Nuclear Fuel in the Destroyed 4th Unit of Chernobyl NPP

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Abstract

The main question, which determines the nuclear safety of the 4th destroyed unit of the Chernobyl NPP, as well as the question about the amount and distribution of nuclear fuel inside the "Sarcofagus" is discussed in the paper. The methods of determination of nuclear fuel quantity inside the "Sarcofagus" and the quantity thrown out of its boundaries are considered in detail. Special attention was paid to the quantity and distribution of the fuel in the under-reactor premise 305/2, which is looked as the most nuclear dangerous. On the base of such investigation and also taking into account the results of fuel containing material sample analysis, it is possible to make some calculation of the fuel containing material criticality and scenario of self-sustaining chain reaction development in the hypothetical situation of nuclear danger. Some of the results of such calculation are also presented in the paper.

1. The general description

One of the basic problems of the destroyed 4-th unit of Chernobyl NPP (somebody began to call it as "Shelter" or "Sarcophagus") which substantially defines the nuclear and radiation safety of the object, is the problem of nuclear fuel inside the Shelter. In order to estimate the nuclear and radiation safety of various premises of the Shelter, the knowledge is necessary about the nuclear fuel amount in each premise, the degrees of its primary enrichment and burnup at the moment of accident, the physical properties of the fuel containing materials (FCM), and opportunity of water incoming into each premise of the Shelter.

It is well known (see for example [1]) that at the moment of the accident there were approximately 214,600 kg of nuclear fuel in the fourth unit of ChNPP. The basic amount (190.2 ton) of this fuel was loaded into the reactor core, the part of the spent fuel was placed in the south cooling pond (14.8 ton), at the stand in the central hall (CH) there were the assemblies of fresh fuel (5.5 ton) prepared to loading to the core, and, at last, there was 4.1 ton of fresh nuclear fuel in a room of fresh fuel preparation.

At the moment of the accident, there were 1,659 fuel assemblies in the reactor core, and each assembly contained 0.1147 ton of uranium. The fresh fuel of the reactor RBMK-1000 before the accident at Chernobyl NPP contained 2 % of uranium-235. By the moment of the accident the majority of fuel assemblies were the first fueling assembly with fuel burnup from 11 up to 15 MWt·day/kg U. There were

Table 1. Burnup distribution of fuel assembly (FA).

Group	Number of FA	Average burnup MWt·day/kg U
1	721	13.7
2	392	12.3
3	154	10.5
4	101	8.8
5	35	7.0
6	43	5.4
7	41	3.5
8	172	1.2
FA total number	1659	
Average burnup		10.9

also fresh fuels in the reactor core. The distribution of fuel assemblies by the burnup level of 8-group approximation is given in Table 1.

A rough estimation shows that the burnup of 10 MWt·day/kg U approximately corresponds to a reduction of uranium-235 concentration by 1 % and an increase of plutonium-239 concentration (at the initial stage of campaign) by 0.4 %. Thus, if assuming an average burnup of 10.5 MWt-day/kg U, there were approximately 1,900 kg of uranium-235 and 760 kg of plutonium-239 in the reactor core at the moment of the accident.

Already in 1986 first estimations [2] have shown that as a result of the accident 3 - 5 % of nuclear fuel originally concentrated in the reactor core was thrown out of the 4-th unit boundary. The researches, which have been carried out during the subsequent 15 years, have confirmed these estimations as a whole. Now it is considered [1] that more then 96 % of fuel of the core + fuel in the pool of endurance + fresh fuel of the central hall remain inside the Shelter. 4.1 ton of fresh fuel was removed in 1986 from the fresh fuel preparation room after the accident.

In order to understand the physical properties of the fuel containing materials of the Shelter it is expedient to remind the basic stages of the accident scenario. As a result of reactor runaway with prompt neutrons there was a destruction of pin claddings, and the heated fuel has entered contact to the coolant (water). The explosive formation of water vapors has caused a sharp increase of pressure inside the reactor. This first explosion has resulted that the reactor cover (scheme “E”, see Fig. 1 [1]) was thrown out in the central hall at the height about 14 m, and the reactor bottom plate (scheme “OP”) was lowered approximately by 4 meter into the under-reactor premise 305/2, and the southeast quadrant of the scheme “OP” was completely destroyed. During the flight of the scheme “E” there was the second explosion which has destroyed the reactor building and particularly a drum-separator premise a part of wall of which has appeared inside the reactor vault. After that explosion the reactor cover was lowered in the position shown in Fig. 1.

As a result of the explosions a part of the fuel was thrown out of the limits of the reactor building, and the residual fuel in reactor began to be heated up due to heat release of fission products and burning of graphite. This process proceeded approximately within 10 days. During this time about 14 thousand tons of various materials: lead, dolomite, marble powder, sand, zeolite sorbent, and absorbers of neutrons containing boron were dropped from helicopters to the central hall and the reactor vault.

A part of these materials which has got into the reactor vault was melted together with fuel, pin claddings, walls of technological channel pipes, and material of the scheme “OP” backfilling (serpentinite). These melted materials have penetrated into the under-reactor rooms, whence then they have spread on

