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ANALYSIS OF EXPERIENCE, SAFETY AND PROSPECTS OF DIVERSIFICATION OF NUCLEAR FUEL AT NUCLEAR POWER PLANTS

Об'єктом дослідження є проектні тепловиділяючі збірки (ТВЗ-А) на ядерних енергоустановках з водо-водяними енергетичними реакторами, розташованими в Україні. Проведено аналіз досвіду та перспектив диверсифікації проектних тепловиділяючих зборок водо-водяних енергетичних реакторів альтернативними тепловиділяючими зборками Westinghouse Electric Company. Аналіз виконано на основі заходів і результатів диверсифікації тепловиділяючих зборок на атомній електростанції Темелин (Чехія), а також Південно-Українській та Запорізькій атомних електростанціях (Україна). У результаті проведеного аналізу показано, що диверсифікація проектних тепловиділяючих зборок альтернативними тепловиділяючими зборками Westinghouse Electric Company забезпечує необхідні умови ядерної безпеки щодо максимально допустимої температури оболонок тепловиділяючих елементів і температури ядерного палива.

У роботі визначена необхідність додаткового аналізу ядерної безпеки і надійності обладнання реакторного контуру щодо умов виникнення критичних гідродинамічних ударів при диверсифікації тепловиділяючих зборок. Виявлено, що відомі результати аналізу ядерної безпеки при диверсифікації тепловиділяючих зборок традиційними підходами моделювання аварій недостатньо обґрунтовані. А також істотно залежать від негативних ефектів «різниці результатів моделювання аварій різними користувачами однакових кодів» та «різниці результатів моделювання аварій різними кодами». Крім того, відомі детерміністські коди не моделюють умови і наслідки гідродинамічних ударів та різних видів теплогідродинамічної нестійкості в реакторному контурі. Показано, що необхідна розробка альтернативних методів аналізу ядерної безпеки і надійності обладнання систем, важливих для безпеки, при диверсифікації тепловиділяючих зборок, які не залежать від наведених вище негативних ефектів. Розрахунковий аналіз впливу швидкості теплоносія на коефіцієнти зовнішньої тепловіддачі визначив, що умови безпеки щодо допустимої температури тепловиділяючих елементів Westinghouse Electric Company забезпечені аж до максимальної проектної температури 90 °С у теплообмінниках систем безпеки.

Ключові слова: *диверсифікація ядерного палива, ядерна безпека, надійність активної зони ядерних реакторів.*

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1. Introduction

An urgent and priority issue for Ukraine's nuclear energy is the diversification of nuclear fuel, which consists in replacing design fuel assemblies (FAs) with alternative ones. As an alternative fuel assembly at the Ukrainian nuclear power plants (NPPs), the introduction of Westinghouse Electric Company (WFA) fuel assemblies has begun.

Currently, 33 power units with WWER (water-to-water power reactor) reactors are in operation in the countries of the European Union (EU) and Ukraine (Table 1).

The main supplier of nuclear fuel to European and Ukrainian nuclear power plants with WWER is Russia. However, the experience of other nuclear markets shows that the monopolization of the supply and storage of nuclear fuel adversely affects both safety and competition to improve technology and economy [1]. In addition, there is positive experience with mixed loading of the reactor core with nuclear fuel from various suppliers.

Table 1

Power units in Europe and Ukraine

Country	NPP, power unit	Type of power unit
Bulgaria	Kozloduy 5–6	WWER-1000
Czech Republic	Dukovany 1–4	WWER-440
	Temelin 1–2	WWER-1000
Finland	Loviisa 1–2	WWER-440
Hungary	Paks 1–4	WWER-440
Slovakia	Bohunice 3–4	WWER-440
	Mochovce 1–2	WWER-440
Ukraine	Khmelnytskyi 1–2	WWER-1000
	Rivne 1–2	WWER-440
	Rivne 3–4	WWER-1000
	South Ukraine 1–3	WWER-1000
	Zaporizhzhia 1–6	WWER-1000

The transnational Westinghouse Electric Company has many years of experience supplying nuclear fuel for various types of reactors and is a promising supplier of alternative fuel for WWER (including Ukraine).

This paper presents an analysis of the experience and safety of introducing diversification of fuel assemblies at nuclear power plants with WWER reactors, as well as further prospects for the diversification of nuclear fuel at Ukrainian nuclear power plants.

2. The object of research and its technological audit

The object of research is the design fuel assemblies (FA-A) at nuclear power plants with WWER reactors located in Ukraine. Power plants with WWER-type projects are widely described in the scientific and technical literature and do not require a detailed description. In connection with the situation of replacing the supplier of nuclear fuel at the aforementioned hazardous facilities, it is necessary to consider the observance of safety criteria when adapting fuel to existing technological regulations.

3. The aim and objectives of research

The aim of research is analysis of the observance of safety criteria when replacing a nuclear fuel supplier and identifying technological problems that need to be solved when implementing this procedure. To achieve this aim it is necessary to perform the following objectives:

1. To analyze the experience of diversification of design fuel assemblies (FA-A) in nuclear power plants with WWER reactors by fuel assemblies of Westinghouse Electric Company (WFA).
2. To analyze the future prospects of diversification of nuclear fuel at Ukrainian nuclear power plants.

