of the "Vector" is ε = 120 mm/year. However, in the recent years it was found [2] based on isotope dating that for the last 60 years the average value of infiltration supply of groundwater is ε = 200 mm/year = 200 liters/(m^2•year) at the first alluvial terrace of the river Pripyat in the zone near Chernobyl. Based on the said above, $T_{за} = M \cdot W \cdot 1000/\varepsilon \approx 9$ years, which indicate high rate of descending migration of water in the unsaturated aeration zone. The only defense mechanism of soil aquifer against radionuclides penetration by hindering the migration processes in the aeration zone are sorption processes in natural minerals.

Sorption processes in unsaturated aeration zone. Numerous studies of sandy soils common in the Eclusion Zone, clearly link their radionuclides sorption properties with presence in the sands of clay minerals (montmorillonite, kaolinite, hydromica, etc.), which are mostly in the dusty (0.05 - 0.005 mm) and clay (<0.005 mm) fractions and therefore have a higher specific surface indicators.

The X-ray diffraction analysis performed for the mineral composition of the clay fraction of loamy sand and sandy loam samples taken from wells at the site of the Centralized Spent Fuel Storage Facility, have shown that, in addition to quartz, it consists of chlorite (water meta-silica-aluminia layered structure), mica, kaolinite, hydrated feldspar and calcite. Experimental determination of sorption capacity indicates that fine-disperse fractions are the components that concentrate and keep the soluble forms of radionuclides, coming to the aeration zone with the infiltrating flow. Short description of the corresponding processes is given in [3]. The experimentally determined values of distribution coefficients for the main dose-forming radionuclides in the Exclusion Zone ^{137}Cs, ^{90}Sr i ^{239}Pu are presented in publications [4-6]. To understand the characteristics of sorption processes in the system "radionuclides-soil" it should be noted that for the sandy soil near the "Vector" site the typical is significant content of fine fractions <0.1 mm (from 13.1 to 51.1%), having enough developed surface with active centers of sorption. At the same time radionuclides in natural water have extremely small mass content – for example ^{90}Sr activity of 1000 Bq/l corresponds to concentration of only $22 \cdot 10^{-4}$ mg/l. No wonder that for low specific activities characteristic is linear adsorption isotherm (Henry isotherms), indicating incomplete filling of active sites on the surface of the mineral components of soil [4]

Given the heterogeneity of grain size and mineral composition of dust and clay fractions of sandy soils in the vicinity of the site "Vector" is not surprising the wide range of the distribution coefficients K_d ^{90}Sr, ^{137}Cs i ^{239}Pu, as defined in [6] at experimental study of radionuclides sorption. It was proposed to use for calculations an averaged minimum median values K_d, which provides a more conservative assessment.

Table 2. Averaged minimum value (median) K_d for sandy soil of the "Vector" site

Radionuclide	Shallow sand	Silty sand	Loamy sand	Light sandy loam	Medium-size sand
^{90}Sr	3,5 [4] 2,65	4,2	7,4	16	2,0
^{137}Cs	180	230	400	370	170
^{239}Pu	100	200	800	950	90

For generalized estimation of protective properties of aeration zone near the site of "Vector" we used available data on its geological structure obtained by geological surveys on the VVER NPP CSFSF site [7] and on the site of centralized repository for pre-storage of spent radiation sources [8], because they are located directly at the site of "Vector" or in close vicinity of it (Figure 1).

With taking into consideration the partial share of each specified in Table 1 lithological types, obtained averaged minimum values (median) of K_d (Table 2) and based on geological sections for 24 wells we calculated integral values for K_d in the aeration zone near the "Vector" site. To provide conservative estimates were made calculations of median values.

Based on these results we can calculate the delay in coming of radionuclides into the soil aquifer due to migration in the aeration zone.

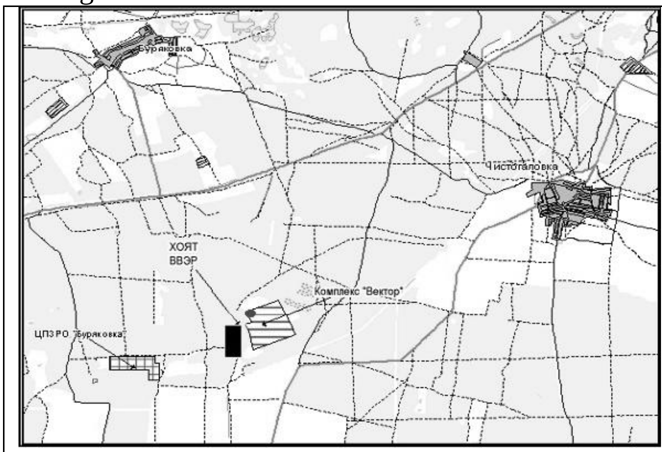

Fig. 1. Map of the sites location

With this purpose, the delay ratio R was calculated, characterizing the degree of reduction of convective and diffusive components of migration in a porous medium in comparison with the speed of the water movement.

$$R = 1 + \frac{\rho \cdot K_d}{n} \qquad (1)$$

where ρ - density of soil in the aeration zone, n — humidity, in volume fractions.

