Express Method of Nuclear Safety Analysis for VVER

V.I. Skalozubov[1], **I.E. Kozlov**[2], **S.I. Kosenko**[2], **V.N. Kolykhanov**[1]

[1] *Institute for Safety Problems of Nuclear Power Plants of National Academy of Science of Ukraine, 12, Lysogorska Str., Bl. 106, Kyev, 03028, Ukraine*

[2] *Odessa National Polytechnic University, 1, Shevchenko Avenue, Odesa, 65044, Ukraine*

Abstract. The original criterion method of nuclear safety analysis for WWER with Western nuclear fuel is presented. The method is based on adequate interdependence of safety criteria for fuel matrix and fuel cladding, and on conservative phenomenological criteria for nuclear fuel heat release power and heat transfer conditions in specific accident scenarios. The tolerability criteria for the temperature of nuclear fuel and fuel cladding from zirconium is analysed. The method based on the conservative criteria for analysis of nuclear safety is proposed. The heat balance equation and the boundary conditions of the external heat exchange are derived. The criteria for the safety for the temperature of fuel rod and cladding was obtained. The proposed method don't require modeling of all possible accident sequences using detailed codes. Therefore, scope of computational studies are essentially reduced. In addition, it enables fast adaptation of criterion method for express-evaluation of the nuclear safety variations for different initial events and conditions, and at modification and/or change of nuclear fuel design.

Keywords: nuclear safety, safety criterion, nuclear fuel, water-moderated water-cooled reactor (WWER), heat exchange

1 Introduction

The main purpose of the analysis of nuclear reactor safety and storage pools for spent nuclear fuel is to assess the feasibility of the safety criteria of the fuel elements (fuel rods) in the accident transient for various initial events (IE). Safety barriers of fuel elements of VVER are directly a fuel matrix and zirconium cladding.

According to the recent studies of the main stages of core damage for reactors with fuel UO_2 versus temperature are shown in Figure 1 [1].