4. Research of existing solutions of the problem

Of undoubted interest is the ten-year experience in diversifying Westinghouse fuel assemblies at Temelin NPP (Czech Republic) [2]. In the process of diversification of Westinghouse FAs at Temelin NPP, the following main problems were identified:

- problems with rod clusters control assembly (RCCA) during operation – incomplete introduction of control rods (IRI);
- bending of fuel assemblies/fuel rods;
- excessive lengthening of fuel assemblies;
- leaking fuel rods.

Incomplete introduction of control rods (IRI). Some RCCA (on both blocks) did not fall completely to the bottom position due to the root cause of the change in the geometry of the fuel assemblies. The worst case occurred at block 1, when test tests showed that two clusters had stopped above the hydraulic shock absorber, which led to an unscheduled stop for refueling. Subsequently, the drop function was tested: the test interval was 30 days (in 2005) and 150 days (in 2010). Based on preliminary tests of the fuel supplier, a safety assessment was carried out and forecasts of the number of IRI situations for upcoming tests were obtained, which provided better preparedness for these situations.

In 2005, modernizations were made that affected the connection of the drive rod and control rods: additional holes were drilled, the weight of the rods was increased, etc.

In 2006–2007 the following upgrades were carried out:

- design of the pipe-in-pipe shock absorber and the upper fuel assembly head (phase 0);
- alignment of the fastening elements of the fuel rods;
- new structural materials for the cladding of fuel rods and other components of a fuel assembly (phase 1X).

Bending of fuel assemblies and fuel rods. Deformations of fuel rods and fuel assemblies (that is, a deviation from their initial geometric shape and elongation of fuel assemblies – «radiation growth») are a common problem for all power units with WWER and PWR.

The first core loading at the Temelin NPP was designed taking into account the reduction of the flow section to 51 %. All subsequent core designs were calculated taking into account conservative assumptions that fuel rods may even touch each other. As well, the sequence of rearrangement of the fuel assemblies was changed, special simulators of the fuel assemblies were used when installing bent and curved fuel assemblies.

The results of the effect of modernization of the cladding material of the fuel rods on the effects of elongation of the fuel assemblies are given in [2].

Leaking fuel rods. The active zones of the reactors installed at Temelin NPP contain 163 fuel assemblies and 312 fuel rods in each fuel assembly and are designed based on the assumption that a certain number of fuel rods has a design leak. At Temelin NPP, fuel rods are checked for leaks by monitoring specific activity in the 1st circuit – the limit of total specific activity is $\leq 3.7 \cdot 10^9$ Bq/l (Table 2).

Table 2

Results of the recorded maximum activity in the coolant of the first circuit on two units ($\times 10^{-6}$ Bq/l)

Year	2003	2004	2005	2006	2007	2008	2009	2010
Unit 1	5.9	2.4	3.6	6.3	7.9	7.1	6.3	4.3
Unit 2	2.7	7	3.3	6.4	8.5	5.8	6.3	6.6

To search for leaks, the Sipping System is used based on the gas method of constant monitoring of Xeon Xe133 concentration.

Mobile equipment (available on both units) was designed and manufactured by Westinghouse (USA). During shutdown, samples are collected on the floor of the reactor compartment and then transported to a fresh fuel container storage pit next to the spent fuel storage pool. The reloading machine moves the fuel assemblies to the mobile transport device and extracts from there. The fuel assemblies are placed on a rotating table, which allows to rotate it 360 degrees during the verification process. Thus, the device is designed to provide a complete check of fuel and fuel assemblies as a whole (Table 3).

A verification system for fuel assemblies is used to determine the position of a fuel rod with a leak. Then it is placed in a basket for storage of damaged fuel rods (FFRSB). Damaged fuel rods are replaced with fuel simulators.

The root cause of leaking fuel rods is fretting «fuel rod – grating», which indicates the vibration of the fuel rod in the spacer grating.

A typical example of damage to a fuel rod can be an incident on the 2nd block after reconstruction at BE24 as-

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semblies. One fuel rod with a leak was removed and replaced with a stainless steel rod. After removing the fuel rod (approximately 150 cm), the rod suddenly twitched and broke. A visual inspection of the fracture site revealed that this was due to secondary hydration, although there was no visual sign of significant hydration. Extensive fretting was discovered between the fuel rod and the grating throughout the cladding. Visual examination at the ends of the kink of the two parts of the fuel rod showed the integrity of the fuel pellets in the absence of empty areas, which indicates the absence of loss of fuel pellets in the event of a kink.

Table 3

Review of detected fuel leaks at Temelin NPP

End of cycle	Number of leaks	Repaired leaks	Number of fuel rods with a leak
Unit 1			
2	1	0	0
3	5	1	1
4	6	1	1
5	6	6	8
5A	4	2	3
6	7	7	10
7	3	1	1
8	0	0	0
Unit 2			
2	3	2	5
3	10	2	4
4	5	2	2
5	7	2	3
6	5	0	0
7	5	1	2

The Post Irradiation Inspection Program (PIIP) has been structured to meet the requirements of the Czech Regulator (SUJB). Additional confirmations of the compatibility of materials, analytical methods of justification and verification, a general demonstration of thermohydraulic processes were obtained (8 fuel assemblies were selected).