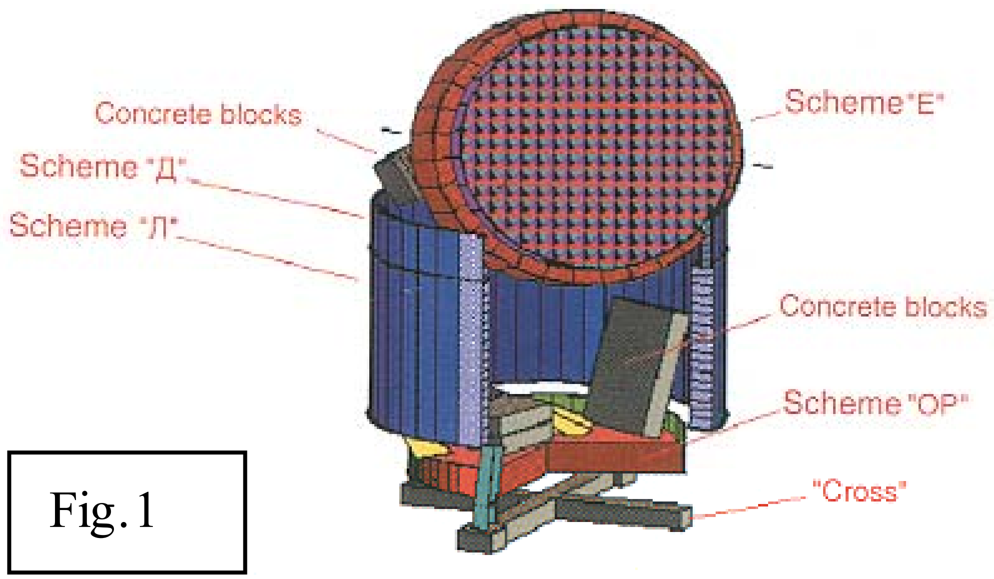

Fig. 1

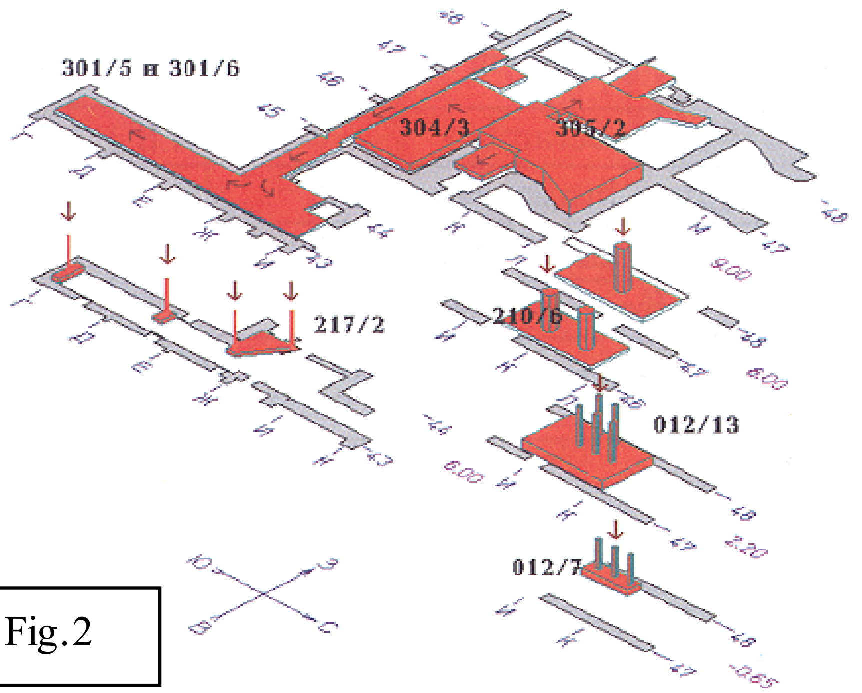

Fig.2

numerous premises of the lower floors of the reactor building (Fig. 2 [1]). In these premises the solidified melt has formed so-called lava-like fuel containing masses (LFCM).

The lava-like FCM of the Shelter represent heterogeneous ceramics of brown or black color with inclusions of various natures. For example, only black ceramics is located in the premise 304/3, and there are both black and brown ceramics in the premise 305/2. The color of black ceramics is caused mainly by radiation defects, and after annealing it gets the bottle-green color which is the characteristics of silica-base glasses. The oxides of iron cause the color of brown ceramics, basically. The average content of different nuclides in ceramics of the premises 304/3 and 305/2 with exception of actinides is submitted in Table 2. A special attention should be given to the boron contents in LFCM. As it is known, boron is "a burning-out absorber", i.e. the quantity of an isotope B-10, which absorbs neutrons, decreases in the media with non-zero neutrons flux. In addition, boron is well-dissolved in water and can be washed away from the porous LFCM. We can not make the exact estimations of the amount of "burnt out" and moreover of the "washed up" boron. Therefore, it is necessary to make criticality calculations with content of boron taken from 10 years old LFCM sample analysis, and also without boron.

Table 2. Chemical composition of the FCMin premises 304/3 and 305/2 and concrete in wt. %.

Chemical element	Mixture		
	FCM, 304/3	FCM, 305/2	Concrete
B	0.06*	0.07*	-
O	43.4	37.1	55.26
Na	4.20	3.34	0.55
Mg	2.40	3.34	0.79
Al	4.80	2.90	2.90
Si	29.8	24.7	26.44
K	1.25	1.05	0.61
Ca	5.50	3.90	8.64
Fe	1.40	0.70	3.64
Zr	3.20	4.00	3.20
C	-	-	0.40
H	-	-	0.77

Let us consider now the LFCM macroscopic properties. According to the data of numerous investigations of samples taken from various under-reactor premises, the density of the LFCM changes over a wide range depending on porosity of the material. In this connection it is necessary to note that the LFCM are a strongly porous material with the sizes of pores and cavities of which change from microscopic dimensions up to the sizes about 1000 - 2000 cc. The LFCM are also a strongly non-uniform material whose density varies depending on the depth within the LFCM: for example, (0.9 - 1.8) g/cc in the premise 304/3 and (1.8 - 3.5) g/cc in 305/2.

At the present time the nuclear fuel in the Shelter is in several modifications. First of all, there are the kept fuel assemblies (the southern cooling pond and the central hall). The fragments of fuel pins and assemblies (core fragments) are found out also in various places. In some places of the under-reactor rooms, non-melted pellets of uranium dioxide were found out. In the LFCM, fuel exists as various inclusions in a silica-base matrix with the sizes from several up to 300 micrometers of various chemical structures [3]. Besides, uranium is also dissolved in a silicate matrix of the LFCM [4]. The concentration of uranium dissolved and included in the LFCM matrix changes from 4 % up to 10 % in various premises of the Shelter, and the mass portion of uranium-235 mainly corresponds to the burnup rate [5,6], though in some samples the portion of uranium-235 was much higher than the average [7,8].