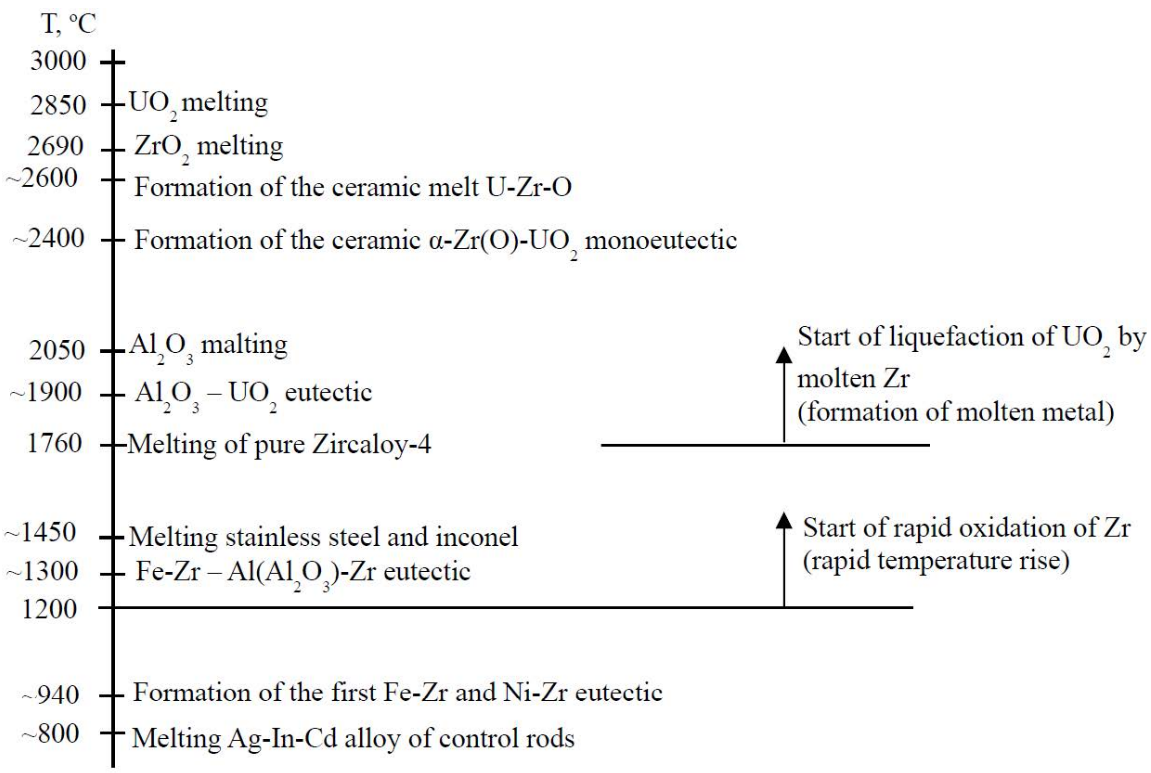

Figure 1. Typical phases of damage the reactor core with fuel UO_2.

2 The Safety Criteria and Deterministic Simulation

As part of the Safety Analysis Reports (SAR) for VVER at modeling accidents without sufficient justification only one criterion is considered – achievement of temperature rapid oxidation of fuel cladding $T_c^0 = 1200°$C. It is assumed that severe accident beginning at temperature T_c^0 and leading to damage of nuclear fuel (see. [1]). The admissibility of such an approach to the assessment of nuclear safety requires further analysis.

Figure 2 shows typical results of numerical simulation of the radial temperature distribution in VVER fuel rods with UO_2-fuel. At nominal operation of reactor the maximum heat flux is about 1×10^9 W/m^3 (or 1×10^5 W/kg), and cooling of the fuel rod surface by coolant flow provides a heat transfer coefficient of about 3×10^4 W/(m^{2}K) [2].

From these results, it follows that the maximum temperature of the fuel in the central part reaches 2300°C at the operating temperatures of the coolant and the cladding temperature of about 300°C. The dominant thermal resistance is the nuclear fuel and the gas gap is filled with helium (see Figure 2).

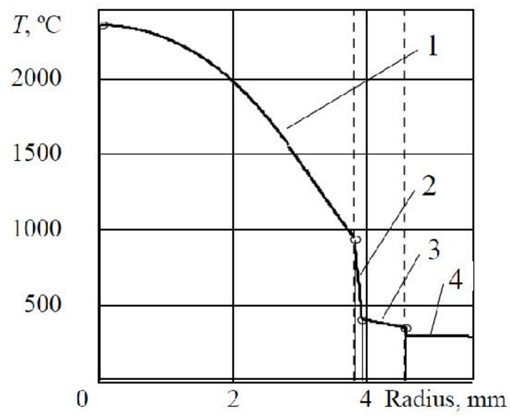

Figure 2. The radial temperature distribution in the fuel elements of the VVER.

Regarding the above results need to note the following:

1. A significant non-uniformity of the radial distribution of nuclear fuel temperature and the helium temperature in gap (temperature drop is 600 and 1200 °C, respectively) at the nominal operating conditions of the reactor is determined by the relatively low thermal conductivity of the helium and fuel – its values of the order $\lambda^{UO_2} = 2$ and $\lambda^{He} = 0.2$ W/(m K), respectively.

The relatively low thermal conductivity of UO_2 determines that the thermal resistance of the fuel is several orders of magnitude greater than thermal resistance of convective barrier to transfer heat during normal operation of reactor. Therefore, the maximum temperature of the fuel $T_f^{\max}$ is substantially greater sensitive than maximum temperature of the cladding $T_c^{\max}$ to changes in temperature regime.

2. In accident conditions associated with a significant deterioration in the value of fuel rods cooling thermal resistance of convective barrier could to increases significantly. Therefore, in general, the relationship between $T_f^{\max}$ and $T_c^{\max}$ depends not only on heat transfer in the fuel rod, but the heat from the thermo-hydrodynamic processes in the primary circuit.

However, even a simple extrapolation of the temperature distribution in the fuel rod during normal operation of the reactor (see. Figure 2) to the accident mode, which allows to conclude that in the central part of the fuel matrix melting will occur before the cladding temperature reaches 1200°C.

