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ANALYSIS OF NUCLEAR SAFETY IN DIVERSIFICATION OF WESTINGHOUSE FUEL ASSEMBLIES AT WWER-1000

The research presents an analysis of the known results in modeling the maximum design accident (MDA) using the code RELAP5/V3.2 whith Westinghouse fuel assemblies' (WFA) diversification in WWER-1000 reactors. According to the known results of MDA calculated model simulation with RELAP5/V3.2 code at the maximum allowable water temperature in the heat WWER emergency cooling system exchanger (90 °C), the fuel elements' claddings temperature reaches 1320 °C and exceeds the admissible nuclear safety limit (1200 °C). Thus, according to known results, these MDA whith WFA engaged pass from the "design" accident status to the "severe" acciden status and means a decrease in safety in relation to the FA-A fuel assemblies. The alternative MDA analysis for WFA-equipped plants showed that, unlike the known calculations, the nuclear safety limit on the maximum permissible fuel cladding temperature is not violated and never reduces the overall safety level in WWER diversification with WFA fuel assemblies.

Keywords: security, diversification of fuel assemblies.

1. Diversification issues relevance

The main supplier of nuclear fuel to European and Ukrainian WWER-equipped nuclear power plants is Russia. However, other nuclear markets' experience shows that the monopolization of nuclear fuel supply and storage does adversely affect both security and competition in terms of improving technology and economical efficiency [1]. In addition, there is a positive experience reactor core mixed loading with nuclear fuel from different suppliers. The transnational Westinghouse Electric Company accumulated many years of experience in nuclear fuel supplies for various reactors types and is a promising alternative fuel supplier for WWER (including those located in Ukraine).

Of undoubted interest is the experience of Westinghouse fuel assemblies diversifying at Temelin NPP (Czech Republic) [2]. In the process of Westinghouse fuel assembly diversification at Temelin NPP, the following main problems were identified: problems with rods control cluster assemblies (RCCA) during operation identified are the problems with incomplete control rods introduction (IRI); bending of fuel assemblies / fuel rods; FA excessive elongation; leakage of fuel elements.

In accordance with the Ukraine Nuclear Fuel Qualification Program (UNFQP), the Westinghouse Company and the United States Department of Energy, with the participation of the Northwest Pacific National Laboratory (PNNL), supply Ukraine with alternative nuclear fuel for reloading the WWER-1000 reactor fuel assemblies (WFA). According to information from the Westinghouse Swedish branch [3] in 2005, the first six test assemblies (Lead Test Assemblies - LTA) for WWER-1000 were loaded to Unit 3 of the South Ukrainian NPP and demonstrated full compatibility with the project nuclear fuel and reactor designs. Further operation demonstrated the adequacy of operational limits and reliability requirements for several fuel campaigns of the reactor. After the planned four fuel campaigns completion, the average burnup depth was more than 43 MW·day/kg of uranium.

In [4 and 5] a comparative analysis of the structural differences and compatibility between Westinghouse LTA-2 and WFA fuel assemblies and the design WWER (TBC-A) fuel assemblies has been presented. The main differences between the designed WWER fuel assemblies and alternative WFAs, consisting in fuel assemblies and fuel elements design, as well as in spatial distributions of nuclear fuel enrichment, can affect nuclear safety.

A preliminary expert analysis of safety criteria feasibility carried out by the Ukrainian nuclear security regulatory authority for fuel assemblies and WFAs mixed loads at WWER-type reactors using the RELAP5/V3.2 code for the maximum design accident (MDA) showed that the hydrodynamic resistance coefficients (HdRC) difference determine the WFA fuel cladding temperature T_{clad} several hundred degrees higher than the corresponding values for fuel assemblies, and at the maximum permissible design temperature of coolant at emergency core cooling system (ECCS) 90 ºC the safety criteria are not fulfilled at all: T_{clad} > 1200 ºC (Table 1) [6]. At the same time, it is necessary to take into account that in the analysis of beyond design accidents, the influence of these discrepancies in T_{clad} values may be even more significant for assessing the nuclear safety criteria and formal transition of entire groups of beyond design accidents without nuclear fuel damage into the severe accidents status.

Table 1. **Results of MDA simulation with WFAs and fuel assemblies cladding temperature difference ($\Delta T_{clad} = T_{clad}$(WFA) – T_{clad}(TBC-A)) at various variance of active zone loading [6], ºC**

Load variance	ECCS water temperature	1-st peak of cladding temperature	2-nd peak of cladding temperature	ΔT_{clad} at 2-nd peak T_{clad}
42 WFA + 121 TBC-A	70	867	1099	250
163 TBC-A	70	874	849	–
163 WFA	70	850	1074	225
163 WFA	90	850	1320	–

Thus, according to the accident simulation results, obtained in [6], WFA loading at WWER reduces the overall nuclear safety level. To exclude that it is suggested in [6] that the conservatism of nuclear safety analysis should be reduced by applying the ECCS coolant temperature no more than 70 °C. However, this approach requires revision of the design safety criteria and coordination with both WWER-1000 and the reactor facility design and construction developing organizations.