Practically in each premise of the Shelter, the finely dispersed fuel particles (fuel dust) are observed with the sizes of particles from parts of micrometer up to hundreds of micrometers. This dust can represent the main radiation danger in conditions of hypothetical caving of the Shelter structures.

At last, it was revealed in 1990 that in water, which accumulates in some places of the bottom floors of the Shelter, salts of uranium, plutonium and americium are dissolved. The estimations show that up to 4,000 cubic meter of water per one year [7] can penetrate through the holes of the roofing of the Shelter and at the process of moisture condensation from the air. Percolating through fuel containing materials, this water dissolves some salts of uranium and transfers them to the bottom premises of the Shelter.

The nuclear safety of each premise of the Shelter is determined by the amount of fuel in this premise, the geometrical arrangement of this fuel, the opportunity of water ingress in this premise and its penetrations inside the fuel containing materials.

2. Estimation of fuel quantity in the Shelter premises and thrown out of its boundaries.

An estimation of the thrown out fuel quantity (3 ± 1.5) % offered in 1986 in the report of the Soviet delegation at IAEA meeting [2] has caused the large doubts. These doubts, which bound up basically with plentiful release of radioactive iodine-131 and caesium-137, were expressed as a rule by the nonprofessionals who did not take into account volatility of some components of the spent nuclear fuel.

It is possible basically to estimate the emission of nuclear fuel by three different ways [9]:

1. Measuring the quantity and the content of activity thrown out into the environment directly during the active stage of the accident;
2. Measuring the density of radionuclide pollution of the territory both directly adjacent to the Shelter and in the remote areas;
3. Determination of the fuel quantity in various premises of the Shelter. Then the knowledge of the total fuel load of reactor make it possible to estimate the quantity of the thrown out fuel by the difference.

It is natural that the most exact estimation of the thrown out fuel amount can be obtained combining all three methods.

The measurement of activity and contents of emission directly during the active stage of the accident was connected to the large methodical difficulties of aerosol sampling above the damaged reactor. These difficulties have resulted in the large enough errors (50 %) in determination of radioactive aerosol concentration in emission. Therefore, the first estimations of the thrown out activity were rather approached. Table 3 contains the results of researches published in work [2].

Table 3. Radioactivity ejection from the 4-th unit (in % to the activity accumulated in reactor to the moment of the accident).

Isotope	Ejection,%	Isotope	Ejection,%
^{133}Xe	~100	^{141}Ce	2.3
^{85m}Kr	~100	^{144}Ce	2.8
^{85}Kr	~100	^{89}Sr	4.0
^{131}I	20	^{90}Sr	4.0
^{132}Te	15	^{239}Np	3.2
^{134}Cs	10	^{238}Pu	3.0
^{137}Cs	13	^{239}Pu	3.0
^{99}Mo	2.3	^{240}Pu	3.0
^{95}Zr	3.2	^{241}Pu	3.0
^{103}Ru	2.9	^{242}Pu	3.0
^{106}Ru	2.9	^{242}Cm	3.0
^{140}Ba	5.6		

The second way requires an estimation of the radioactive pollution of the large territories in the different countries and is very labour consuming. However, since the remote territories became polluted mainly by the volatile radionuclides (iodine, tellurium, caesium), and finely dispersed fuel particles containing heavy transuranium elements accumulated in the majority within the limits of a 30-kilometer zone around the Shelter, then it is possible to make an exact enough estimation of fuel emission by having carefully performed pollution investigations of the zone of alienation. In addition to this, undoubtedly, it is necessary to estimate a degree of reduction of the pollution by transuranium elements with increase of distance from a source of emission.

Such estimations were executed in 1986 by the group of researchers of Kurchatov Institute, and they were continuously became more precise during all 15 years past after the accident [10,11,12]. These estimations once again confirm a conclusion of work [1]: more than 95 % of fuel from the destroyed reactor core is concentrated in the Shelter.

3. The distribution of nuclear fuel on the Shelter rooms.

It seems that the third way of determination of the nuclear fuel amount which has been thrown out from the Shelter, i.e. the determination of the amount of fuel located in various premises of the Shelter, is exactest and accessible. However, a plenty enough of reasons exist obstructing to the detailed inspection of the Shelter. It is possible to attribute such handicaps to the followings: high radiation fields in the Shelter premises, the blockages of various materials dropped from helicopters to the central hall, the overflows of stiffened "fresh" concrete (1986) on the LFCM congestion and the large thickness of the LFCM layer in the premise 305/2, where the basic LFCM congestion is located.

The determination of the nuclear fuel quantity inside the Shelter and its distribution on premises is very important also from the point of view of nuclear safety of the Shelter and its radiation influence on the ChNPP personnel and the environment in the case of possible emergencies. Therefore below, we shall consider in detail the items of information about the distribution of nuclear fuel in the Shelter premises obtained up to the present time.

Let's begin from the central hall (CH) - one of the most complex places of the Shelter premises for an estimation of fuel quantity. Let's remind that in the CH before the accident, there were 5.5 ton of fresh fuel prepared for loading into the reactor core and 14.8 ton of the spent fuel in the southern cooling pond. The fuel pins are also in so-called “Helen hair” – the rests of technological channels which are hanging down from the scheme “E”. Besides, in various places of the CH there can be fuel dispersed by the second explosion. It is possible only to use the indirect methods to estimate the quantity of the dispersed fuel, since the materials dropped from helicopters during the active stage of the accident cover the CH. The estimation of fuel quantity in “Helen hair” also can be carried out by indirect methods in connection with high radiation fields in the CH.