The verification program included:

– FA measurements – visual inspection, measurement of bending, torsion and general elongation;

– measurements of an individual fuel rod – visual inspection, corrosion measurement, profilometry;

– verification of RCCA clusters – total wear and bore of pipes.

Positive PIIP Results. Visual examinations revealed the absence or slight corrosion on the surface of the cladding of the fuel rods, which confirmed the compatibility of the zirconium alloy with the water-chemical regime of WWER. No wear of RCCA guide tubes detected (criterion – wear less than 10 %).

Negative. Measurements of bending, torsion and length of fuel assemblies showed some differences from design fuel assemblies. A visual inspection of the fuel rods with a leak revealed the presence of fretting.

The experience of diversification of Westinghouse fuel assemblies at Temelin NPP allows to draw the following conclusions:

1. Diversification of non-project fuel is, of course, a rather lengthy and complex process, but it can be successfully implemented with the effective cooperation of the supplier and the operating organization for the modernization of the reactor core.

2. The revealed negative effects (fuel rod deformations, fuel assemblies, fretting, etc.) are mainly characteristic of WWER design fuel assemblies and are a common problem requiring further improvements.

In accordance with the Ukraine Nuclear Fuel Qualification Program (UNFQP), Westinghouse and the United States Department of Energy, with the participation of the Pacific Northwest National Laboratory (PNNL), supply Ukraine with alternative nuclear fuel to restart WWER-1000 fuel assemblies (WFA). According to the Swedish branch of Westinghouse [3], in 2005 the first six test assemblies (Lead Test Assemblies – LTA) for WWER-1000 were loaded at Unit 3 of the South Ukrainian NPP and demonstrated full compatibility with the design nuclear fuel and reactor designs. Further operation demonstrated the adequacy of operational limits and reliability requirements for several reactor fuel campaigns. After the completion of the planned four fuel campaigns, the average burnup depth is more than 43 MW·day/kg of uranium. The final LTA verification program at the end of the 20th campaign cycle in 2010 demonstrated the acceptability of Zr–1 % Nb alloys (as a spacer material) and ZIRLO® (as a cladding of fuel rods and structural components) for WWER-1000. The fall time of the assembly of absorbing rods and resistance forces were within the accepted values [3]:

– all assemblies visually without signs of anomalies;

– all spacer gratings remained in design positions along the fuel assembly axis, without signs of excessive corrosion;

– the surface of the fuel rods without deposits and distortions, and corrosion was within the expected levels;

– there was no fretting and other types of wear of peripheral fuel rods; RCCA cluster simulator traction was acceptable, which determined the absence of any significant LTA bending or bending.

The LTA program shows that the design of the alternative WFA fuel is in principle suitable for use with a fully loaded WWER-1000 core [3]. The first load of 42 WFAs was made at the Westinghouse nuclear fuel production facility in Sweden and loaded at Unit 3 of the South Ukrainian NPP in February 2010. This WFA load has accumulated another cycle of safe operation.

In [4], a comparative analysis of structural differences and compatibility of Westinghouse LTA-2 and WFA fuel assemblies and design assemblies for WWERs is given (Table 4), which established the following:

1. All fuel assemblies examined have hexagonal designs with the same assembly pitch (23.6 cm).

2. The fuel matrices of the LTA-2 and WFA fuel rods are solid, while the fuel assemblies are ring-shaped with helium inside. In the general case, these differences can affect the conditions of heat and neutron physical processes.

3. LTA-2 contains six massive stiffening rods (the so-called «P-rods»), and the corresponding stiffness angles are installed in the FA-A.

4. Fuel rods in WFA fuel assemblies have three different levels of fuel enrichment (fuel rods with the lowest level are located on the periphery of the fuel assembly), and fuel assemblies (adapted fuel assemblies) have only two. Both types of fuel assemblies contain six fuel rods with gadolinium, 18 guide tubes and one central tube with certain differences in size.

Table 4

Key fuel design features

Parameter	WFA/LTA-2	FA-A
Fuel rod	–	–
The pitch of the fuel rods, mm	12.75	12.75
The number of fuel rods in the fuel assembly	312	312
Sheath material	ZIRLO	Zirconium alloy E110
The outer diameter of the cladding, mm	9.14	9.1
The inner diameter of the cladding, mm	8	7.72
Cladding thickness mm	0.57	0.7
Fuel material	UO_2	UO_2
The outer diameter of the tablet, mm	7.84	7.57
The inner diameter of the tablet, mm	–	1.4 or 1.5
Number of fuel rods	306	306
Number of fuel rods with 5 % Gd203	6	6
Enrichment of fuel rods with Gd_2O_3	3.0 %	4.4 %
Burnable absorber	–	–
Burnable absorber material	Gd	Gd
Helium pressure in a fuel road, kPa	1896.0	1896.0
Guide tubes	–	–
Number of guide tubes	18.0	18.0
The outer diameter of the guide tubes, mm	12.6	12.6
The inner diameter of the guide tubes, mm	11.0	11.0
Guide tube thickness, mm	0.800	0.800
Guide tube material	ZIRLO	Zirconium alloy E635
Central measuring tube	–	–
Number of tubes	1	1
Tube outer diameter mm	12.6	13.0
The inner diameter of the tube, mm	11.0	11.0
Tube thickness mm	0.800	1.000
Tube material	ZIRLO	Zirconium alloy E635
Reinforcing corners	–	–
Angle thickness, mm	–	0.65
Corner material	–	Zirconium alloy E635
P-Rod Data	–	–
The number of P-rods in the fuel assembly (LTA-2)	6	–
Material	ZIRLO	–
Outer diameter mm	9.75	–