Also, at simulation as in [6], the generalizing design accidents, some unjustified and/or "excessively" conservative provisions were used including the absence of coolant flows mixing in different fuel assemblies. WFA/fuel cladding constructions, as well as developed turbulence hydrodynamic modes, determine the possibility of coolant flows intensive mixing on the greater part of the fuel assembly height. Thus, an urgent issue becomes an alternative analysis of nuclear safety in Westinghouse fuel assemblies' diversification.

2. Alternative analysis of nuclear safety in fuel assemblies' diversification

Let us consider the influence of differences between WFA and design fuel assemblies on the change in the WWER reactor safety level on the basis of the nuclear reactors' thermohydrodynamics fundamental provisions [7 - 11].

The main reason for the hydraulic characteristics change in WWER core, in this case, are the consequences of modernization to strengthen the WFA fuel assembly structure and to increase its reliability, including for the prevention of fuel rods deformation. The resulting increase in HdRC (due to additional WFA designs) leads to a relative decrease in coolant' flow rate and average velocity, and, accordingly, involves a decrease in heat exchange intensity on the fuel rods outer surface.

The boundary condition for heat transfer on the fuel element surface

$$q_{\text{T}} = \alpha\left(T_{\text{clad}} - T_{\text{T}}\right),$$

where q_{T} – density of heat flow from the fuel element, determined by the fuel matrix thermal mode and properties and the fuel element thermal resistance R_{T}; α – the heat transfer coefficient at the fuel element surface; T_{clad}, T_{T} – the temperature of the fuel element cladding and the coolant, respectively.

Then the differences between the temperatures of the fuel element cladding WFA and fuel assembly at the current time (under the assumption of q_{T} identity)

$$\Delta T_{\text{clad}} = T_{\text{clad}}(\text{WFA}) - T_{\text{clad}}(\text{TBCA}) =$$

$$= q_{\text{T}}\left[\frac{1}{\alpha(\text{WFA})} - \frac{1}{\alpha(\text{TBC-A})}\right]. \qquad (1)$$

The indicator of change in heat release intensity $\kappa_{\text{т}} = \alpha(\text{WFA}) / \alpha(\text{TBC-A})$ in this case is determined by the indicator of changes in coolant average velocity v

$$\kappa_{\text{т}} = \frac{v(\text{WFA})}{v(\text{TBC-A})}.$$

In accordance with the known phenomenological dependencies for the heat exchange modes/conditions and the coolant phase in nuclear reactors (including those used in different correlations of the RELAP 5 code)

$$\alpha \sim v^n,$$

where n ≤ 1 (see, for example [8 - 11]).

Correspondingly, the dependence between the heat transfer rate change parameters and the coolant velocity

$$\kappa_{\text{т}} = (\nu)^n, \qquad (2)$$

and the current discrepancies in the fuel cladding temperature

$$\Delta T_{\text{clad}} = q_{\text{T}}\left[\frac{1}{(\nu)^n \alpha(\text{WFA})} - \frac{1}{\alpha(\text{TBC-A})}\right] =$$

$$\frac{q_{\text{T}}}{\alpha(\text{TBC-A})} \cdot \frac{1-(\nu)^n}{(\nu)^n}. \quad (3)$$

Thus, the differences in the WFA and fuel elements cladding temperature current values under assumption that the fuel elements' structures and the thermal resistances R_{T} are identical, are determined by the change in the coolant average velocity v and the intensity ratio between the fuel elements internal and external heat exchange

$$\kappa_q = \frac{q_{\text{T}}}{\alpha(\text{TBC-A})}.$$

In the absence of fuel rods structural identity and thermal resistance, current difference in the temperature of the WFA and fuel elements claddings

$$\Delta T_{\text{clad}} = \frac{1-(\nu)^n}{(\nu)^n} \cdot \frac{q_{\text{T}}(\text{TBC-A})}{\alpha(\text{TBC-A})}. \quad (4)$$

The relationship between the heat flow density q_{T} and the thermal resistance R_{T} of the fuel rod

$$q_{\text{T}} = \frac{T_{\text{FM}} - T_{\text{clad}}}{R_{\text{T}}}, \quad (5)$$

where in the admissible "flat" approximation

$$R_{\text{T}} = \frac{\delta_{\text{clad}}}{\lambda_{\text{clad}}} + \frac{\delta_{\text{g}}}{\lambda_{\text{g}}} + \frac{\delta_{FM}}{\lambda_{\text{FM}}}, \quad (6)$$

δ_{clad}, δ_{g}, δ_{FM} – thickness of the cladding, gas gap and fuel matrix of the fuel rod, respectively; λ_{clad}, λ_{g}, λ_{FM} – coefficient of the fuel rod's thermal conductivity of the cladding, gas gap and fuel matrix respectively; T_{FM} – the nuclear fuel temperature in the central part of the fuel element's fuel matrix.

