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AN AUTOMATED SYSTEM FOR CONTROL OF VVER-1000 FUEL PROPERTIES AT THE EXPENSE OF REACTOR POWER OPTIMIZATION

Described is the use of WWER-1000 type reactor fuel rod claddings' resource estimating method, taking into account the creeping as the main process of shell damage accumulation; this method being applied when creating an automated nuclear fuel properties control system to optimize the reactor loading regime. Revealed is the necessity of shell damage change simulation for the correct calculation of fuel rod envelopes' stresses and deformations evolution, taking into account the complete history of reactor loading. Elaborated is the scheme of automated system for the nuclear fuel properties control by optimizing the reactor loading mode, with a detailed consideration of the suggested control system elements.

Key words*: automated control system, nuclear fuel properties, WWER-1000 reactor, power optimization, CTEV method.*

Introduction. According to forecasts of electricity production in Ukraine, in this century coming decades, the share of electricity generated at nuclear power plants will reach about 50% of the total power generation, the WWER type reactors will remain the basic equipment of nuclear energy industry. Since there takes place a transition to reactors specific with much more severe fuel elements operating conditions as compared to WWER-1000 type ones, and of high likeliness is that all or some of the reactors in Ukraine will be put into maneuver operation, that imposes the need to increase the reactor plants (RP) safety with simultaneous increase in their economical parameters. However, increasing the operational efficiency of the reactor plant due to the increased nuclear fuel burn up depth leads to a more intensive fuel elements load and an increased risk of their shells depressurization, which leads not only to a decrease in reliability and safety of reactor operation, but also to a decrease in profitability [2].

The development of an automated soft-and hardware system for fuel rods properties control of at the stage of WWER-1000 reactor equipped power plant operation, taking into account the safety and fuel efficiency balance of, will expand the boundaries and increase the operational efficiency of the reactor plant, while improving its safety.

Materials and methods used. The following methods were used in this work: CTEV method for calculating the damage to the fuel cladding; simulation of changes in the fuel cladding damage, depending on the reactor operating parameters; computer modeling; methods of accounting the fuel rod cladding damage parameter in the criterion of efficiency of fuel elements properties management; methods of controlling the fuel elements properties due to the reactor power optimization.

The aim. This work aim is to develop the theoretical foundations for the WWER-1000 reactor nuclear fuel properties control automated system by optimizing the reactor power, while operating the reactor fuel elements, taking into account damage to the shells accumulated under normal conditions, and to increase the fuel rods economic efficiency by controlling their properties, safety requirements following.

CTEV method of calculating the fuel cladding damage. To assess the shell integrity under normal operating conditions, including the variable loads regime, necessary is to take into account its loading entire history, since the development of the stress-strain state in the shell strongly depends on the sequence of loading parameters. When using the creep theory energy variant (CTEV) to account for creep as the main mechanism for fuel cladding depressurizing under normal operating conditions, the EVTP criterion for the enveloping shell depressurization [1]:

$$\omega(\tau) = A(\tau) \; / \; A_0 = 1; \;\; A(\tau) = \int_0^{\tau} \sigma_e \cdot \dot{p}_e \cdot d\tau \; , \qquad (1),$$

where $\omega(\tau)$ – is the shell damage parameter;

$A(\tau)$, A_0 – is the specific energy of scattering at time τ and when the shell is destroyed (J/m^3), respectively;

$\sigma_e(\tau)$, $\dot{p}_e(\tau)$ – are equivalent stress (Pa) and the rate of equivalent creep deformation (c^{-1}), respectively.

The A_0 limiting component magnitude in the criterion (1) does not depend on the sequence of the fuel element exploitation factors sets and is determined only by the shell material properties.

Modeling changes in the fuel cladding damage. According to the CTEV method model for fuel element properties changing, the development of fuel element deformations is determined using finite element method (FEM), and the $\omega(\tau)$ accumulation is found based on EVTP, which ensures:

– consideration to the effect of gap δ changing between the shell and fuel pellet, and the effect of oxide layer thickness on the shell outer surface, for the shell thermo mechanical condition;

– consideration of creep as the main physical accumulation process under WWER operation normal conditions;

– the bounding component in the seal leakage criterion independence from the shell operating conditions, that eliminates the problem of inconsistency between the shell leakage criterion limiting components conditions obtained and the shell actual operating conditions;

– taking into account the fuel element operating conditions sets actual sequence effect onto $\omega(\tau)$.

