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Method of Analysis of Thermophysical Properties and Composition of Nuclear Fuel During Modernization of Active Zones of Nuclear Power Reactors

Volodymyr Vashchenko[1], Volodymyr Skalozubov[2], Iryna Korduba[3], Yuriy Komarov[2], Olena Zhukova[3*]

[1] Department of Ecology, National Aviation University, 1, Huzara Lyubomyra Ave., Kyiv, 03058, Ukraine

[2] Interdepartmental Center for Fundamental Scientific Research in the Field of Energy and Ecology of the National Academy of Sciences of Ukraine, "Odessa Polytechnic" and the Ministry of Ecology of Ukraine, 1, Shevchenko Ave, Odesa, 65044, Ukraine

[3] Department of Environmental Protection and Labor Protection Technology, Kyiv National University of Construction and Architecture, 31, Povitroflotskyi Ave., Kyiv, 03037, Ukraine

* Corresponding author's e-mail: elenazykova21@gmail.com

ABSTRACT

The developed alternative method of analyzing the safety of the active zone of reactor installations is justified for improving the thermophysical properties and composition of nuclear fuel, designs of heat-releasing assemblies, reactor operation modes at increased or reduced power, etc. The impact of modernization on the safety criteria and conditions of reactor installations (RF) was analyzed. Attention was focused on the fact that until now there have been no sufficiently substantiated and accepted criteria and conditions for "steam" explosions in the Russian Federation. It was shown that when analyzing the safety of HF modernization with deterministic codes, it is necessary to take into account the possibility of negative effects of "user code" (EUC) and "code difference" (ECD), which can significantly affect the results of re-simulation of accidents with deterministic codes taking into account modernizations.

Keywords: safety analysis of RF upgrades, criteria and conditions of "steam" explosions in RF, deterministic codes, "user code" effects, "code difference" effects.

INTRODUCTION

In most deterministic codes for accident modeling, the determining criterion for the nuclear safety of nuclear power plants is the maximum allowable temperature of the shells of heat-emitting elements, the parameters of which correspond to the conditions of intensification of the zirconium vapor reaction with the formation of explosive hydrogen and intense heat release. However, safety analysis also requires criteria and conditions for the maximum permissible temperature of the beginning of nuclear fuel degradation (severe accidents) and the occurrence of "steam" explosions with the destruction of protective safety barriers. In particular, "steam" explosions occurred at the 4th unit of the Chornobyl NPP and at the 2nd unit of the Fukushima Daiichi NPP. Until now, there have been no well-grounded and generally accepted criteria and conditions for "steam" explosions in the Russian Federation.

According to the safety rules and regulations in nuclear energy, when upgrading the active zones of reactor facilities (RF), it is necessary to analyze the impact of modernization on safety criteria and conditions. The traditional approach to safety analysis is based on probabilistic and deterministic modeling of possible initial emergency events (Vasylchenko V.N., 2002; Kim V.V., 1999; Podlazov L.N., 1994; Skalozubov V.Y., 1999). The use of such an approach to safety analysis in the case of numerous modernizations of RF is impractical for the following main reasons:

1. In most cases, a two-way rupture of the main circulation pipeline (maximum design accident – MPA) is analyzed as the primary emergency event dominant for safety. However, MPA is not a generalized accident in terms of probability of occurrence and accident consequence. In particular, from the experience of operation (Vasylchenko V.N., 2002), the more likely initial events for RF with VVER are inter-loop flows in the volume of steam generators, which can have more significant consequences for nuclear and radiation safety than MPA. In addition, it is necessary to take into account that the accidents with inter-loop leaks have more critical configurations of safety systems, increasing difficulties in identifying initial events and managing emergency processes, etc. After the major accident at the Fukushima-Daiichi NPP in 2011, the emergency events with a complete long-term power outage of the RF, which previously did not have a priority for VVER due to the low probability of their occurrence, also gained increased relevance.
2. The maximum allowable temperature of the fuel shells is used as a determining criterion for the nuclear safety of the RF in the majority of known deterministic accident modeling codes, the parameters of which correspond to the conditions of intensification of the vapor zirconium reaction with the formation of explosive hydrogen and intense heat release. However, the criteria and conditions for the maximum permissible temperature of the beginning of nuclear fuel degradation (severe accidents) and the occurrence of "steam" explosions with the destruction of protective safety barriers are also necessary for safety analysis. In particular, "steam" explosions occurred at the 4th unit of the Chornobyl NPP and at the 2nd unit of the Fukushima-Daiichi NPP. Until now, there have been no well-grounded and generally accepted criteria and conditions for "steam" explosions in RF.
3. The negative effects of "code user" (ECK) and "code differences" (ECC) can have a significant impact on the results of re-simulation of accidents with deterministic codes, taking into account the modernization of RF. EKK is related to the fact that when simulating the same accidents with the same codes, but by different users, the final results of the calculation simulation may differ significantly. EVK is related to the fact that when calculating the same accidents, but with different codes of the same level of verification, the final results of the calculations can also differ significantly.