Analysis of dependences (5) and (6) shows that the thickness differences between fuel matrix and the gas gap of the fuel elements WFA and fuel assemblies TBC-A (δ_{FM} (WFA) / δ_{FM} (TBC-A) = 1.1; δ_{g} (WFA) / δ_{g} (TBC-A) = 1.1 [4 and 5]) determine the discrepancies q_{T} (WFA) and q_{T} (TBC-A):

$$q_{\text{T}}(\text{WFA}) = 0{,}9 q_{\text{T}}(\text{TBC-A}).$$

The effect of differences in material and thickness of the fuel cladding for q_{T} is even less significant.

The relatively smaller q_{T} (WFA), all other conditions being equal, on the one hand, decreases the fuel cladding temperature, and on the other hand, increases the nuclear fuel temperature in the central part of the fuel element's fuel matrix.

Influence of differences in thermophysical properties of fuel element elements WFA and fuel assemblies onto conditions of emergency processes run requires a separate analysis.

Pressure drop in the reactor core in the absence of flow mixing (model [6])

$$\Delta P_{\text{R}} = \kappa_{\text{AS}} \frac{\rho}{2} v_{\text{A}}^2, \quad (7)$$

$$\Delta P_{\text{R}} = \kappa_{\text{WS}} \frac{\rho}{2} v_{\text{W}}^2, \quad (8)$$

where κ_{AS}, κ_{WS} – are the total HdRC of TBC-A fuel assemblies and WFA, respectively; ρ – the coolant density; $v_{\text{A}} = Q_{\text{A}}/F_{\text{A}}$, $v_{\text{W}} = Q_{\text{W}}/F_{\text{W}}$ – the coolant flow average velocity in TBC-A and WFA, respectively; Q_{A}, Q_{W} – volumetric flow of coolant in TBC-A and WFA, respectively; F_{A}, F_{W} – area of passage section at TBC-A and WFA, respectively.

Then from formulas (7) and (8) we get the ratio of average coolant velocities in WFA and TBC-A (rate difference)

$$\kappa_v = \frac{v_{\text{W}}}{v_{\text{A}}} = \sqrt{\frac{\kappa_{\text{AS}}}{\kappa_{\text{WS}}}} < 1. \quad (9)$$

The HdRC values for TBC-A and WFA accordingly to [6] are given in the Table 2.

Table 2. **HdRC values for TBC-A and WFA [6]**

HDRC	Fuel element	
	TBC-A	WFA
HDRC at FA inlet	0,71	1,03
HDRC at FA active part	8,58	12,67
HDRC at FA outlet	2,58	2,49
Total HDRC	11,87	16,19

In accordance with MDA computational modeling results in [6] (after the 1st peak of T_{clad} $\kappa_q \approx$ 700 ºC) and the established HdRC values for WFA and TBC-A (see Table 2) the maximum divergences of current values $\Delta T_{\text{clad}} = T_{\text{clad}}$ (WFA) - - T_{clad} (TBC-A) considering (3) and (9) is no more than 115 ºC (with the greatest influence of the coolant flow velocity onto the different heat exchange modes intensity n = 1), and the maximum WFA fuel cladding temperature is 965 ºC.

The thermodynamic model, which takes into account the mixing of flows from different fuel assemblies with WFA partial loading, is based on the following provisions:

1) the possible difference in the coolant flows rates at different fuel assemblies is taken into account at the entrance to the reactor active zone;

2) the flows are mixing in the rest of the core with the coolant average velocity v_0 establishment (along the entire area of core pass-through section);

3) the necessary condition for flows mixing is the contact placement of different fuel assemblies.

The coolant flow balance equation for the model of partial/ full load mixing at WFA shall be

$$v_0(N_A F_A + N_W F_W) = v_{Ai} N_A F_A + v_{Wi} N_W F_W, \quad (10)$$

where N_A, N_w – TBC-A and WFA quantities respectively; v_{Ai}, v_{wi} – the thermal input average velocity at the entrance to the active zone for TBC-A and WFA respectively (for $N_w \geq 1$).

The coolant flow rates balance equation at full load of TBC-A and WFA partial /full load:

$$v_A N_0 F_A = v_{Ai} N_A F_A + v_{Wi} N_W F_W, \quad (11)$$

where v_A is the coolant average velocity at full WFA load; N_0 – total amount of fuel assemblies in the reactor core.

