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The use of express analysis of nuclear safety analysis in the modernization of nuclear power plants.

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Annotation: *An original method of nuclear safety analysis is presented. The method is based on an adequate interpretation of changes in safety conditions with respect to the temperature of the shells of fuel elements and nuclear fuel during modernization of nuclear power plants.The temperature limits of nuclear fuel and shell made of structural material are analyzed. The proposed method, based on conservative criteria for nuclear safety analysis. The safety conditions for the temperature of the fuel element and its shell are obtained. The proposed method does not require modeling of all possible sequences using special codes.Therefore, the volume of computational research is significantly reduced. In addition, it provides a quick adaptation of the criteria method for the rapid assessment of changes in nuclear safety for various initial events and conditions, as well as when modifying and / or changing the design of nuclear fuel.*

Key words:*nuclear safety, safety criterion, nuclear fuel, nuclear power plant, heat transfer.*

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I. THE URGENCY OF THE PROBLEM

The primary objective of analyzing spent nuclear fuel reactor systems is to assess the feasibility of safety criteria in a transient accident for various initial events. The fuel barriers of a nuclear power plant are directly protected by the fuel matrix and zirconium sheath.In accordance with [1], we list the characteristic stages of damage to the core of reactors with UO_2 fuel depending on temperature conditions (tab. 1).

Table 1

Stagename	Temperaturerange,K
Melting Ag-In-Cd Alloy Control Rods	1073—1173
The formation of the first Fe-Zr and Ni-Zr eutectics	~1223
Fe-Zr — Al(Al_2O_3)-Zreutectics	1473— 1573
Melting stainless steel and Inconel	~1723
SmeltingofPure Zircaloy-4	2033
Al_2O_3-UO_2eutectic	~2173
Melting Al2O3 of combustible slag	2323
The formation of ceramic α-Zr (O) -UO2 monoeutectics	~2673
The formation of ceramic U-Zr-O melt	~2873
Melting ZrO_2	2963
Melting UO_2	3123

II. SAFETY CRITERIA AND THEIR CRITICAL REFLECTION

In the framework of information on safety analysis for nuclear power plants, when modeling an accident without sufficient justification, only one criterion is considered - the achievement of the temperature of the fast oxidation of the cladding of a fuel rod, namely T_c^0 =1473K. It is assumed that a severe accident starts at temperature T_c^0 and leads to nuclear fuel damage. The appropriateness of such an approach to assessing nuclear safety requires further reflection.

As is known, at nominal reactor operation, the maximum heat flux is approximately 10^9Wt/m^{3}and the cooling of the surface of the fuel rod by the coolant flow provides a heat transfer coefficient of $3 \cdot 10^4$Wt/(m$^2 \cdot$K)[2].

From the analysis of these data it follows that the maximum temperature of the fuel in the central part reaches 2573 K at operating temperatures of the coolant and the sheath temperature of about 573 K. The dominant thermal resistance is nuclear fuel and a gas gap filled with helium. Significant non-uniformity of the radial distribution of the temperature of nuclear fuel and the temperature of helium in the gap (temperature difference of 873 and 1473 K, respectively) under nominal operating conditions of the reactor is determined by the relatively low thermal conductivity of helium and fuel: λ^{UO_2} = 2 and λ^{He} = 0.2 Wt/(m$\cdot$K), respectively.The relatively low thermal conductivity of UO_2 means that the thermal resistance of the fuel is several orders of magnitude higher than the thermal resistance of the convective barrier to heat transfer during normal reactor operation.This means that the maximum temperature of the fuel $T_f^{\max}$ is more sensitive to changes in the temperature regime than the maximum temperature of the fuel cladding $T_c^{\max}$. In emergency conditions, the thermal resistance of the convective barrier can increase significantly. Therefore, in the general case, the ratio between $T_f^{\max}$ and $T_c^{\max}$ depends not only on heat transfer in the fuel rod, but also on thermohydraulic processes in the primary circuit.But, even a simple extrapolation of the temperature distribution in the fuel rod during the nominal operation of the reactor to emergency operation allows us to conclude that the temperature in the central part of the shell reaches 1473 ° C.Also, characteristics such as fuel burnup and change in its chemical influence the maximum allowable temperature of the fuel composition.Thus, with the burning of uranium oxide and the accumulation of plutonium, the melting temperature of nuclear fuel can be reduced by 20—40 K. Implementation of measures to increase the efficiency of thermal and neutron-physical properties of nuclear fuel by changing its chemical composition (for example, [3]) lead to a change in the relationship between $T_f^{\max}$ and $T_c^{\max}$ even stationary in the operating conditions of a nuclear reactor.Another limitation on the applicability of the results of the probabilistic safety assessment is the modeling of accidents using the deterministic method only for the most probable scenarios.The main disadvantage of the probabilistic approach is that such an approach to ranking accident events virtually excludes from the consideration variants of events that have a relatively low probability and, as a result, the final status of most accident sequences is determined a priori without sufficient research.At this point in time, the main methods for full-scale modeling of severe accidents remain the calculation methods, with which, under given initial and boundary conditions, the mathematical implementation of mathematical models of the processes is based on the calculation software (codes).