The model for calculating the fuel element energy release distribution on the basis of a two-group diffusion approximation, uses the following initial data: fuel element structural parameters: WWER-1000 mode parameters; reactor N power variation program characteristics; coordinates of the control system location. Output data are the values of $< q_{l,\,i,\,j} >$ the average linear power in the conditionally separated axial FA layers (i) located in Safety elements/AKZ cells (j).

The model for calculating the distributions of temperature, stress, and deformation in a fuel element according to the FEM uses as input data: $< q_{l,\,i,\,j} >$ values; fuel rod design parameters; WWER-1000 mode parameters; characteristics of program for changing N. The output data are: temperature, stress and strain in the fuel rod axial segments (AS) (one AS length being assumed equal to the length of two axial layers).

The $\omega(\tau)$ computation model uses voltages and deformations in fuel cladding AS as input data, and the output data are $A(\tau)$ and $\omega(\tau)$ values in the shell AS. At the time step $\Delta\tau_{n+1}$ the shell creep equivalent deformation p_e is defined as

$$p_{e,n+1} = p_{e,n} + \Delta\tau_{n+1} \cdot \dot{p}_e(\tau) \, . \qquad (2)$$

To calculate $p_e(\tau)$ and $\sigma_e(\tau)$ iterations between the calculation of temperature and temperature-dependent values, we perform calculation of the gap between the fuel pellet and the envelope δ to achieve convergence at the end of each time step.

Computer modeling. When choosing the Femaxi software [3], the computational analysis of the development of WWER-1000 reactor fuel element cladding stresses and deformations took into account such important advantages of this software tool as:

– concurrent solution of the heat conduction and mechanical deformation equations;

– ability to determine the fuel rod reaction to the combined effect of linear power changes in the fuel element, heat carrier parameters, other factors over the entire range of normal operating conditions of WWER-1000 reactor, up to fuel burn up depths exceeding 50 MW • day / kg-U.

According to this tool-implemented design model, the fuel element is conditionally divided into 10 AS, for center point of each AS its maximum linear power $q_{l,\max}$ AS is set. The linear power at other AS points is calculated by extrapolating the values specified for the central points. The temperature distribution in the fuel element is predicted for a one-dimensional radial geometry based on internal heat dissipation, changes in the pellet and the gap thermal conductivity, heat exchange between the fuel element surface and the coolant. The fuel temperature calculation is performed with a difference between numerical and analytical solutions never exceeding 0,1%. Both elastic and plastic deformation, mechanical interaction between fuel and envelope (MIFE), as well as creep deformation are calculated. The gaseous fission products release, their diffusion is calculated according to the adopted model with the calculus of formed bubbles reaching the grain boundaries and the fuel element internal space, also calculated is the growth of fuel element internal pressure. The time step is given automatically from the condition of the numerical solution stability and convergence in calculating creep for all AS [3].

According to the ETCP method, the $\omega(\tau)$ calculation requires to find $A(\tau)$ for the conditions of a shell long-term variable temperature-force loading under radiation conditions. Therefore, the decisive factor to choose the Femaxi tool was that this software allows correctly calculate with FEM for a WWER-1000 reactor mode parameters given sequence, the shell's σ_e and $\dot{p}_e$ taking into account the radiation effects. The rate of equivalent creep deformation was presented in the following form [3]:

$$\dot{p}_e = f(\sigma_e, \bar{\varepsilon}^H, T, \Phi, F), \qquad (3)$$

where $\bar{\varepsilon}^H$ – is the dimensionless hardening parameter; T is the temperature, K; Φ – fast neutron flux density, 1 / m^2 s; F – is the division rate, 1 / m^3 s.

Accounting for shells damage in the criterion for the fuel elements properties control effectiveness.

For the application of the EVTP method, taking into account simultaneously the fuel rod operation safety and economy limiting conditions, the criterial model (CM) of fuel element properties management efficiency is used on the following principles basis [1]:

– The purpose of WWER-1000 fuel elements properties controlling is to increase the fuel elements normal operation efficiency due to joint account of the fuel element claddings ω (τ) damage and the reactor operational efficiency economic and technological indicators;

– the fuel rod properties control is carried out on the basis of the requirements to fuel elements and core properties (ACZ), the controlled parameters determination and determinative factors;

– the control effectiveness criterion structure is the same for all control tasks, but the criterion components are not invariant;

– determined are the controlled parameters, such as:

1) Parameter of fuel rod cladding damage as a factor describing the nuclear fuel (nuclear fuel) operation safety;

2) Nuclear fuel burn up depth as a factor that its operation economy.

