Problems and Experience of Ensuring Nuclear Safety in NPP Spent Fuel Storage Facilities in Russia

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The amount of Nuclear Power Plant (NPP) spent fuel in special storage facilities of Russia runs to more than 15000 tons and the annual growth is equal to about 850 tons. The storage facilities for spent nuclear fuel from the main nuclear reactors of Russia (RBMK-1000, VVER-440, VVER-1000, BN-600, EGP-6) were designed in the 60s – 70s. In the last years when the concept of closed fuel cycle and safety requirements had changed, the need was generated to have the nuclear storage facilities more crowded. First of all it is due to the necessity to increase the storage capacity because the RBMK-1000, VVER-1000, EGP-6 fuel is not reprocessed. So there comes the need for the facilities of a bigger capacity which meet the current safety requirements.

The paper presents the results of studies of the most important nuclear safety issues, in particular:

- development of regulatory requirements;
- analysis of design-basis and beyond-the design-basis accidents (DBA and BDBA);
- computation code development and verification;
- justification of nuclear safety when water density goes down;
- the use of burn-up fraction values;
- the necessity and possibility to experimentally study the storage facility subcriticality;
- development of storage norms and rules for new types of fuel assemblies with mixed fuel and burnable poison.

KEYWORDS: spent fuel, storage facilities, safety requirements,

1. Regulatory requirements. Design and beyond-the-design-basis accidents

In 1991 the SSC RF IPPE developed the safety rules of NPP nuclear fuel[1] with formulation of technical requirements to various safety systems and nuclear safety analysis as well as with an approximate list of DBAs and BDBAs. These rules were based on the safety requirements experience reflected in the IAEA document № 50-SG-D10 and IAEA report № 189. The nuclear safety requirements formulated in[1] are based on three main principles:

- k_{eff} values must not exceed 0.95 under normal conditions and initial DBA events (principle of normalization);
- probability of spontaneous chain reaction in the storage facility must not exceed 10^{-7} per year; BDBA consequences in the storage facility must be limited (principle of justification);
- norms of storage and transportation within the plant must provide k_{eff} control at the minimum possible level (principle of conservatism).

Among 19 approximate initial DBA events listed in the rules there is an indication of the following: natural phenomena, external effects, black-out, various types of FA and equipment drop, formation of highly explosive mixtures as a result of water radiolysis, etc.

The following BDBAs should be considered:

- spontaneous chain reaction initiation;
- complete dry-out of storage pools filled with water;
- process equipment drop onto the fuel being stored.

When the initial events are analyzed it is necessary to consider a possible change in water density in the storage pool, fuel pin and fuel assembly geometry, fuel pin spacing and thus, increase in the k_{eff} value.

2. Computation code selection and verification

The codes developed for reactors are usually used to calculate storage facilities. The codes being used for calculating K_{eff} of storage facilities should be applicable for the list of emergency conditions given above.

The significant difference between spent fuel storage facilities and reactors consists in the fact that moderation processes take place both inside FAs and between them because gaps between FAs can have the size of 50 mm – 200 mm. Due to hydrogen scattering anisotropy a great number of fast neutrons find their way to the gap. The value of thermal neutron flux density in the gap is much higher than that inside FAs. The use of neutron poison in the racks of the crowded storage results in significant jumps of neutron flux density near the interface. Consideration of water density variation reveals various neutron energy spectra: fast, intermediate and thermal. Substantial non-uniformity of neutron flux density makes it necessary to use the codes which allow the accurate geometrical modeling. So the current codes which are based on the Monte – Carlo method with the use of neutron and physical constants obtained from the estimated nuclear data, e.g. ENDF/B-6, are widely used for storage calculation. These codes are MCU-RFFI[2], MCNP[3], SCALE[4], MMKFK-2[5]. However in some cases, when the problem is not geometrically symmetrical and the storage dimensions are quite big, these codes are not capable of calculating the entire system with due regard for heterogeneity inside FAs. In order to calculate these situations small – group (usually 4-group) macrocostants obtained by the software of the WIMS-D4 type are used. Quite a lot of critical experiments are available to verify the computation codes for storage facilities with FAs or fuel pins of PWR, BWR, VVER types. However there are actually no critical experiments for storages with other FA types. Under these conditions the only possible way to determine the uncertainty is to perform test calculations with various codes which have been already verified. The comparative calculations of k_{eff} show a good agreement of above mentioned codes within 0.5 % for k_{eff}. A little higher discrepancy is observed when water density is lower than 0.05 g/cm^3.