Scientific groups from Khlopin Radium Institute and Kurchatov Institute carried out such estimations on the basis of measurement of radiation dose rate and localization of its sources in 1992. The estimations of such type can be only qualitative in connection with impossibility of exact localization of radiation sources. The results of calculations based on measurements of such type give that the quantity of fuel on the scheme “E” can be in limits from 10 up to 30 ton. There was found also up to 1 ton of fuel on the walls of the CH and other structures. Totally together with the spent fuel of cooling pond (the periscope inspection of which have shown the absence of water and the presence of all nondestroyed fuel assemblies) there can be from 31 up to 51 ton of fuel in the CH.

On the basis of video and periscope inspections of the reactor vault, it is possible to make a conclusion that there are no ordered structures of the former reactor in the shaft and now it is communicated with the under-reactor premise 305/2. Therefore, it is meaningful to estimate the quantity of fuel in the reactor shaft together with a premise 305/2.

Already in 1986 – 89, the first thermometric measurements of fuel quantity in the under-reactor premises were performed. These measurements were based on the fact that the integrated thermal flow outgoing from these premises completely should be determined by the thermal source power, and consequently by the complete mass of the fuel [13]. The specified estimations of 1990 [14] taking into account an error of measurements give for fuel mass in the premise 305/2 the value of 75±25 ton.

It is possible also to estimate the fuel quantity in lavas of the under-reactor premises by balance of caesium and magnesium [3]. The last specified data [11,12] show that the complete emission of caesium-137 has made an activity about 2 MCi that forms about 28 % from the value of 7 MCi initially accumulated in the reactor core at the moment of the accident (compare with initial assessments of works [2], Table 3). This caesium took off only from the fuel, which has melted during the active stage of the accident and has formed the lava-like FCM.

On the other hand, the data of the numerous analyses show that no more than 40 % of caesium-137 remained in lava. It means that about 60 % of caesium-137 from its initial quantity in fuel, which has formed lava, has taken off, and these 60 % of caesium-137 have given the activity of 2 MCi. Therefore the initial activity of the caesium-137, which contained in the fuel forming a lava, is equal to (2/0.6) =3.3 MCi. It makes (3.3/7) =47 % from the amount originally accumulated in reactor, so 47 % of reactor fuel loading (190.2×0.47) =89.39 ton has come in a lava. Taking into account an error of estimations, we have 90±27 ton.

The similar estimations can be made by the amount of magnesium whose average concentration in lava is equal to 3 % (compare with Table 2). The magnesium enters only into the structure of serprntinite, by which the scheme “OP” was filled up and its contents in serpentinite are 25.1 %. During the formation of lava, about 140 ton of serpentinite were melted [3]. Taking into account the percentage of magnesium in serpentinite and in the samples of a lava, it is possible to say that during the melting in the process of a lava dilution by other materials its weight has increased in (25.1/3) =8.5 times and has reached approximately 1200 ton. The average content of uranium in lava samples is about 7 % [3]. Therefore the lower estimation of its weight in lava of the under-reactor premises makes about (1200×0.07) =84 ton. Taking into account the possible error, we have 80±24 ton. All three estimations are in good coincidence among themselves.

It is necessary to note that at these estimations the presence of reactor core fragments is not taken into account, i.e. the fragments of pins and fuel assemblies and also non-melted fuel pellets which were observed in the under-reactor premises, and also can be under the melt in connection with their large densities. Therefore, and also with other reasons, it is necessary to consider the estimation of fuel quantity by the amount of caesium and magnesium to be underestimated.

Nevertheless, there was a work [15] in 1992 in which these estimations are put under doubt. The

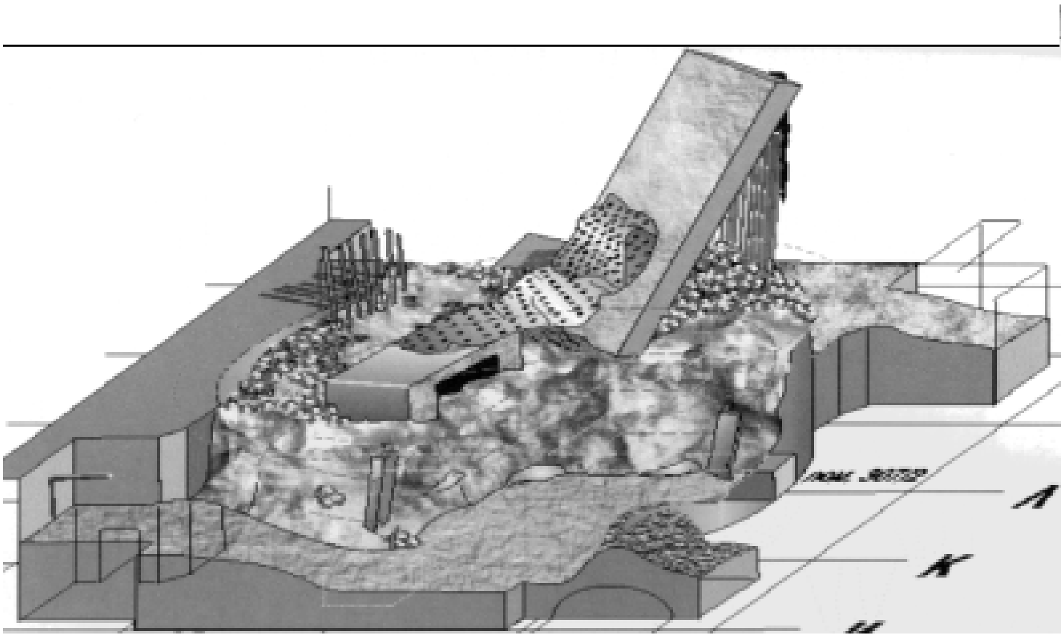

Fig.3 Computer model of the premise 305/2 and the reactor shaft.

authors of the paper [15] used for the estimations only the data of visual inspection and principle of communicated vessels. In their opinion, the under-reactor premises contain no more than 25±5 ton of uranium.