Codes traditionally used for severe accident analysis are divided into integrated and detailed. Detailed codes are characterized by a relatively narrow field of application (simulate individual processes, stages, stages) and relatively high realistic modeling. Integrated codes have a wider field of application (simulate more than one stage of the development of a severe accident) and relatively low realism of modeling individual processes.However, this separation is rather arbitrary, since modern severe accident modeling codes usually contain features of both detailed and integrated codes. Thus, the integrated ASTTEC severe accident analysis code actually consists of detailed codes that simulate individual processes and steps, and SCDAP/RELAP5-3D, a detailed code for the first stage of a severe accident, also uses the «built-in» code to simulate the processes of accumulation and decay of radionuclides. To simulate all the main stages in the development of severe accidents, the use of ASTEC computer code is promising [16].

But it must be recognized that the full simulation of the accident requires a lot of effort, which for the most part will be ineffective. Therefore, the development of an express method of nuclear safety analysis is an urgent problem.

III. THE MAIN PROVISIONS OF THE METHOD OF ANALYSIS OF NUCLEAR SAFETY

The proposed nuclear safety analysis method is based on the following assumptions:

• the fuel rod is modeled as a cylindrical system with lumped parameters;

• fuel temperature T_f corresponds to the maximum value of the spatial temperature distribution in the fuel rod;

• the external conditions of heat exchange with the fuel rod are usually determined by the specific scenarios of the transient accident.

Under the assumptions, the heat balance equation can be represented as [4]:

$$\frac{d\left(\rho_f h_f\right)}{dt} = Q_f\left(t\right) - \frac{\lambda_f}{2r^2}\left(T_f - T_c\right), \qquad (1)$$

$$\alpha\left(t\right)\left(T_c - T_{cool}\right) = \frac{\lambda_f}{r}\left(T_f - T_c\right), \qquad (2)$$

$$T_f\left(t=0\right) = T_f^0, \qquad (3)$$

where ρ_f, h_f —density and specific enthalpy of nuclear fuel, respectively; T_f, T_c, T_{cool} — temperature of fuel, shell and heat carrier, respectively; $\alpha\left(t\right)$ — heat transfer coefficient on the outer surface; t — time, r — radius of the fuel segment; Q_f — specific heat flux. The value Q_f is determined by non-stationary neutron-physical processes in nuclear fuel [6 — 8]:

$$Q_f = \Phi \sum_j N_j \overline{\sigma_f^j} q_j, \qquad (4)$$

where Φ — specific neutron flux density; N_j — nuclear concentration for a nuclide j; $\overline{\sigma_f^j}$ — the neutron fission cross section averaged over the energy spectrum for the nuclide j; q_j — fission energy for nuclide j.

We introduce the temperature and time scale, respectively: $T_M = T_c^{melt}; t_M = t_A$ (t_A —duration of the simulated accident process).

Then equations (1) — (3) after the transformations are written as:

$$\frac{dT_f}{dr} + K_1 T_f = K_2, \qquad (5)$$

$$T_f = \left(Nu+1\right)T_c + NuT_{cool}, \qquad (6)$$

$$T_f\left(t=0\right) = T_f^0, \qquad (7)$$

where $\tilde{T}_f = T_f / T_f^{melt}; \tilde{T}_c = T_c / T_f^{melt}; \tilde{T}_{cool} = T_{cool} / T_f^{melt}; \hat{t} = t / t_A, Nu = \alpha\left(t\right) r / \lambda_f$ — Nusselt criterion;

$$K_1 = \frac{\lambda_f t_A \left(1 - \frac{1}{Nu+1}\right)}{2r^2 \left(\rho_f C_p^f + h_f \frac{d\rho_f}{dT_f}\right)};$$

$$K_2 = \frac{t_A}{T_f^{melt}} \frac{Q_f + \frac{\lambda_f Nu}{2r^2(Nu+1)} T_c}{\rho_f C_p^f + h_f \frac{d\rho_f}{dT_f}}.$$

Then the analytical solution (5) — (7) and the corresponding fuel cell safety criteria will be obtained in the form:

$$\tilde{T}_f = \left\{ T_f^0 + \int_0^t K_2(r) \exp\left[\int_0^r K_1\left(\xi\right) d\xi\right] dr \right\} \exp\left[-\int_0^t K_1\left(r\right) dr\right] < 1, \qquad (8)$$

$$\tilde{T}_c = \tilde{T}_f \frac{1}{Nu+1} - \frac{Nu}{Nu+1} \tilde{T}_{cool} < \frac{T_c^0}{T_f^{melt}}. \quad (9)$$

In a generalized form, the nuclear safety criteria of the proposed method are:

$$K_f\left\{Nu, K_1\left(Q_f\right), \tilde{T}_{cool}\right\} < 1, \quad (10)$$

$$K_c\left\{Nu, K_2\left(Q_f\right), \tilde{T}_{cool}\right\} < \frac{T_c^0}{T_f^{melt}}. \quad (11)$$

Limit values $K_f = 1, K_c = T_c^0 / T_f^{melt}$ mean: conditions that are critical to nuclear safety can be determined using the critical values of heat flux and heat transfer. Thus, the proposed criterion for the method of nuclear safety analysis does not require a detailed simulation of all possible accident sequences, but allows one to obtain conservative estimates of the critical values of the heat flux in nuclear fuel and heat transfer for the shell surface for specific scenarios.