The calculation quality program is strongly required as it is very important and presents a crucial problem. Insufficient study of emergency scenarios, various types of errors during calculations, insufficient personnel qualification can result in serious mistakes during the design work.

3. Justification of nuclear safety at decreasing water density in the storage pool

The emergency situation analysis has revealed some general problems for storage facilities. First of all, it is the possible increase in k_{eff} due to decrease in water density in the entire storage volume or in its separate areas.

Table 1 shows the calculation results obtained with various computation codes for the test task, i.e. the storage cell for VVER-1000 assemblies with UO_2 fuel with the enrichment of 4.4 %, shown in Fig. 1. The cell consists of two parts, i.e. FA and water. The external area consists in a hexagon with a flat – to – flat distance of 400 mm.

The maximum k_{eff} value in the storage is observed when the water density is ~ 0.05÷0.3 g/cm^3 and depends on FA spacing. At the presence of boric steel poison in the storage the k_{eff} value decreases when the water density goes down. When stainless steel wrappers are used with the thickness of 3–5 mm the k_{eff} value usually increases insignificantly with water density decrease. With the thickness of 8 mm the absorption property of stainless steel is comparable with that of boric steel, with boron content of 1.1 % and its thickness of 5 mm. The water density variation in the storage during a long black-out and boiling was studied experimentally.

At the "Hydropress" special thermal hydraulic experiments were carried out to determine the coolant density along the FA height in the storage cell for spent fuel with FA spacing of 400x400 mm, at the VVER-1000 NPP. As it turned out, in case of boiling the coolant density in the gap between FAs was equal to 0.9–0.96 g/cm^3, inside the FA at the outlet it was no less than 0.45 g/cm^3. Besides, the situations were studied with additional air injection in the boiling mode, thus simulating a break in the scheduled work, when the cooling system actuation can be interrupted because the area hasn't been completely drained. This fact results in air bubbles rising to the surface and accumulation of steam – air voids in separate areas of the cooling pond during a limited time. Thermal hydraulic calculations showed that the k_{eff} value did not exceed 0.95 in the indicated emergency conditions. The similar situations were calculated for the storage pools near the RBMK reactors. It has been determined that in case of crowded storage of FAs without tubes the water density value will be no less than 0.8 g/cm^3 during boiling if FAs have been cooled at least 8–15 days before being crowded. The k_{eff} value at this density will not exceed 0.95.

Table 1 k_∞ for a VVER-1000 storage cell.

Water density, g/cm^3	Computation code			
	MMKFK-2	MCU	MCNP	KENO
0,00	0,7060	0,6916	0,6989	0,6931
0,02	1,0190	1,0183	0,9963	1,0382
0,03	1,1210	1,1259	1,0994	1,1400
0,05	1,2620	1,2645	1,2547	1,2798
0,10	1,3780	1,3791	1,3653	1,3895
0,20	1,3100	1,3071	1,3016	1,3112
0,30	1,1560	1,1559	1,1523	1,1503
0,40	1,0290	1,0246	1,0230	1,0234
0,50	0,9410	0,9447	0,9399	0,9449
0,60	0,9040	0,9140	0,8988	0,8956
0,70	0,8930	0,8859	0,8834	0,8876
0,80	0,8960	0,8891	0,8903	0,8937
0,90	0,9150	0,9063	0,9061	0,9109
1,00	0,9340	0,9266	0,9273	09328

4. Problems of crowded fuel storage

As a rule the feasibility and the level up to which the storage pool can be crowded primarity depend on FA spacing, i.e. nuclear safety requirements. These are several ways how to make the pool more crowded, e.g. to decrease FA spacing due to a precise consideration of storage geometry, to use special burnable poison in the form of boric or stainless steel wrappers, to take into account the fuel burn-up fraction as a controllable nuclear safety parameter.