In addition, the maximum allowable temperature of fuel is affected by characteristics such as fuel burn-up and a change in its chemical composition. So, during a burn-up of uranium oxide and plutonium accumulation the melting point of the nuclear fuel could be reduced by 20–40°C. Implementation of measures to improve the efficiency of heat and neutron-physical properties of nuclear fuel by changing its chemical composition (see, eg, [3]) could also lead to a change in the relationship between the $T_f^{\max}$ and $T_c^{\max}$ even at stationary operating conditions of a nuclear reactor.

Another limitation of the applicability of the results of probabilistic safety assessment (PSA) evaluation is the accident simulation using deterministic methods only for the most probable scenarios. Because of the large amount of time needed for calculations of accident scenarios using computer codes and many possible accident scenarios outside of research often remain the rare accident, including beyond design basis accidents.

The main disadvantages and negative effects of conventional PSA probabilistic approach and the limited amount of deterministic modeling, in our opinion, are as follows:

1. A probabilistic approach ranging of accident events actually excludes from consideration the events which have relatively low probability. As a result of it the modeling, analysis and development of appropriate mitigation actions for these events are not carried out. Lessons of Fukushima accident in 2011 clearly demonstrated the unacceptability of such an approach [4].

2. The final status of most accident sequences are determined a priori, without sufficient deterministic studies. A typical example could be IE - the tornadoes at NPP site [4]. The SAR a priori assumed that tornado of any intensity eventually leading to severe accident with damage to the fuel rods in reactor. However, the probability of such events is relatively small and has a very small contribution to the integrated probabilistic safety indicators.

It must be acknowledged that the deterministic simulation of accidents in full scope requires a lot of effort, which for the most part will be ineffective. Therefore, the development of the deterministic express method of analysis of nuclear safety is an actual problem.

3 The Concept of the Method of Nuclear Safety Analysis

Proposed criterion conservative method of nuclear safety analysis is based on the following general terms assumptions:

- The fuel rod modelled as a cylindrical system with lumped parameters.
- Conservatively assumed that the temperature of fuel T_f corresponds to the maximum value of the spatial temperature distribution in the fuel rod, as the thermal conductivity of the nuclear fuel is $\lambda_f = f(T_f^{\max})$.
- The external conditions of heat transfer to the fuel rod is generally determined by the specific scenarios of the accident transient.

Within the framework of the assumptions of the heat balance equation and the boundary conditions of the external heat transfer can be expressed as [1]:

$$\frac{d(\rho_f i_f)}{dt} = Q_f(t) - \frac{\lambda_f}{2r^2}(T_f - T_c), \tag{1}$$

$$\alpha(t)(T_c - T_{\text{cool}}) = \frac{\lambda_f}{r}(T_f - T_c), \tag{2}$$

$$T_f(t=0) = T_f^0, \tag{3}$$

where ρ_f, i_f are the density and the specific enthalpy of nuclear fuel, respectively; T_f, T_c, T_{cool} – the temperature of the fuel, cladding and coolant, respectively; $\alpha(t)$ is the heat transfer coefficient on the surface of the cladding; t – the time; r – the radius of the fuel rod; Q_f – the specific heat flux.

The value of Q_f is determined by the non-stationary neutron-physical processes in nuclear fuel [5-7]:

$$Q_f = \Phi \sum_j N_j\, \overline{\sigma}_f^j\, q_j, \tag{4}$$

where Φ is the specific neutron flux density; N_j – the nuclear concentration for nuclide j; $\overline{\sigma}_f^j$ – averaged over the energy spectrum of the neutron fission cross section for nuclide j; q_j – the average energy of fission for nuclide j.