This work was fairly criticized by many authors. However, it has served as a certain stimulus for new careful auditing of all data on the fuel quantity in the premise 305/2 [16]. For this auditing, the authors used the results of the analysis of more than hundred samples taken from the premise 305/2 in 1986-1997, the measurements of dose rate of γ - radiation, the results of all video- and photography of the premise 305/2 and the reactor shaft.

Then the whole space of the premise 305/2 was separated by the squares with a cross section 2×2 m. The estimation of the fuel quantity was performed in the LFCM volume over every square taking into account all data mentioned above. The square which data were absent or even partly was considered as empty that certainly underestimated the results. The computer model of the premise 305/2 and the reactor shaft was composed on the base of such detailed analysis of all materials which are located in this premise (Fig.3).

The precise consideration of all data concerning the premise 305/2 and the reactor shaft allows to make the following conclusion: there are not less than 60 ton of fuel in these premises. The LFCM passed from the premise 305/2 through the concrete wall destroyed by the explosion (or burns by the LFCM?) into the room 304/3 and then to steam distribution corridors (see Fig.2). The second flow of the heated lava passed through the steam outlet valves of different floors even to the first floor of the pool bubbler. The estimations of fuel quantity in all these premises raise no doubts. They are presented in Table 4 where the data of all measurements and estimations of fuel amount in the Shelter are summarized [1].

As we can see from this Table, the premise 305/2 and the central hall are the most “suspicious” from the point of view of nuclear safety. At the steam distribution corridors with marks 9 and 6 m, the lava layers are thin (the maximum value of 0.6 m in the premise 304/3), and in spite of sufficiently large fuel amount the probability of self sustaining chain reaction (SCR) ignition in these premises is negligible according to some evaluation.

In the southern cooling pond of the spent fuel where the water is absent at the present time the situation is safe until the pitch of the assembly suspension is conserved. The danger can arise only in the case of assembly caving to the bottom of the pond and its affluxion by water.

The water in the lowest premises of the Shelter has no nuclear danger at the present time. However,

Table 4. FCM distribution inside the Shelter.

Premises (mark)	FCM type and state	Estimated fuel in FCM (on uranium basis in metric tons)
- The central hall (35.50) - Other upper floor premises	- Core fragments (most of them are buried under materials dropped at the active stage of the accident. Under them LFCM can be found) - Fuel dust. - Fresh fuel assemblies - In the area of the scheme "E"	? 30? 5.5 10-30
- The south cooling pond (18.00-35.50)	- Fuel assemblies with the spent fuel	~14.8
- The under-reactor premises: 305/2 (9.00) + 307/2 + the "OR" system + the reactor vault	- Lava-like FCM, core fragments	75 (+25,-35) was proved to be >60 t.
- 304/3, 303/3, 301/5, 301/6, the "elephant's foot " and others	- LFCM	11±5
- Stream-distribution corridor (SDC) (6.00), including FCM in the valves.	- LFCM	25±11
- Pool bubbler, 2d floor (PB-2)	- LFCM	8±3
- Pool bubbler, 1st floor (PB-1)	- LFCM	1.5±0.7
- Lower premises of the reactor unit,	- Water with dissolved salts of uranium	~3000 m^3 of water <3 kg U

as far as the concentration of uranium salts will increase in the future the risk of the SCR ignition can also increase.

In order to estimate the nuclear safety of various premises of the Shelter two different ways exist. The first way is calculation of the neutron multiplication factor in these premises in different conditions including the most unfavorable condition of FCM water affluxion. The criticality calculation should be supplemented with calculation of SCR scenario development in the case if the criticality calculation shows a possibility of the multiplication factor exceeding of unity.

The second way consists in direct measurements of the neutron multiplication factor in the Shelter premises, and possible organization at this base of the continual monitoring of the FCM reactivity. The most expedient way is a creation of the mutual experimental and calculational method due to the absence of the full data to both precise criticality calculation and interpretation of experimental measurements.

4. The evaluation of the nuclear safety of different Shelter premises and possible scenario of SCR development.

The nuclear safety of the “Shelter”, which is actually the question of criticality of the fuel-containing masses, has been considered by several groups of researchers [17-19]. Most of these calculations used the models whose properties are rather far from that of the real LFCM.

A model of the LFCM, closest to reality, was used in [18]. The LFCM in this paper were simulated by a multilayer system with variation of the LFCM density by layers in a rectangular geometry approximately corresponding to the layout of the rooms 304/3 and 305/2. The neutron reflection from concrete walls and floor was also taken into account. The nuclide composition of the LFCM and fuel concentration was chosen closest to the experimental data that were obtained during a study of the LFCM composition. The calculations show a deep subcriticality of both dry LFCM and LFCM filled by water. Nevertheless, the question of LFCM criticality remains open because in this work the existence of the core

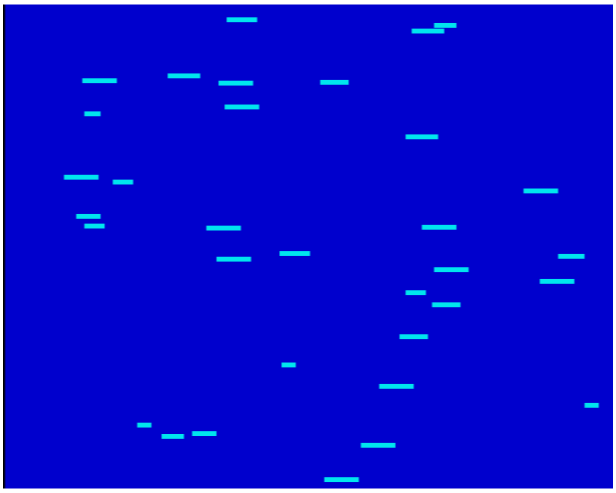

Fig. 4. Vertical cross section of rectangular parallelepiped of FCM with fuel pellets randomly distributed over its volume (model 1d).