The main WFA modernization measures for Ukrainian nuclear power plants were aimed at preventing known problems for PWR and WWER reactors. These problems are associated with the curvature/deformation during operation of fuel rods and other fuel assemblies (including, based on the experience of the first Westinghouse delivery of VV6 fuel assemblies for Temelin NPPs [5]), which lead to incomplete absorption rods (Incomplete Rod Insertion – IRI).

It should be noted that the initial VV6 fuel assemblies delivered at Temelin NPPs significantly differ from the WFAs intended for Ukrainian NPPs. The main differences affecting the possibility of incomplete introduction of absorbing rods for each of these two FA assemblies:

1) material of the ZIRLO guide tube for WFA upon irradiation provides growth and creep levels are lower than for VV6 (zircaloy-4), which leads to a decrease in the bend of the fuel assembly;

2) WFA design includes 16 distancing gratings, while in VV6 only 9 is used. Increasing the number of gratings reduces the distance between the gratings, which affects the stiffness of the guide tubes and reduces the amount of bending and, as a result, reduces the traction forces for the RCCA cluster.

The design of the VV6 type fuel assemblies was improved by adding a pipe-in-pipe shock absorber to eliminate design flaws and use ZIRLO guides. In addition, the method of fastening the spacer gratings was replaced from one protrusion to a double to increase the rigidity of the lateral edge of the fuel assembly structure. The method of attaching spacer gratings using a double protrusion is also embodied in the WFA design for Ukrainian reactors.

The effect of these upgrades to the fuel assembly design is presented in terms of an increase in the stock of lateral stiffness and shows a relative comparison with the transverse rigidity of the VV6 design. Mechanical tests have confirmed that the specified WFA upgrade is equivalent to the successfully operating Westinghouse square fuel rods.

As a result of modeling neutron-physical processes at various distributions of fuel assembly enrichment in [4] it was established:

1) P-rod increases energy release in adjacent fuel rods; however, the low enrichment distribution in the LTA-2 assembly effectively compensates for the increase in power caused by P-rods;

2) influence of P-rods on the coefficient of uneven energy release in WFA is secondary and amounts to an increase of the order of 1 %;

3) if there are three or less FA-A on the same side as the WFA, then the coefficient of uneven energy release increases slightly; but if FA-A symmetrically surround the WFA, then the coefficient of uneven energy release decreases.

The design features of WFA are as follows:

1. Detachable top head: spring-loaded device for optimal spring stiffness. The design allows to check and repair each individual fuel rod, and also eliminates the potential risk from loose parts.

2. The upper grate is made of Inconel alloy (a family of austenitic heat-resistant alloys) and has springs to minimize the bending of fuel rods.

3. The lower grating of Inconel with higher spring stiffness, which ensures the retention of the rod throughout the entire service life and eliminates the grating-fuel rod fretting.

4. Intermediate gratings are made of Zr–1 % Nb by a double method for increase of lateral rigidity and resistance to distortions.

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5. The shanks of the guide tubes are made of ZIRLO to increase and bend resistance of the assembly.

6. The cladding of the fuel rods is made of ZIRLO, which has improved mechanical properties and corrosion resistance (including resistance to changes in the water-chemical regime of the coolant).

7. The material of the burnable absorber Gd_2O_3 provides a reduction in the cost of the fuel cycle with a high fuel burnup.

The main differences between the design of WWER FA-A and alternative WFAs, consisting in the design of fuel assemblies and fuel rods, as well as in the spatial distribution of nuclear fuel enrichment, can affect nuclear safety.

The structural differences between fuel assemblies are mainly due to the fact that additional «reinforcing» WFA designs lead to a relative increase in the total coefficient of hydraulic resistance (CHR). The increase in CHR, ceteris paribus, leads to a decrease in the flow rate and speed of the coolant, and, accordingly, to deterioration in the conditions of heat transfer and an increase in the temperature of the claddings of fuel rods, which must be taken into account when analyzing safety.

Differences of WFA assemblies from design for WWER FA-A related to the strengthening of WFA structures are aimed at improving the reliability of reactor structures.

However, technical solutions to strengthen the WFA are not effective enough to prevent the root causes of deformations/integrity of fuel rods caused by both external (with respect to the cladding) and internal cyclic and/or static loads. In particular, as a result of the operational inspection of the WFA at the South Ukrainian NPP (after the 20th campaign), one of the fuel rods was depressurized: for I-131, the activity was $7.5\cdot10^{-8}$ Ku/kg; for Xe-133 – $2.6\cdot10^{-6}$ Ku/kg [3].