Pressure drops at the core entrance

$$\Delta P_{ai} = \kappa_{Ai} \frac{\rho}{2} v_{Ai}^2, \quad (12)$$

$$\Delta P_{ai} = \kappa_{Wi} \frac{\rho}{2} v_{Wi}^2, \quad (13)$$

where κ_{Ai}, κ_{wi} – HdRC at the coolant inlet to the active zone for TBC-A and WFA, respectively.

Transforming formulas (10) - (13) we obtain

$$\kappa_{Vi} = \frac{v_{Wi}}{v_{Ai}} = \sqrt{\frac{\kappa_{Ai}}{\kappa_{Wi}}}, \quad (14)$$

$$v_0 = \frac{N_A F_A + N_W F_W \sqrt{\kappa_{Ai}/\kappa_{Wi}}}{N_A F_A + N_W F_W} v_{Ai}, \quad (15)$$

$$v_{Ai} = \frac{N_0 F_A}{N_A F_A + N_W F_W \sqrt{\kappa_{Ai}/\kappa_{Wi}}} v_A. \quad (16)$$

Therefore, the rate of coolant velocity change when WFA loaded respectively to the TBC-A total fuel loading

$$\kappa_{VW} = \frac{v_0}{v_A} = \frac{N_0}{N_0 - (1 - F_W/F_A) N_W}. \quad (17)$$

With a conservative $F_w \approx F_A$ assumption, $\kappa_{vw} = 1$ and there are no differences in the fuel cladding temperature: $\Delta T_{clad} = 0$.

3. Conclusions

1. Based on known results alternative analysis of the computational modeling using code RELAP5/V3.2 for maximum design accident during WFA fuel assemblies diversification at WWER-1000 reactors, it has been established that the calculated maximum fuel cladding temperature is unreasonably overestimated. Opposite to the known results, the allowable fuel cladding temperature (1473 K) safety criterion is not violated for design accidents, should WFA loading into the WWER-1000 active zone be partial either full complete.

2. The differences between the WFA assemblies and the WWER-designed TBC-A, connected with the WFA structures reinforcement, do not reduce the overall safety level in the event of design accidents.

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АНАЛІЗ ЯДЕРНОЇ БЕЗПЕКИ ПРИ ДИВЕРСИФІКАЦІЇ ПАЛИВНИХ ЗБІРОК WESTINGHOUSE НА ВВЕР-1000

Представлено аналіз відомих результатів моделювання максимальної проектної аварії (МПА) кодом RELAP5/V3.2 при диверсифікації паливних збірок Westinghouse (WFA) в реакторах типу ВВЕР-1000. Відповідно до відомих результатів розрахункового моделювання МПА кодом RELAP5/V3.2 при максимально допустимій температурі води в теплообміннику системи аварійного охолодження ВВЕР (90 °C) температура оболонок тепловиділяючих елементів досягає 1320 °C і перевищує допустиму межу ядерної безпеки (1200 °C). Таким чином, згідно з відомими результатами МПА з WFA переходить зі статусу «проектної» аварії в статус «важкої» аварії і означає зниження безпеки по відношенню до проектних паливних збірок ТВЗ-А. Наведений у роботі альтернативний аналіз МПА з WFA показав, що на відміну від відомих розрахунків межа ядерної безпеки по максимально допустимій температурі оболонок тепловиділяючих елементів не порушується і не знижує загальний рівень безпеки при диверсифікації ВВЕР паливними збірками WFA.

Ключові слова: безпека, диверсифікація паливних збірок.

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АНАЛИЗ ЯДЕРНОЙ БЕЗОПАСНОСТИ ПРИ ДИВЕРСИФИКАЦИИ ТОПЛИВНЫХ СБОРОК WESTINGHOUSE НА ВВЭР-1000

Представлен анализ известных результатов моделирования максимальной проектной аварии (МПА) кодом RELAP5/V3.2 при диверсификации топливных сборок Westinghouse (WFA) в реакторах типа ВВЭР-1000. Согласно известным результатам расчетного моделирования МПА кодом RELAP5/V3.2 при максимально допустимой температуре воды в теплообменнике системы аварийного охлаждения ВВЭР (90 °C) температура оболочек тепловыделяющих элементов достигает 1320 °C и превышает допустимый предел ядерной безопасности (1200 °C). Таким образом, согласно известным результатам МПА с WFA переходит из статуса «проектной» аварии в статус «тяжелой» аварии и означает снижение безопасности по отношению к проектным топливным сборкам ТВС-А. Приведенный в работе альтернативный анализ МПА с WFA показал, что в отличие от известных расчетов предел ядерной безопасности по максимально допустимой температуре оболочек тепловыделяющих элементов не нарушается и не снижает общий уровень безопасности при диверсификации ВВЭР топливными сборками WFA.

Ключевые слова: безопасность, диверсификация топливных сборок.

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