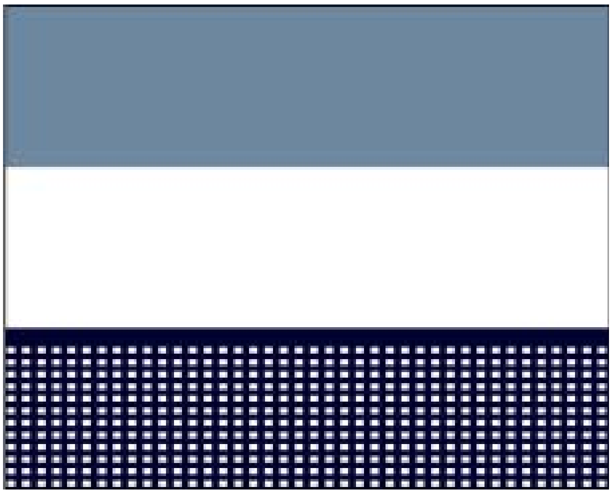

Fig. 5. Vertical cross section of rectangular parallelepiped of FCM: three layer model of FCM, limited by a layer of concrete from below with nonmelted part of fuel in bottom sublayer as a cubic lattice of pellets (model 3b).

fragments in the premise 305/2 did not take into account.

What is it necessary to know for realization of criticality calculations?

1. Nuclide composition of the FCM and concrete, on which the FCM are located and also which covers the FCM in some places. This concrete can play a role of a reflector of neutrons.
2. Macroscopic properties of the FCM, such as density and porosity.
3. Geometrical parameters of the FCM – the size and shape of FCM accumulations.
4. Quantity and spatial distribution of fuel – this is the main question.

Nuclide composition of the FCM and concrete and also macroscopic properties of the FCM are known rather well from the results of the numerous analyses of the FCM samples. The geometrical parameters of the FCM in the under-reactor premises are known at the present time with sufficient accuracy due to investigation [16]. However, the question on quantity and spatial distribution of fuel in the FCM remains open in many details, especially in distribution of non-melted core fragment and fuel pellets, and for realization of real calculations it is required to involve the additional assumption about fuel distribution.

To determine the greatest possible value of the effective multiplication factor the various models of the FCM in the premise 305/2 were considered [20]. The schematic image of some of these models is given in Figures 4 and 5.

The enrichment of fuel was determined on the basis of the average fuel burnup of the order of 11.5 MWt·day/kgU [3]. As there was also the fresh fuel in the reactor before the accident, some models of the FCM were calculated with enrichment of fresh fuel. The total amount of fuel in the FCM was defined by both the average and the top values given by thermal measurements. The results of calculations are given at Figures 6-7 and in Table 5. All values were calculated depending on the contents of water in the FCM. In Figure 6, the neutron flux density spatial distribution is presented taking into account the presence of upper concrete reflector (ceiling). The analysis of such figures can help to determine the places of neutron detector installation in the FCM accumulations. In Figure 7, the neutron energy spectrum inside the FCM is given depending on a degree of FCM filling by water. In Table 5 the dependencies of the effective multiplication factor on a degree of FCM filling by water are given.

As it is evident from Table 5, the account of FCM heterogeneity not always results in increase of the effective multiplication factor. The effect depends on water concentration that is influencing on slowingdown properties of the medium. The results of calculations show that the value of the effective multiplication factor weakly depends on the accepted model of an arrangement of non-melted inclusions, which allows to use for the majority of calculations the lattice models. For all models with the average amount of fuel (Model 3-5), the effective multiplication factor quickly increases from values 0.25-0.35 for dry FCM up to 0.65-0.70 at 20 % filling by water. At the further filling FCM by water, the effective multiplication factor decreases up to 0.60-0.65 for first two models of the FCM (Model 3, 4) and slowly increases up to value 0.8 for the last model of the FCM with non-melted inclusions in the bottom sublayer (Model 5). For infinite medium with a maximum quantity of fuel and the average burnup with a cubic lattice of pellets (Model 1f), the multiplication factor reaches 0.87 at 20 %-filling by water. The same model with fresh fuel (Model 1g) gives $k_{\infty} \approx 0.99$ at 20 %-filling by water, which is increased till 1.07 at 40 %-filling by water.

Thus, some models of fuel containing masses of the "Shelter" give the FCM multiplication factor exceeding unity. Our recent calculations show that there can be other models with the multiplication factor exceeding unity even with the spent fuel but with the bigger thickness of the layers. However, these values of the multiplication factor are reached only when enough quantity of water is contained in FCM volume. It means that during affluxion of the FCM by water from any external sources (rains, the condensation) the self-sustaining chain reaction of fuel nucleuses can arise inside the FCM.

The dynamics of ignition and development of SCR in FCM in various conditions of filling FCM by water and at various values of greatest possible reactivity, i.e. maximal effective multiplication factor, taking into account the Doppler-effect, is analyzed below. It is shown that, depending on the speed of FCM filling by water, the various modes of SCR development can be realized.

We shall assume that the heterogeneous composition with greatest possible effective multiplication factor exceeding unity is realized in the FCM (both the presence of such composition and an opportunity of its affluxion by water up to achievement of maximal reactivity are rather problematic), and we shall consider a qualitative picture of SCR development.