EKK and EVK were installed in programs for verification of various deterministic accident simulation codes at RF/ A characteristic example of these effects can be the results of the experimental verification of simulation codes for the accident with leaks in the reactor circuit in the city of Elektrogorsk (experimental stands of ISB-VVER and PSB-VVER) (Skalozubov V.Y., 1999). Verification programs at the ISB-VVER and PSB-VVER stands were carried out in two stages.

At the first stage of verification (pre-test calculations), users were provided with the necessary output data without the results of the experiments.

At the second stage of verification (post-test calculations), users were provided with the results of experiments with codes.

As a result of the pre-test calculations, significant ECC and EWC were established, in addition to significant discrepancies of calculated and experimental thermohydrodynamic parameters for all codes and users. As a result of the post-test calculations, the discrepancies between the calculated and experimental data were significantly reduced, but the ECC and EWC remained.

Thus, when analyzing the safety of RF modernizations with deterministic codes, it is necessary to take into account the possibility of negative ECC and EWC. The research aimed at the above provisions, which determine the development of alternative methods of safety analysis during the modernization of RF, should be considered relevant.

The conducted research was aimed at developing an alternative method of safety analysis during modernization of the active zone of the reactor, which does not require the involvement of re-simulation of accidents with deterministic codes.

MATERIALS AND METHODS

The main provisions of the method of nuclear safety analysis in the modernization of the active zone of the reactor.

1. Deterministic criteria (indicators) of RF nuclear safety are accepted as follows:
   - maximum permissible temperature of the beginning of irreversible degradation of nuclear fuel T_{Fm};
   - the maximum permissible temperature of the beginning of the intensification of the vapor zirconium reaction with the generation of explosive hydrogen and heat emissions in the shell of the heating element T_{obm}.

Nuclear safety conditions:

$$T_F < T_{\text{Fm}} \tag{1}$$

$$T_{\text{ob}} < T_{\text{obm}} \tag{2}$$

The maximum allowable temperature of nuclear fuel is determined by its chemical composition. For “pure” UO_2 fuel, T_{Fm} = 3113°K. For zirconium shells of nuclear reactors with VVER T_{obm} = 1473°K.

2. The method of analysis of nuclear safety conditions is based on the determination of analytical dependences of changes in the maximum temperature of nuclear fuel T_{F} and fuel cell shell T_{ob} during an accident or transition mode of a modernized reactor core (m) from the basic design model (b). In the general case, modernization of thermophysical properties and design of fuel elements, chemical composition of fuel elements, structural and technical parameters of reactor heat-generating assemblies, non-design modes of operation of the reactor at increased or reduced power, etc. were analyzed.
3. A one-dimensional “two-zone” model of the fuel matrix of a fuel cell, consisting of the main (central) and surface (rim-zone) zone of the fuel matrix, was analyzed (Fig. 1).