Internal unacceptable for reliability loads can be caused by processes directly in the fuel matrix and gas gap as a result of violations of normal operation conditions. External – by various types of thermohydrodynamic instability of the coolant (including those contributing to an increased vibrational state) [5]. Strengthening the fuel assembly structures and replacing the cladding material of a fuel rod does not generally prevent the occurrence of both internal and external loads that are unacceptable for the reliability of a fuel rod. For example, under conditions of an increased vibrational state of fuel rod or thermoacoustic instability of a coolant, additional fastenings can reduce the deformation amplitudes, however, they do not prevent the processes that caused these phenomena. As a result of this, the locations of the fuel rod attachments experience an increased cyclic load, which after some time can lead to unacceptable deformation or damage to the integrity of the fuel rod cladding.

The design features of WFA associated with variable enrichment of nuclear fuel along the height of the fuel rod and increased energy release from neighboring fuel rods in the places of the «bearing» assembly structures (P-rods) also do not reduce the overall level of safety. Elimination of the negative consequences of these features during the operation of WFA is possible through the effective modernization of systems for in-reactor monitoring [6] and the placement of fuel rods with lower enrichment at the sites of P-rods [4].

In [7], a preliminary expert analysis of the feasibility of safety criteria for mixed loadings of FA-A and WFA reactors of the WWER type was performed using the RELAP5/V3.2 code. The analysis showed that the CHR differences determine the temperature of the cladding of the WFA fuel rods T_{cl} several hundred degrees higher than the corresponding values for the FA-A, and at the maximum permissible design temperature of the emergency core coolant cooling system (ECCS) 90 °C, safety criteria are not fulfilled at all: $T_{cl}>1200$ °C (Table 5). It should be borne in mind that in the analysis of beyond design basis accidents, the influence of these discrepancies in the T_{cl} values can be even more significant for assessing nuclear safety criteria.

Table 5

Accident simulation results on the difference in the temperature of the claddings of the fuel rods WFA and FA-A ($\Delta T_{cl}=T_{cl}$(WFA)$-T_{cl}$(TA-A)) for different core loading options [7]

Loading option	ECCS water temperature, °C	1st peak of cladding temperature, °C	ΔT_{cl} of the 1st peak of T_{cl}, °C	2nd peak of cladding temperature, °C	ΔT_{cl} of the 2nd peak of T_{cl}, °C
42 WFA+121 TA-A	70	867	–7	1099	250
163 TA-A	70	874	–	849	–
163 WFA	70	850	–24	1074	225
163 WFA	90	850	–24	1320	–

Thus, according to the results of accident modeling obtained in [7], loading WFA into WWER reduces the overall level of nuclear safety. To exclude these conclusions, it is proposed to reduce the conservatism of nuclear safety analysis by applying a coolant temperature of ECCS of not more than 70 °C. However, this approach requires a review of the design safety criteria and coordination with the design and engineering organizations of the WWER-1000 reactor plant.

However, when modeling generalized design basis accidents in [7], individual insufficiently substantiated and/or «excessively» conservative provisions were used:

1. The lack of mixing of coolant flows in different FAs.

The WFA/FA-A designs, as well as the hydrodynamic regimes of developed turbulence, determine the possibility of intensive mixing of the coolant flows over a large part of the FA height. A necessary condition for mixing flows at most of the FA height is the contact location of the «case-free» FAs.

A necessary condition for the complete hydrodynamic mixing of coolant flows in case-free FAs:

$$\xi_{tr} \ll \xi_l,$$

where ξ_{tr}, ξ_l – hydraulic resistance coefficient of the transverse and longitudinal heat carrier flow, respectively.

The coefficient of hydraulic resistance to flux mixing ξ_{tr} for the characteristic «packing» of fuel rods in FA-A/WFA can be estimated from the well-known semi-empirical dependence [8]:

$$\xi_{tr} = \frac{0.3164}{\mathrm{Re}_{tr}^{0.25}},$$

where Re_{tr} – the Reynolds criterion for the transverse velocity of the coolant.

Taking into account the verification results of modeling [7, 9], the condition for complete hydrodynamic mixing of the coolant flows is fulfilled for both operating and emergency conditions.

In addition, it should be noted that the strengthening of the WFA structures is an additional «turbulizer» of the active zone, intensifying the mixing of flows with the contacting fuel assemblies and increasing the critical heat fluxes q_{hf}, which determine the conditions of supercritical heat transfer. The critical heat flux density (CHF) in the channel with «turbulators» is determined by the known relation [10]:

$$q_{hf} = q_{hf0}\sqrt{\sum_{i=1}^{n}\Delta q_i^2},$$

where q_{hf0} – CHF without «turbulizers»; Δq_i – the efficiency of one «turbulizer»:

$$\Delta q_i = 4.5\cdot 10^{-3}\rho w\left(\frac{P}{P_{cr}}\right)^{0.25}(1-x)^{0.7}\left(\frac{F_m}{F_0}\right)\exp\left(-\frac{z_i}{l_r}\right),$$

where ρw – the mass flow rate of the coolant; P – the pressure; P_{cr} – critical pressure; x – the mass vapor content, F_m/F_0 – the ratio of the midsection section of the «turbulizer» to the area of the flow area of the coolant; z_i – the distance from the i-th «turbulator» to the section under consideration; l_r – relaxation length:

$$l_r = 0.18(1-2\rho w x^3)^{0.12}.$$

The analysis shows that the effect of increasing CHF due to additional «turbulators» of WFA is more significant in comparison with the effect of decreasing CHF as a result of a relative decrease in consumption through fuel assemblies, as [10].