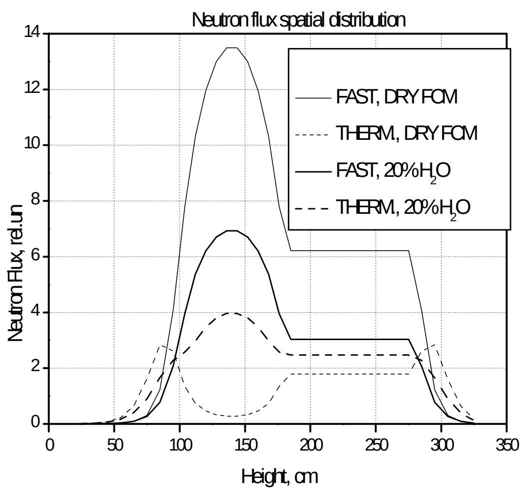

Fig. 6. A fast and thermal part of a neutron flux in a premise 305/2 in dependence on height for model 2a taking into account the presence of a concrete ceiling at height 1м from a FCM surface.
(0-100 cm – concrete, 100-180cm – FCM, 180-280 cm – air, >280cm – concrete)

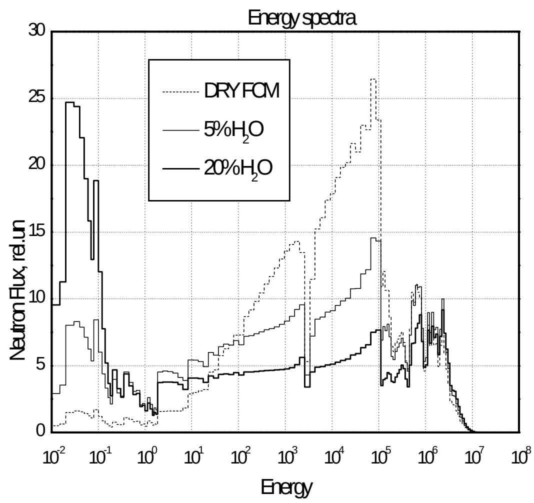

Fig. 7. A neutron energy spectrum in a premise 305/2.

Table 5. Results of the effective multiplication factor calculations for different models of the FCM in the room 305/2

Model of the FCM	Submodel	Effective multiplication factor		
		Dry FCM	FCM+ 20% H_2O	FCM+ 40% H_2O
1. Model of the FCM in the form of infinite medium	Completely homogenized fuel	0.26	0.76	0.70
	1a. cubic lattice of pellets in the uniform FCM	0.28	0.75	0.68
	1b. square lattice of pins in the uniform FCM	0.29	0.75	0.67
	1c. repeating cubes of the FCM, containing lattice of pellets	0.28	0.65	0.70
	1d. random distribution of pellets	0.28	0.65	0.70
	1e. cubic lattice of pellets in the uniform FCM – fresh fuel	0.32	0.81	0.73
1. Model of the FCM in the form of infinite medium (maximal quantity of the fuel)	Completely homogenized fuel	0.30	0.85	0.90
	1f. cubic lattice of pellets in the uniform FCM	0.32	0.87	0.96
	1g. cubic lattice of pellets in the uniform FCM –fresh fuel	0.40	0.99	1.073
2. Model of the FCM in the form of the uniform layer with vacuum boundary conditions	Completely homogenized fuel	0.11	0.64	0.64
	2a. cubic lattice of pellets in the uniform FCM	0.11	0.63	0.62
	2b. square lattice of pins in the uniform FCM	0.12	0.63	0.62
	2c. random distribution of pellets	0.11	0.63	0.61
3. Model of the FCM in the form of the uniform along height layer, located over the concrete layer	Completely homogenized fuel	0.25	0.66	0.65
	3a. cubic lattice of pellets in the uniform FCM	0.25	0.65	0.63
	3b. square lattice of pins in the uniform FCM	0.26	0.65	0.62
4. Three-layer model of the FCM with sublayer density dependence on the height, located over the concrete layer	Completely homogenized fuel of sublayers	0.25	0.66	0.66
	4b. cubic lattice of pellets in the uniform FCM of sublayers	0.26	0.65	0.63
	4c. square lattice of pins in the uniform FCM of sublayers	0.25	0.65	0.62
5. Three-layer model of the FCM, located over the concrete layer with nonmelted part of the fuel in the lower sublayer	Completely homogenized fuel of sublayers	0.32	0.73	0.78
	5b. cubic lattice of pellets in the uniform FCM of sublayers	0.33	0.67	0.76
	5c. square lattice of pins in the uniform FCM of sublayers	0.34	0.68	0.74

In order to analyze the FCM behavior in conditions of its reactivity changing due to filling FCM by water, it is necessary to solve the system of kinetic equations describing the situation. First of all, this is the equation of neutron transfer in the FCM medium, which should take into account the change in mean neutron lifetime due to the presence of delayed neutrons and dependence of reactivity on water quantity into the FCM and its temperature. The steady state calculations show that the dependence of reactivity on water quantity seems to be quadratic, which represents the moderating and absorbing properties of the water.

The second equation actually is the law of energy conservation, which takes into account fission energy release, the heating of the FCM, heat removing from the FCM surface and water evaporation. The third equation represents the law of water mass conservation, which takes into account the water incoming from the external source and its evaporation due to fission heating.

These equations in the frame of point model were investigated by qualitative stability methods and were solved by numerical methods [20]. The mentioned above different modes of SCR development depending on the parameter values were found. These modes are: the single neutron burst both in subcritical and overcritical regimes, the damped and stable neutron oscillations. The realization of one of