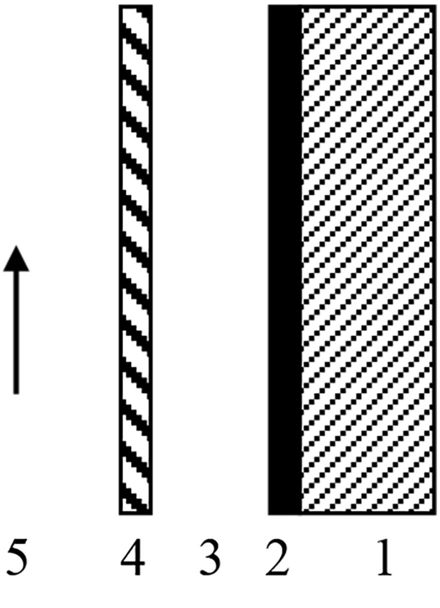

Figure 1. Two-zone model of the fuel matrix of the fuel oil: 1 – the main (central) zone of the fuel matrix; 2 – surface rim zone of the fuel matrix; 3 – fuel gas gap; 4 – fuel shell; 5 – coolant flow.

The heat balance equation for the fuel matrix and fuel cell shell, respectively, are as follows:

$$\begin{aligned} C_{\text{pF}} M_F \frac{dT_F}{dt} = N_F(t) - \\ -R_T^{-1} \Pi_T (T_F - T_{\text{ob}}) \end{aligned} \tag{3}$$

$$\begin{aligned} C_{\text{pob}} M_{\text{ob}} \frac{dT_{\text{ob}}}{dt} = R_T^{-1} \Pi_T (T_F - T_{\text{ob}}) - \\ -\alpha_T \Pi_T (T_{\text{ob}} - T_T) \end{aligned} \tag{4}$$

$$T_F(t=0) = T_{\text{F0}};\ T_{\text{ob}}(t=0) = T_{\text{ob0}} \tag{5}$$

where: T_F, T_{ob} – the maximum temperature of the nuclear fuel and the fuel shell, respectively;
t – time;
C_{pF}, C_{pob} – the average heat capacity per unit mass of the fuel and the fuel shell, respectively;
M_F, M_{ob} – the mass of the fuel matrix and fuel cell shell, respectively;
$N_F(t)$ – the power of the heat-releasing fuel matrix;
P_T – fuel cell surface area;
α_T – is the coefficient of heat transfer on the surface of the heating element;
T_T – coolant temperature;
R_T – the thermal resistance of the heating element.

$$R_T = \frac{\delta_{F1}}{\lambda_{F1}} + \frac{\delta_{F2}}{\lambda_{F2}} + \frac{\delta_g}{\lambda_g} + \frac{\delta_{ob}}{\lambda_{ob}} \tag{6}$$

where: δ_{F1}, δ_{F2}, δ_g, δ_{ob} – the thickness of the central zone of the fuel matrix, the rim-zone, the gas gap, and the fuel shell, respectively;
λ_{F1}, λ_{F2}, λ_g, λ_{ob} – coefficient of thermal conductivity of the central zone of the fuel matrix, the rim-zone, the gas gap, and the fuel shell, respectively.

After the transformation of the heat balance equations for the fuel matrix and the fuel shell 3 and 4, a system of equations in the format of criteria for the influence of structural and technical parameters on nuclear safety conditions 1 and 2 will be obtained.

$$\frac{dT_F}{dt} = K_N - K_{\mathrm{RF}}(T_F - T_{\mathrm{ob}}) \tag{7}$$

$$\begin{gathered}\frac{dT_{\mathrm{ob}}}{dt} = K_{\mathrm{Rob}}(T_F - T_{\mathrm{ob}}) - \\ -K_\alpha(T_{\mathrm{ob}} - T_T)\end{gathered} \tag{8}$$

where: the criterion for the influence of the power of nuclear fuel energy releases on the conditions of nuclear safety is:

$$K_N = \frac{N_F}{C_{\mathrm{pF}} M_F} \tag{9}$$

Criterion for the effect of thermal resistance of fuel oil on nuclear safety conditions:

$$K_{\mathrm{RF}} = \frac{R_T^{-1}\Pi_T}{C_{\mathrm{pF}} M_F}; K_{\mathrm{Rob}} = \frac{R_T^{-1}\Pi_T}{C_{\mathrm{pob}} M_{\mathrm{ob}}} \tag{10}$$