In computational and experimental studies [11], the mixing processes in the pressure and collection chambers of WWER-type reactors were analyzed for modes with asymmetric operation of the circulation loops. The obtained results confirmed the validity of the models of complete mixing of the coolant in the regimes with asymmetric operation of the circulation loops even in conditions of «sluggish» natural circulation. The considered conditions and modes are much more conservative with respect to the processes of mixing the coolant flows in the contacting case-free fuel assemblies, which determines the groundlessness of the models of the absence of mixing of the coolant flows in the contacting FA-A and WFA.

2. An increase in the power of heat generation by 10 % after the reactor shutdown by the emergency protection system.

After emergency protection is triggered, the heat dissipation power of nuclear fuel decreases sharply and heat transfer processes are reduced to the removal of residual heat due to the influence of γ- and β-radiation, «delayed» neutrons, the thermal state of reactor structures and other effects. The power of residual heat after a few seconds of the emergency process is not more than 7 % of the power of the reactor heat in nominal operating conditions [10]. Moreover, the conservative increase in residual heat by 10 % adopted in [7] requires clarification.

3. There is no analysis of the reliability of the reactor loop equipment during diversification of FAs. The most dangerous phenomenon in terms of reliability and operability of the reactor loop equipment in operating, transient and emergency conditions can be hydrodynamic shocks, which are accompanied by a pulsed high-amplitude hydrodynamic effect [12, 13].

5. Methods of research

The differences in the designs and thermal resistances of the fuel rods FA-A and WFA are mainly associated with different thicknesses of the fuel matrix and gas gap, the material of the claddings. Also, with the discrepancy of thermophysical properties determined by different manufacturing techniques and enrichment of nuclear fuel. For example, differences in the thickness of the fuel matrices of the WFA and FA-A fuel rods [4] leads to a relative increase in the thermal resistance R_f of the WFA fuel rods relative to the fuel assemblies of the FA-A. Moreover, an increase in R_f has a double effect: on the one hand, an increase in R_f leads to a decrease in the temperature of the cladding of a fuel rod (ceteris paribus), and on the other hand, to a relative increase in the temperature of nuclear fuel.

Let's consider the effect of differences in WFA (when they partially load the reactor core in the framework of the «non-mixing» flows) from the design FA-A on the change in the safety level of a WWER-based reactor based on the fundamental principles of thermodynamics of nuclear reactors.

The main reason for changing the hydraulic characteristics of the WWER core in this case is the consequences of modernization to strengthen the structure and increase the reliability of the WFA, including to prevent deformation of the fuel rods. The resulting increase in CHR (due to additional WFA designs) leads to a relative decrease in the flow rate and average coolant velocity, and, accordingly, to a decrease in the heat transfer intensity on the outer surface of the fuel rods.

The boundary condition of heat transfer on the surface of fuel rods:

$$q_f = \alpha(T_{cl} - T_c),$$

where q_f – the heat flux density from the fuel rod, determined by the thermal state and properties of the fuel matrix and the thermal resistance of the fuel rod R_t; α – the heat transfer coefficient on the surface of the fuel rod; T_{cl}, T_c – the temperature of the cladding of the fuel rod and coolant, respectively.

Then the temperature differences between the claddings of the WFA and FA-A fuel rods at the current time (assuming q_f is identical):

$$DT_{cl} = T_{cl}(\text{WFA}) - T_{cl}(\text{FA-A}) = = q_f\left[\frac{1}{a(\text{WFA})} - \frac{1}{a(\text{FA-A})}\right].$$

The rate of change in the intensity of heat transfer:

$$\kappa_f = \frac{\alpha(\text{WFA})}{\alpha(\text{FA-A})}$$

in this case is determined by the rate of change of the average velocity of the coolant v:

$$\kappa_f = \frac{v(\text{WFA})}{v(\text{FA-A})}.$$

In accordance with the known phenomenological dependences for the modes/conditions of heat transfer and phase state of the coolant in nuclear reactors (including those used in different correlations of the RELAP5 code):

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

where $n \leq 1$ [10].