When developing the way how to make the storage pool more crowded with FAs it is necessary to take into account both DBAs and BDBAs. The latter requires the control measures to be developed. For some types of storage facilities where fuel is inside the tubes the situation can happen when in the course of drying-out water remains inside the tubes or boils out of them, thus resulting in a certain increase in k_{eff}. These situations are considered as BDBAs. From the standpoint of nuclear safety the simplest way to control these accidents is to take measures that could rule out criticality even without any special poison, especially as sometimes it is impossible to use it. For instance, for RBMK storage facilities the use of additional poison is restricted by the strength of cantilever beams on which it is suspended as well as by the rules that prohibit to use removable absorbers. So in[6] it was proposed that the spent fuel burn-up fraction should be taken into account in the BDBA analysis, because in practice the spent fuel with a significant burn-up level is being stored. If we consider additional fission product absorption and decrease in fuel enrichment in the course of fuel burning-out in the reactor, the problem can be solved without any introduction of additional technical measures. The burn-up fraction can be also considered in the long-term storage facilities where fuel is delivered from near-the-reactor cooling ponds and where the emergency part of the reactor cannot be discharged. In view of that the concept of nuclear safety for RBMK fuel storage facilities has been formulated[7]. Nuclear safety for normal conditions and initial DBA events must be proven in the assumption of fresh fuel, for BDBAs it should be done with the account of burn-up fraction. A serious problem consists in determining norms of storage and transportation in view of fuel burn-up. When solving this problem it is necessary to consider five main factors:

- calculation error for fuel isotopic composition;
- calculation error for k_{eff} which is determined by a complex fuel composition with the mixture of uranium, plutonium, actinides;
- the effect of accumulated fission products on k_{eff};
- non-uniform distribution of fissile materials along the FA height;
- FA irradiation history.

The analysis shows that nuclear safety requirements for long-term storages of RBMK spent fuel are met under both normal operation conditions and design-basis accidents in the assumption of fresh fuel. At the same time in case of BDBAs, for instance, boiling during 7–8 days and water escape from the storage tubes with FA smaller spacing of 110x115 mm or cooling pond dry-out when water remains inside the tubes, k_{eff} can exceed 1, if there is fresh fuel in the pond. At the same time if the designed fuel burn-up fraction is taken into account (~20 MW·day/kg), subcriticality ($k_{eff} \leq 0.95$) will have a big margin even in case of these BDBAs, i.e. the storage facility acquires the property of self-protection due to the inner properties of the system. The limit of safe burn-up at which $k_{eff} \leq 0.95$ is not exceeded if fuel burn-up is no less than 10 MW·day/kg for initial enrichment of 2.4 % ^{235}U. Due to the fact that along with FAs having designed burn-up in the storage facility there might be defective FAs with their burn-up lower than 10 MW·day/kg, it is necessary to avoid the local concentration of FAs with small burn-up and to place them uniformly among the FAs with burn-up of 15 – 20 MW·day/kg. In accordance with the Rules the burn-up fraction value can be used as a nuclear safety parameter if it is controlled by special instruments in order to avoid mistakes of the personnel or incorrect burn-up calculation. However at present a significant number of FAs have been already accumulated in the RBMK storage. So burn-up measurement which assumes FA movement to the measurement device and back to the storage place will unacceptably increase not only the loading time to the colling pond but also the number of nuclear and radiation dangerous operations. Under these conditions it is reasonable to use methods and tools which allow the storage safety to be experimentally confirmed on the whole. In order to measure the storage k_{eff} the pulse method was proposed. There exist various types of pulse method for measuring k_{eff}. However, as the measurement practice showed, especially in the conditions of profound subcriticality, these methods reveal a strong dependence on spatial effects. So the general relationship between $\mathbf{k_{eff}}$ and decrement of attenuation α is used:

$$\alpha = [(1 - k_{eff}(1 - \beta_{eff}))] / l,$$

where the decrement of attenuation α is an integral parameter that characterizes the system subcriticality on the whole and is independent of the source and detector location; β_{eff} is an effective fraction of delayed neutrons; l is an average prompt neutron lifetime.

The comparison of calculation and experimental results shows that the difference is equal to 5–10 % at $k_{eff} \sim 0{,}5$–$0{,}6$, thus indicating the possibility to use the results of subcritical experiments in practice, with significant subcriticality of cooling ponds being taken into account.