Criterion for the influence of external heat exchange conditions on the surface of the heating element:

$$K_\alpha = \frac{\alpha_T \Pi_T}{C_{\mathrm{pob}} M_{\mathrm{ob}}} \tag{11}$$

Parameters of modernization of the criteria of impact on nuclear safety conditions:

$$\begin{gathered}m_N = \frac{K_N(m)}{K_N(b)}; m_{\mathrm{RF}} = \frac{K_{\mathrm{RF}}(m)}{K_{\mathrm{RF}}(b)}; \\ m_{\mathrm{Rob}} = \frac{K_{\mathrm{Rob}}(m)}{K_{\mathrm{Rob}}(b)}\end{gathered} \tag{12}$$

$$m_\alpha = \frac{K_\alpha(m)}{K_\alpha(b)} \tag{13}$$

Conditions for ensuring or improving nuclear safety during the modernization of the reactor core:

$$\frac{dT_F(m)}{dt} - \frac{dT_F(b)}{dt} \leq 0 \tag{14}$$

$$\frac{dT_{\mathrm{ob}}(m)}{dt} - \frac{dT_{\mathrm{ob}}(b)}{dt} \leq 0 \tag{15}$$

Then, taking into account the modernization parameters 12 and 13, the conditions for ensuring or increasing nuclear safety during the modernization of the reactor core 14 and 15, respectively:

$$\begin{gathered}(m_N - 1)K_N(b) - \\ -(m_{\mathrm{RF}} - 1)\Delta T_{\mathrm{Fob}}(b) \leq 0\end{gathered} \tag{16}$$

$$\begin{gathered}(m_{\mathrm{Rob}} - 1)\Delta T_{\mathrm{Fob}}(b) - \\ -(m_\alpha - 1)\Delta T_{\mathrm{obT}}(b) \leq 0\end{gathered} \tag{17}$$

$$\begin{gathered}\Delta T_{\mathrm{Fob}} = T_F(b) - T_{\mathrm{ob}}(b); \\ \Delta T_{\mathrm{obT}} = T_{\mathrm{ob}}(b) - T_T(b)\end{gathered} \tag{18}$$

The practical application of the obtained conditions for ensuring or increasing nuclear safety can be demonstrated on the example of two urgent tasks for modern nuclear power industry: modernization of the active zone of nuclear power plants with VVER – diversification of heat-emitting assemblies of nuclear reactors and improvement of thermophysical properties of nuclear fuel.

MODERNIZATION OF VVER HEAT-RELEASING ASSEMBLIES

It should be noted that the practical application of the obtained conditions for ensuring or improving nuclear safety during the modernization of the nuclear power plant can be demonstrated on the basis of the experience of diversifying the TVS-A VVER project to alternative TVS-W of the Westinghouse company. The main differences between TVS-A and TVS-W are related to individual differences in assembly designs, which determine their differences in hydraulic resistance: the total coefficient of hydraulic resistance ξ TVS-W is greater than TVS-A. Therefore, other things being equal, the consumption of cooling TVEL coolant in TVS-W is less than in TVS-A; and, accordingly, the external heat exchange conditions worsen ($m_\alpha < 1$ under safety conditions 17). Compensation of the resulting relative deterioration of external heat exchange conditions can be carried out via modernization of the criteria for the effect of the thermal resistance of the heating element K_{Rob}, K_{RF} on the conditions of nuclear safety 16) and 17. In particular, a preventive decrease in the thermal resistance of the fuel element R leads to a relative decrease in the temperature of the fuel element shells (a negative effect for the safety condition 17). Therefore, it is necessary to justify the optimal value of R for the joint feasibility of nuclear safety conditions 16 and 17 in the case of deterioration of the external heat exchange conditions on the surface of the heating element due to the increased hydraulic resistance of the TVS-W.

MODERNIZATION OF PHYSICAL PROPERTIES AND COMPOSITION OF NUCLEAR FUEL

The issues of modernization of the thermophysical properties and composition of nuclear fuel are closely related to two urgent problems of nuclear energy: increasing the depth of nuclear fuel burnout and the duration of fuel campaigns (economic effect); reducing the damage of nuclear fuel (the effect of ensuring nuclear and environmental safety).