Accordingly, the dependence of indicators of changes in heat transfer intensity and coolant velocity:

$$\kappa_f = (\kappa_v)^n,$$

and the current discrepancies in the temperature of the cladding of the fuel rods:

$$\Delta T_{cl} = q_f \left[\frac{1}{(\kappa_v)^n \alpha(\text{WFA})} - \frac{1}{\alpha(\text{FA-A})} \right] =$$

$$= \frac{q_t}{\alpha(\text{FA-A})} \frac{1-(\kappa_v)^n}{(\kappa_v)^n}. \quad (1)$$

Thus, the discrepancies in the current temperatures of the claddings WFA and FA-A under the assumption that the structures and thermal resistance R_f of the fuel rods are identical are determined by the indicators of the change in the average coolant velocity κ_v and the ratio of the intensity of the internal and external heat transfer of the fuel rods:

$$\kappa_q = \frac{q_f}{\alpha(\text{FA-A})}.$$

In the absence of the identity of the structures and thermal resistance of the fuel rods, the current temperature differences of the claddings of WFA and FA-A:

$$\Delta T_{cl} = \frac{1-(\kappa_v)^n}{(\kappa_v)^n} \frac{q_f(\text{WFA}) - q_f(\text{FA-A})}{\alpha(\text{FA-A})}.$$

The relationship between the heat flux density q_t and thermal resistance R_f of a fuel rod:

$$q_f = \frac{T_{tm} - T_{cl}}{R_f}, \quad (2)$$

where in an acceptable «flat» approximation:

$$R_f = \frac{\delta_{cl}}{\lambda_{cl}} + \frac{\delta_g}{\lambda_g} + \frac{\delta_f}{\lambda_f}, \quad (3)$$

where Δ_{cl}, Δ_g, Δ_f – the thickness of the cladding, gas gap and fuel matrix of the fuel rod, respectively; λ_{cl}, λ_g, λ_f – the thermal conductivity coefficient – the maximum temperature of nuclear fuel in the central part of the fuel matrix of a fuel rod.

6. Research results

An analysis of dependences (2) and (3) shows that the differences in the thickness of the fuel matrix and the gas gap of the fuel rods WFA and FA-A ($\Delta_f(\text{WFA})/\Delta_f(\text{FA-A}) = 1.1$; $\Delta_g(\text{WFA})/\Delta_g(\text{FA-A}) = 1.1$ [4]) determine the discrepancy $q_f(\text{WFA})$ and $q_f(\text{FA-A})$:

$$q_f(\text{WFA}) = 0.9 q_f(\text{FA-A}).$$

The effect of differences in the material and the thickness of the cladding of a fuel rod on q_f is even less significant.

A relatively lower value of $q_t(\text{WFA})$, ceteris paribus, on the one hand, decreases the temperature of the cladding of a fuel rod, and on the other hand, increases the temperature of nuclear fuel in the central part of the fuel matrix of a fuel rod.

The pressure drop of dissipative losses in the reactor core in the absence of flow mixing (analog of model [7]):

$$\Delta P_a = \kappa_{AS} \frac{\rho}{2} v_A^2, \quad (4)$$

$$\Delta P_a = \kappa_{WS} \frac{\rho}{2} v_W^2, \quad (5)$$

where κ_{AS}, κ_{WS} – the total CHR of FA-A and WFA, respectively; ρ – the density of the coolant; $v_A = Q_A/F_A$, $v_W = Q_W/F_W$ – the average flow rate of the coolant in FA-A and WFA, respectively; Q_A, Q_W – volumetric flow rate of the coolant in the FA-A and WFA, respectively; F_A, F_W – the flow area of the FA-A and WFA, respectively.

Then, from the formulas (4) and (5), the ratio of the average coolant speeds in WFA and FA-A follows (an indicator of the difference in speeds):

$$\kappa_V = \frac{v_W}{v_A} = \sqrt{\frac{\kappa_{AS}}{\kappa_{WS}}} < 1. \quad (6)$$

Values of CHR for FA-A and WFA in accordance with [7] are given in Table 6.

Table 6

Values of hydraulic resistance coefficients of FA-A and WFA [7]

CHR	FA	
	FA-A	WFA
CHR at the entrance to the fuel assembly	0.71	1.03
CHR of the active part of the fuel assembly	8.58	12.67
CHR exit from fuel assemblies	2.58	2.49
Total CHR	11.87	16.19

In accordance with the results of computational modeling of MPA in [7] (after the 1st peak of $T_{cl} \cdot \kappa_q \approx 700$ °C) and the established values of CHR of WFA and FA-A (Table 6), the maximum discrepancies of current values $\Delta T_{cl} = T_{cl}(\text{WFA}) - T_{cl}(\text{FA-A})$ according to (1) and (6) are no more than 115 °C. The maximum cladding temperature of the WFA fuel rods is 965 °C.

For models that take into account the effects of intercassette mixing of coolant flows in «case-free» fuel assemblies, the maximum calculated values of ΔT_{cl} will be much less.

The more realistic estimates of the maximum values of $\Delta T_{cl} \leq 115$ °C obtained are significantly less than the corresponding value in [7]: 257 °C (load 42WFA+163FA-A) for the period from the 1st and 2nd peak of the temperature of the claddings of fuel rods (Table 5).