Introduce the temperature and time scales, respectively: $T_M = T_c^{\text{melt}}$; $t_M = t_A$ (t_A – the duration of the simulated accident process). Then, equations (1) — (3) after the transformations to dimensionless form are

$$\frac{dT_f}{d\tau} + \mathrm{K}_1 T_f = K_2, \tag{5}$$

$$T_f = (Nu + 1)T_c + Nu\, T_{\text{cool}}, \tag{6}$$

$$T_f(t=0) = T_f^0, \tag{7}$$

where: $\widehat{T}_f = T_f/T_f^{\text{melt}}$; $\widehat{T}_c = T_c/T_f^{\text{melt}}$; $\widehat{T}_{\text{cool}} = T_{\text{cool}}/T_f^{\text{melt}}$; $\widehat{t} = t/t_A$; and $Nu = \alpha(t)\, r/\lambda_f$ is the Nusselt number;

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

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

Then the analytical solution (5)–(7) and the relevant safety criteria of the fuel element have the form

$$\widehat{T}_f = \left\{ T_f^0 + \int_0^t K_2(\tau) \exp\left[\int_0^\tau K_1(\xi) d\xi\right] d\tau \right\} \times \exp\left[-\int_0^t K_1(\tau) d\tau\right] < 1, \tag{8}$$

$$\widehat{T}_c = \widehat{T}_f \frac{1}{Nu+1} - \frac{Nu}{Nu+1}\widehat{T}_{\text{cool}} < \frac{T_c^0}{T_f^{\text{melt}}}. \tag{9}$$

In general case at the certain phenomenological dependencies to heat flux $Q_f(t)$ and the conditions of heat transfer the nuclear safety criteria are

$$K_f\left\{Nu,\ K_1(Q_f),\ \widehat{T}_{\text{cool}}\right\} < 1, \tag{10}$$

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

For the limit values $K_f = 1$ and $K_c = T_c^0/T_f^{\text{melt}}$ it may be determined the conditions that are critical for the nuclear safety us a critical value of the heat flux and heat transfer.

Thus, the proposed criterial method of nuclear safety analysis not require a detailed simulation of all the possible accident sequences using codes, but allows to obtain an conservative estimates of the critical values of heat flux in nuclear fuel and heat transfer on surface of cladding for specific accident scenarios.

4 Conclusions

1. It is shown that the use of SAR criterion concerning the cladding temperature range with selective deterministic simulation of accidents is generally insufficient to adequately assess the safety of nuclear reactors.

2. The express method of nuclear safety analysis is proposed, which based on adequate interdependence of safety criteria of the fuel matrix and cladding, as well as on conservative phenomenological criteria of heat flux in the nuclear fuel and the heat transfer on surface of cladding.

3. The proposed express method of nuclear safety analysis not require a detailed simulation of all the possible accident sequences using codes, that significantly reduces the amount of ineffective studies and extends the possibility of

apply this method to the express estimation of the nuclear reactor safety to the whole spectrum of initial events, as well as the possible changing the design of a nuclear fuel.

4. Further research will be directed to a specific application of the proposed method for the nuclear safety analysis of the VVER units, including for spent fuel pool.

References

[1] Skalozubov V.I., Kluchnikov A.A., Kolyhanov V.N., Basics of beyond design basis accidents with loss of coolant in nuclear power plants with VVER, ISP NPP NAS of Ukraine, Chernobyl (2010) (in Russian).

[2] Solodov A.P., Fuel Element Heat Conductivity, Energoatomizdat, Moscow (1986) (in Russian).

[3] Shimkevich A., Proshkin A., Sedov A., Through innovation. Prospective dense fuel for power reactors, *Rosenergoatom* No 10 (2011) 36-41. (in Russian).

[4] Skalozubov V.I., Analysis of the causes and consequences of accident on NPP Fukushima as a factor in the prevention of severe accidents in tank reactors, ISP NPP NAS of Ukraine, Chernobyl (2012) (in Russian).

[5] Weinberg A., Theory of Nuclear Reactor Physics, Publishing House of Foreign Literature, Moscow (1961) (in Russian).

[6] Bartholomew G.G., Basic theory and methods for calculation of nuclear power reactors, Energoatomizdat, Moscow (1989) (in Russian).

[7] Feinberg S.M., Theory of Nuclear Reactors, Atomizdat, Moscow (1978) (in Russian).