IV. APPLIED ASPECTS OF THE APPLICATION OF THE METHOD.

We carry out a partial analysis of the possible application of the method in a comparative version with the results from materials [8 — 11]. One of the factors that determine the critical safety parameters is the maximum temperature of the cladding of the fuel rods, which is formed under the influence of all kinds of engineering uncertainties. The reasons for these uncertainties are manifold — manufacturing and assembly errors of reactor units, design formulas, experimentally obtained dependencies and constants, accuracy of maintaining operational parameters in operating conditions, data processing, methodological and metrological errors, etc.

To carry out pre-design calculations, a procedure has been developed for statistical estimation of the effect of random deviations of core parameters on the temperature of fuel claddings [12]. It introduces the overheating factor F — a random variable characterizing the maximum deviation of a certain parameter P, which determines the temperature difference from its nominal value. As a rule, random deviations of the parameters are significant in magnitude and have arbitrary distribution laws.In general terms, the maximum value of the nominal temperature of the inner surface of the cladding of a fuel rod is correctly calculated as a superposition of the average temperature of the coolant in the surrounding channels $\left(t_{in} + \overline{\Delta t_{LQ}}\right)$; the wall-liquid average temperature perimeter of the shell $\left(\overline{\Delta t_\alpha}\right)$; half local temperature non-uniformity around the perimeter of the shell $\left(\left(t_w^{\max} - t_w^{\min}\right)/2\right)$; overheating of the sheath under the spacer wire (if any) $\left(\Delta t_p\right)$ and temperature difference on the fuel cell shell $\left(\Delta t_{sh}\right)$:

$$t_w = \left(t_{in} + \overline{\Delta t_{LQ}}\right) + \Delta t_\alpha + 0{,}5\left(t_w^{\max} - t_w^{\min}\right) + \Delta t_p + \Delta t_{sh}. \quad (12)$$

In this case, for correlated values, the temperature deviation caused by the influence of overheating factors can be determined from the relation

$$\delta t_i = \left(\delta t_{i-1}^2 + A_i \Delta t_i^2 + 2\sum_{j=1}^{i-1} A_{i,j} \Delta t_i \Delta t_j\right)^{0,5} \quad (13)$$

Here δt_i —is the deviation of the temperature preceding the calculatedthe chain you are looking for; Δt_i — nominal value of the i-th temperature difference; Δt_j — the nominal value of the j-th temperature difference, the deviation of which under the influence of overheating factors is associated with a linear dependence with the deviation Δt_i; $A_i = \sum_m (K_m \cdot a_m \cdot F_m)^2$ — the sum of the squares of the products of the coefficients of relative dispersion, the coefficients a_m (a_m —an indicator of the degree to which the mparameter of the i-th temperature difference is raised) and the overheating factors determining the temperature difference; $A_{i,j} = \sum_{m_1 < m}\left(K_{m_1} \cdot a_{m_1} \cdot F_{m_1}\right)^2$ — the sum of the squares of the products of the coefficients of relative dispersion, the coefficients a_{m_1} and overheating factors affecting simultaneously temperature differences Δt_i and $\Delta t_j \left(j < i \right)$ (tab. 2)

Table 2

Parameters	Contribution of errors in heating
Physical characteristics of the core	0,11
The mass of the fuel core	0,16
The outer diameter of the shell	0,0011
Differential pressure along the length of the fuel rod	0,103
Feedthrough	0,52
FuelDensity	0,04
Fuelspecificheat	0,25
The specific heat of the coolant is	0,08
Kinematic viscosity of the coolant	0,015

Based on the method proposed for fast reactors, the influence of overheating factors on the deviation of the maximum temperature of the cladding of a fuel.

V. CONCLUSIONS.

1.It has been shown that the use of safety criteria in relation to the temperature range of a shell with selective deterministic accident modeling is usually not enough to adequately assess the safety of nuclear reactors.
2. An express method for the analysis of nuclear safety is proposed, based on an adequate interdependence of the safety criteria of the fuel matrix and the shell, as well as the conservative criterion of heat flux in nuclear fuel and heat transfer on the shell surface.
3. The proposed express method of nuclear safety analysis does not require detailed modeling of the entire possible sequence of the accident, which significantly reduces the number of ineffective studies and expands the possibilities of using this method for express assessment of the safety of a nuclear reactor over the entire spectrum of operation.
4. Further research will focus on the specific application of the proposed method for the analysis of nuclear safety of nuclear power plants.

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