According to the results of modeling a design basis accident when the main circulation pump (MCP) is jammed, the RELAP5/M3.2 code [7, 14] established significant differences in the calculated value of the maximum temperature of the claddings of fuel rods WFA: 751 °C in [7] and 928 °C in [14]. Such results may be the result of the long-known so-called «user effect». The main reasons for the «user effect» are related to the difference in the nodalization schemes of computational modeling.

7. SWOT analysis of research results

Strengths. This research summarizes information on the experience of diversification of design fuel assemblies (FA-A) in nuclear installations with WWER reactors and shows the possibility of using indirect analysis methods in the context of this topic.

Weaknesses. When conducting such studies, the analyzed information can be verified by several sources and in the time interval considered over the past 5 years.

Opportunities. The research is typical of Ukraine and can be disseminated in countries where a narrow segment of energy production is used by power plants with WWER reactors of old projects.

Threats. Research in this matter does not require additional capital costs in production and can be updated when new data is received.

8. Conclusions

1. An analysis is made of the experience of diversification of design fuel assemblies (FA-A) in nuclear power plants with WWER reactors by fuel assemblies of Westinghouse Electric Company (WFA).

In the course of the research, well-known developments on safety justification during diversification by WFA assemblies were analyzed and use was made while observing safety criteria and monitoring these criteria by indirect methods.

An alternative calculation analysis of the influence of the coolant speed on the heat transfer coefficients showed that the safety conditions for the allowable temperature of the claddings of the WFA fuel rods are provided up to a maximum design temperature of 90 °C in the heat exchangers of the safety systems.

2. It is shown that the direction of scientific research on the diversification of nuclear fuel at Ukrainian nuclear power plants requires a vector for the study of dry fuel storages.

References

1. Kirst, M., Benjaminsson, U., Önneby, C. (2015). Diversification of the VVER, Fuel Market in Eastern Europe and Ukraine. *ATW, 60 (3),* 63–87.
2. Daniel, E., Milisdörfer, L. (2010). 10 years of experience with Westinghouse fuel at NPP Temelín. *International Topical Meeting VVER-2010 Experience and Perspectives.* Prague, 124–137.
3. Höglund, J., Riznychenko, O., Latorre, R., Lashevych, P. (2011). Performance of the Westinghouse WWER-1000 fuel design. *International conference on WWER fuel performance, modelling and experimental support.* Helena Resort, 245–256.
4. Kaichao, S. (2008). *MCNP modeling of hexagon VVER fuel.* Stockholm: Royal Institute of Technology, 75.
5. Dyversyfikatsiia postachannia yadernoho palyva na ukrainski AES. Tsentr Razumkova (2009). *Natsionalna bezpeka i oborona, 6,* 38–49.
6. Ivanov, V. V. (2002). Reactor core monitoring in terms of mixed fuel loading. *Symposium during International Youth Nuclear festival Dysnai-2002, Technical section.* Visaginas, 65–89.
7. Shevchenko, Y. A., Vorobev, Yu. Yu. (2015). Proverka kryteryev bezopasnosty smeshannikh zahruzok yadernoho toplyva dlia reaktorov typa VVER-1000. *Yaderna ta radiatsiina bezpeka, 2 (66),* 3–7.
8. Dziubenko, B. V., Ashmantas, L.-V., Segal, M. D. (1994). *Modelirovanie stacionarnykh i perekhodnykh teplogidravlicheskikh processov v kanalakh slozhnoi formy.* Vilnius: Pradai, 240.
9. Grebennikov, A. N., Deulin, A. A., Manoshina, I. O. et. al. (2013). Adaptaciia, verifikaciia i ispolzovanie paketa programm LOGOS dlia resheniia zadach atomnoi energetiki. *Obespechenie bezopasnosti AES s VVER.* Podolsk: FGUP OKB «GIDROPRESS», 23–32.
10. Kirillov, P. L., Iurev, Iu. S., Bobkov, V. P. (1990). *Spravochnik po teplogidravlicheskim raschetam.* Moscow: Energoatomizdat, 360.
11. Bykov, M. A., Lisenkov, E. A., Bezrukov, Iu. A. et. al. (2009). Modelirovanie processov peremeshivaniia teplonositelia v reaktore kodami TRAP-KS, DKM i KORSAR/GP. *Obespechenie bezopasnosti AES s VVER.* Podolsk: FGUP OKB «GIDROPRESS», 45–60.
12. *Predvaritelnii otchet po obosnovaniiu bezopasnosti ispolzovaniia uprochnennoi konstrukcii TVS kompanii «Vestinkhauz» na energobloke No. 3 IUUAES* (2004). Kharkiv: NNC KHFTI.CPAZ, 548.
13. Mazurenko, A. S., Skalozubov, V. I., Chulkin, O. A., Pirkovskiy, D. S., Kozlov, I. L. (2017). Determining the Conditions for the Hydraulic Impacts Emergence at Hydraulic Systems. *Problems of the Regional Energetics, 2 (34),* 98–104.
14. Skalozubov, V., Chulkin, O., Pirkovskiy, D., Kozlov, I., Komarov, Yu. (2019). Method for determination of water hammer conditions and consequences in pressurizers of nuclear reactors. *Turkish Journal of Physics, 43,* 229–235. doi: http://doi.org/10.3906/fiz-1809-5

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