Table 3. Integrated distribution coefficients (median) and the coefficients for delays the aeration zone of the "Vector" site

	^{90}Sr	^{137}Cs	^{239}Pu
Partition coefficient K_d	4,8	246	329
Delay ratio R	66	3320	4450

Time of radionuclide migration through the aeration zone to the ground water level can be found as $T_{рн} = T_{за} \cdot R$ and the corresponding values are 590 years for ^{90}Sr, 30000 years for ^{137}Cs and 40000 years for plutonium isotopes.

Discussion of the results. The results on the time of radionuclides migration through the aeration zone may cause some surprise, as according to the conventional opinion the soil aquifer in the Quaternary sediments in the near Chernobyl zone is not protected against radioactive contamination. However, a comparison with the previously completed expert estimates of the protective properties of the aeration zone give us the grounds to assert that the unsaturated sandy soil of the aeration zone is quite an effective natural barrier against radionuclides penetration in groundwater. Thus, the received by E.A. Yakovlev estimated travel time for water-soluble nuclides for the 5-10 meters thick aeration zone is 65 - 520 years (for ^{90}Sr) [9].

For the aeration zone composed exclusively of shallow sand and medium size sand, an extremely conservative calculation was made of the volume of contaminated water, which can safely filtered through the unit of cross-sectional area of the flow if capacity of the soil layer is 11 meters [6]. These amounts are equal to 17 m^3, 1165 m^3, 632 m^3 for ^{90}Sr, ^{137}Cs and ^{239}Pu respectively.

Re-calculation of these data for M = 14 meters and ε = 200 mm/year = 200 l/(m^2•yr) gives us evaluation of the time of radionuclides migration through the area of aeration to the water level $T_{рн}$, wghich is 108, 7400 and 4000 years, for the corresponding radionuclides. It should be emphasized that these results do not take into account the absorption properties of silty sand, sandy loam and loam, which constitute about 50% of the aeration zone and largely determine the protective properties of unsaturated aeration zone.

Of course, the above data do not include many factors that can affect the rate of radionuclides migration at the site of the “Vector”. One of these factors can be presence at this site of the so-called anomalous zones - depressions, which is marked by an increased rate of radionuclides migration from the surface to the upper aquifer. [10] Another factor, the effect of which has not determined yet, is possible change of hydrochemical regime in the aeration zone due to degradation of reinforced concrete engineering barriers at the stage of the storage facility decommissioning, which will lead to the formation of high concentrations of Ca and Na and to increase of pH of water.

It is known that increase of concentrations of cations Na^+ and Ca^{2+}reduces the effectiveness of sorption of ^{137}Cs and ^{90}Sr due to the competitive impact of these cations [11]. Simultaneously, the alkaline nature of the solutions can positively influence on the improvement of sorption of hydroxyl compounds of plutonium.

Conclusions

1. Taking into consideration the performed calculations it can be noted that the sandy soil of the unsaturated aeration zone near the “Vector” site is quite an effective natural barrier to the penetration of radionuclides in groundwater. Even the most conservative calculations of radionuclide migration through the aeration zone to ground water level gives reason to believe that during this time activity of most mobile radionuclide ^{90}Sr in natural conditions of Polissia will be reduced at least an order of magnitude, and ^{137}Cs due to radioactive decay will not reach the aquifer. The least effective protection is provided by the aeration zone against long-lived ^{239}Pu, which will reduce its activity by only 12 - 68%.
2. Due to the presence of clay minerals, concentrated in fine fractions of sandy soils of the unsaturated aeration zone, the complex “Vector” has considerable potential for safe placement of the conditioned solid radioactive waste generated in the process of decommissioning of Chernobyl NPP, and during operation and decommissioning of VVER units. It is caused by sorption properties of such natural barrier as the aeration zone.
3. Our calculations make it possible to use the delay of radionuclides migration in the aeration zone near the “Vector” site at determining the limits of activity for safe placement of radioactive waste in surface facilities.

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