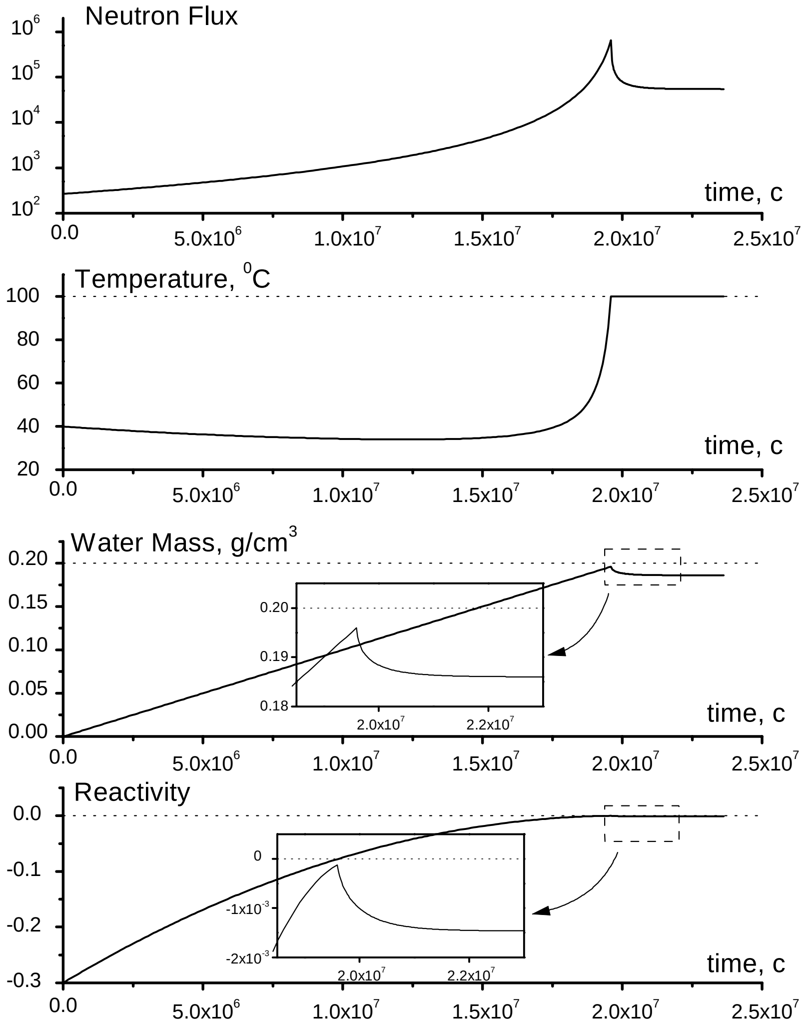

Fig. 8. Mode of an aperiodic subcritical burst.

these modes depends on the value of the neutron multiplication factor, on the velocity of FCM filling by the water, on the rate of heat removing etc. One of the most probable regimes of the subcritical neutron burst is presented in Fig. 8.

It is necessary to note that in 1990 and 1996 there were two neutron bursts registered by the neutron detectors located in the premises 304/3 and 305/2. There were a lot of discussions on the causes of these bursts, but it is evident that in both cases they occurred after the intensive rains in Chernobyl region. The pictures of burst development were very similar to that presented in Fig. 8.

5. The measurement of the FCM reactivity in the Shelter premises

Let us emphasize once more that the nuclear safety of the Shelter is defined first of all by the value of the neutron multiplication factor in the FCM of premises 305/2, 304/3, 210/6, the central hall, the reactor shaft and the southern cooling pond. Since the space distribution of the fuel in those places and the geometry of the FCM also are known inaccurately, the neutron calculation in those premises could be performed with the help of not very realistic models giving only not exact evaluations of the multiplication factor. In order to make the well-founded decision to increase the nuclear safety of the Shelter, those evaluations were inadequate. Therefore, a necessity arisen to measure the multiplication factor in those premises.

It is necessary to note that the attempts of such measurements in the Shelter were made in 1991 [21]. Those measurements were performed on the base of neutron pulse method, and authors have used the standard geophysical equipment of neutron logging of the bores. They have no possibility to vary the frequency of neutron pulse emission, and they have small time of registration (2 ms) of the system response. Therefore, they could not detect the delayed neutrons and determine the multiplication factor with a sufficient precision.

One can conditionally divide the experimental methods of reactivity measurement on two classes: active and passive methods. The active methods assume some impact to multiplying system and subsequent measurement of the system response. One can realize the active methods both by the introduction into the system of the subsidiary reactivity (positive or negative) or with the influence by the external neutron source.

The methods connected with the introduction of additional reactivity are not suitable to measure the neutron multiplication factor in the FCM of the Shelter. This is, first of all, due to impossibility to estimate the value of injected reactivity even with known physical parameters of the injected materials. The cause of such situation is the same that leads to impossibility of exact calculation of the neutron multiplication factor. The pulse neutron method is the most widely distributed among the methods of external neutron source influence. But it is necessary to insert the pulse neutron source inside the FCM and to have a good electronic equipment in order to obtain the precise result. This method needs the boring of cavities inside the LFCM that is impossible at the present time.

Contrarily, the passive methods of reactivity determination are based on the measurement and analysis of the fluctuations of steady neutron background which is caused by the buildup of transuranium elements during the reactor operation or by the stationary external neutron source. By now in the FCM of the Shelter, the steady neutron background is determined by the spontaneous fission of some transuranics and (α,n)-reactions, and it's fluctuations are due to statistical (probabilistic) nature of those processes.

By now there are several monitoring systems of the FCM neutron characteristics in operation at the Shelter («Finish», «Pilot», «KSFCM»). The current measurements of neutron flux density in the locations of neutron detectors are carried out with the help of these systems. The indications of neutron detectors are averaged over some intervals of time, considerably larger than the neutron lifetime. So there are measurements of the average on-time neutron flux density.

The indications of existing FCM monitoring systems can give the information only about the tendencies of neutron flux density changes, nothing speaking about the real value of such important characteristic of nuclear material accumulations as the effective multiplication factor k_{eff}. The knowledge of k_{eff} and keeping it at the certain level is one of the requirements of regulating bodies to organization maintaining, storing or supervising of the nuclear material accumulations.

It should be noted that the measurement of the neutron statistical characteristics can give much more information on nuclear material accumulation parameters than simple establishment of the tendency to reduction or growth of the neutron flux density. The methods of the analysis of the neutron flux density fluctuations and definition on their basis of the nuclear critical characteristics are conditionally divided into two parts: statistical discrete methods and methods of the neutron noise analysis. Both methods are basically possible to use for the subcriticality control of nuclear material congestion in the Shelter. The condition of modern electronics allows joining practically all methods in one experimental device executed on the basis of personal computer, which provides processing of the information according to algorithms determined by theoretical approaches. By now, such device is manufactured, and it is planned to use it in the near future to measure the effective multiplication factor in FCM of the Shelter.

6.Conclusion

As it is evident from the above that the Shelter at the present time is the nuclear dangerous object, it is necessary to make the significant efforts in order to define the degree of the danger and then to make the Shelter ecologically safe.

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