The consequence of increasing the thermal conductivity or decreasing the heat capacity of nuclear fuel by adding various chemical components is:

- a relative decrease in the temperature of nuclear fuel and damage to fuel matrices (the effect of ensuring safety); additional temperature "reserve" of nuclear fuel to ensure nuclear safety conditions (1) in matrix emergency modes (effect of ensuring safety);
- other things being equal, a relative increase in the temperature of the fuel cell shells, and accordingly, a decrease in the temperature "reserve" under the safety condition 2 in emergency modes (a decrease in the effect of ensuring safety); reduction of the depth of nuclear fuel burnout (reduction of the economic effect).

An increase in the concentration of plutonium or other highly active nuclides provides an increase in the depth of nuclear fuel burnout, but reduces the provision of nuclear safety conditions 1. Therefore, modernization of the thermophysical properties and composition of nuclear fuel has a complex and optimization nature. The parameters of modernization of thermophysical properties and composition of nuclear fuel m_N and m_{RF} should ensure the feasibility of nuclear environmental safety conditions 16 and 17. At the same time, modernization of the thermophysical properties and composition of nuclear fuel has a complex and optimization nature. The parameters of the modernization of thermophysical properties and composition of nuclear fuel mN and mRF should ensure the implementation of nuclear safety conditions.

CONCLUSIONS

The presented alternative method of safety analysis during the modernization of the reactor core excludes the influence of "user code" and "code difference" effects. The method of safety analysis during the modernization of the active zone of reactor installations is based on the obtained analytical dependences of the change in the maximum temperature of nuclear fuel and shells of heat-emitting elements in the modernized and basic model of the active zone of the reactor.

The main limitation of the validity of the implementation of deterministic codes in the analysis of the safety of modernization of reactor installations is related to the effects of "user code" and "code differences". These negative consequences make it difficult to objectively assess the impact of modernization of reactor facilities on safety criteria and conditions

The presented alternative method excludes the influence of "code user" and "code difference" effects when modernizing the thermophysical properties and composition of nuclear fuel, designs of heat generating assemblies, reactor operating modes at increased or reduced power, etc.

The method of safety analysis during upgrades of the active zone of reactor installations is based on the obtained analytical dependences of changes in the maximum temperature of nuclear fuel and shells of heat-emitting elements in the modernized and basic model of the active zone of the reactor. The method is justified for use in the modernization of thermophysical properties and composition of nuclear fuel, designs of heat-releasing assemblies, operating modes of the reactor at increased or reduced power, etc.

REFERENCES

1. Vasylchenko V.N., Emelyanenko E.Z., Skalozubov V.I. 2002. Modeling of accidents at nuclear power plants of nuclear power plants. Odessa, Rezon.
2. Security analysis report. Block 5. Zaporizhia NPP. Final report. 2002. Energodar: SE NAEK "Energoatom".
3. Western European practice of verification and validation of calculation codes. 1997. Report on the TACIS program UK/TS/09.
4. Kim V.V., Skalozubov V.I. 1999. On the issue of regulating the applicability of software tools in the analysis of accidents at domestic reactors. Nuclear and radiation safety, 1, 45–51.
5. Analysis of experimental data with the results of computational studies of dynamic modes of operation of VVER-1000 power units. 1991. Questions

of atomic science and technology. Ser. Physics and technology of nuclear reactors, 17.

6. Podlazov L.N., Trekhov V.E., Cherkashov Yu.M. 1994. Computational modeling of the accident at the fourth unit of the Chernobyl NPP. Atomic Energy, 77(2), 9–15.
7. Skalozubov V.Y., Emelyanenko E.Z., Kim V.V. 1999. Analysis of the similarity of thermohydrodynamic processes in the circulation circuit at full-scale installations and integral experimental stands. Nuclear and radiation safety, 2(3), 45–55.
8. IAEA-TECDOC-848-1995. Simulation of a loss of coolant accident without high pressure injection but with secondary side bleed and feed. Results of the fourth standard problem exercise on the simulation of loss of coolant accidents. Report of the IAEA.-Technical Co-operation Project Aspects of WWER-440 Model 213 NPP.
9. Skalozubov V.Y., Vasylchenko V.N., Kim V.V., Emelyanenko E.Z. 1999. Analysis of the similarity of thermohydrodynamic processes in the active zone of full-scale installations and integral experimental stands. Nuclear and radiation safety, 2(1), 69–76.