5. BDBA analysis

In accordance with the Rules it is necessary to study the initial BDBA events related to the occurrence of self – sustaining chain reaction for material storage and management systems, the complete dry – out of spent fuel storage, the drop of heavy process equipment onto the storage pool. The latter initial event is usually reduced to consideration of the two former events. The self-sustaining chain reaction is the result of initial events or a number of failures or mistakes of the personnel. The most probable scenarios of self-sustaining chain reaction consists in water density decrease as a result of long black-out, drying process, when water remains inside FA tubes, drop or drawing together of a significant number of FAs at the cooling pond bottom due to a drop of wrappers, heavy objects or to an external effect (aircraft fall, explosion, earthquake, etc.). In order to solve this problem the BURST code was developed. It can jointly solve the equations of transient processes within the framework of point model of kinetics and those of thermal hydraulic processes in one elementary cell of fuel pins or FAs. The dependence of reactivity on water – steam mixture density and temperature is calculated in advance and is used as input data for the BURST code. In the course of self – sustaining reaction progression fuel pin heat removal modes are changing, each of them influencing the k_{eff} value of the system.

The developed method of calculation of thermal physical parameters makes it possible to calculate spatial and time distribution of water – steam mixture during boiling and, based on that, the k_{eff} value. The method was implemented in the TEHRA code. As the calculation results show the amount of fission products and actinides formed and their part which goes to the environment are determined by several factors. The determining factors are the total number of fissions during the self – sustaining chain reaction, fuel burn-up, the number and state of fuel pins which lost their tightness. The conservative model is used to calculate the worst consequences. The model considers FAs with maximum burn-up, i.e. the maximum amount of fission products accumulated.

The analysis of self – sustaining chain reaction caused by the drop of a big number of FAs in various parts of near – the – reactor and intermediate NPP storages with RBMK fuel shows the following regularities: energy release caused by reactivity of several β_{eff} within 1s is equal to 10^{19}–10^{21} fissions, the fuel pin temperature increase results in the loss of tightness of these fuel pins and in the failure of the central part of the system within less than 1s after the beginning of self – sustaining chain reaction. The energy release determines the yield of radioactive noble gases and iodine. The yield of cesium and α-emitting products is determined by their generation in the reactor, i.e. the fuel burn-up fraction value. Radioactive discharges into the atmosphere (in the assumption that they will by-pass the ventilation system) are equal about 2000–3000 Curie of cesium. These values are lower than the regulatory requirements to the NPP location for the DBA with the most severe consequences.

6. Conclusions

The experience of operation of spent fuel storage under water and the results of studies show that the developed set of measures provides nuclear safety of existing storage facilities. However due to the necessary to make them more crowded with FAs, to introduce new storage facilities with new FA types and to introduce more stringent safety requirements, additional studies are required.

The practical use of the software for k_{eff} calculation, i.e. such codes as MMKFK-2, MCU-RFFI, etc., allows the calculation of a spent fuel storage with the error of k_{eff} not higher than 1 %. At the same time additional studies are required to determine the calculation errors of mixed U-Pu, U-gadolinium, U-erbium fuel.

The analysis of emergency conditions shows that the determining factor of nuclear safety is the correct account of water density variations in the entire volume of the storage and its separate areas.

The use of solid absorbers makes it possible to significantly increase the storage capacity and to decrease the risk of criticality when water density goes down.

The consideration of burn-up fraction allows a more crowded location of fuel in a long – term storage and safety justification in BDBA conditions. However additional studies are required to determine the effect of complex isotopic composition of fuel, non-uniform distribution of fissile nuclides along the FA height, FA irradiation history in the reactor on the k_{eff} values. The instrumentation control of burn-up fraction is required when FAs are located in the storage with consideration of their burn-up fraction.

The measurement of storage subcriticality makes it possible to determine the real k_{eff} value in the storage facility and is an additional measure to provide nuclear safety. Besides, these measurement results allow the prediction of k_{eff} in storage facilities in case of DBDAs. However the studies are required to determine the k_{eff} measurement error in case of profound subcriticality. A more detailed study of DBDA consequences is also required in order to develop the measures for personnel and population protection.

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