As a variable factor determining the fuel rod cladding damage parameter and the nuclear fuel burn up depth the nuclear reactor loading regime is used.

Results. In fig. 1 shown is a scheme for nuclear fuel properties control by optimizing the nuclear power plant loading regime. Let's consider in more detail the circuit elements controlling properties of nuclear fuel reactor type VVER-1000 due to reactor power optimization.

– The active core zone of a nuclear reactor (ACZ) is an area where nuclear fuel assemblies (FA) are located, where a controlled chain reaction of heavy nucleus division takes place. During the chain reaction in ACZ the energy evolution process takes place. The active zone includes: nuclear fuel; retarder; heat carrier, which passes outside the reactor heat generated at ACZ; Reactor Control and Protection System (RCPS). Externally, the active area is surrounded by a neutron reflector. The reflector partially returns to the ACZ neutrons that have flied out of the active zone, therefore the reflector increases the efficiency of nuclear reactor fuel use.

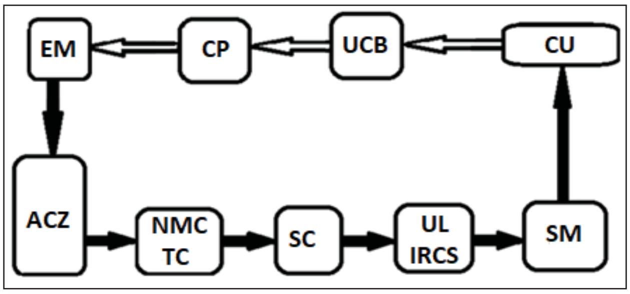

Fig. 1. Circuit controlling WWER-1000 reactor type nuclear fuel properties due to optimization of reactor power: ACZ – active zone; NMC – neutron measuring channel; TC – thermocouple; SC – switch cabinet, lower level; UL IRCS – upper level of the intra-reactor control system; SM – simulation model; CU – comparison unit; UCB -unit control board; CP – control panel; EM – executive mechanism.

– Neutron measuring channel (NMC) is designed to measure neutron power and reactivity of the reactor based on the signals received from neutron flux sensors located in the reactor plant. The NMC structure includes: neutron flux sensors for neutrons registration in the reactor and their transformation into electrical signals; signaling sensors rated sets; units for signals conversion into digital code; software and technical modules that perform information processing and calculations to get output signals corresponding to the reactor neutron power measured values, period of power change and reactivity.

– Thermocouple (TP) is a sensitive element of a thermoelectric converter in formed by two isolated conductors from heterogeneous materials connected at one end, the principle is based on the thermoelectric effect use for temperature measuring. In the proposed system, used are thermocouples of TXA type (chromel – alumilum), which temperature range is from -50 to 1300ºC.

– Switch cabinets (SC) do serve to convert physical quantities (obtained from the above sensors) into an electrical signal further fed to the internal reactor control system for analysis. The SC do place: 288 pieces of resistors to receive discrete signals of 220 V voltage and information output voltage 24 V;

terminal blocks (72 pieces each containing 16 terminals); two fan blocks.

– Upper level intra-reactor control system (UL IRCS) provides information on the reactor active zone parameters and characteristics necessary to ensure the ACZ design technological operation. Its main task is to restore the power output field in the ACZ volume to ensure the nuclear fuel safe operation. The role of IRCS in the developed system is to ensure reactor's safe and economical operation in the energy range by collecting, processing and displaying on the operator's monitors information on the ACZ state and the parameters of the first contour.

– Simulation model (MI) is implemented with a software package designed to simulate changes in the nuclear fuel reactor properties over time. The generated simulation model is designed to simulate the nuclear fuel operation at given parameters and to calculate the FA shells damage parameter taking into account their full history of loading. The simulation model includes two program environments (PE):

1) Software "Femaxi" – used to calculate the shells damage parameter.

2) Software "Reactor Simulator" (RS) – designed to create a behavior model for the reactor with given input data [4]. To note is that when using this regulation system, all necessary input data for FEMAXI are read by sensors directly from the reactor core. Therefore, the RS PE will only perform an auxiliary function.

– Comparison unit (CU), designed to compare the shells damage parameter value with its maximum allowable value (set before the reactor introduction to operation to establish limit, involving deactivation when excessive operation mode of reactor can cause an emergency situation) obtained from the simulation model. The conclusion on the need to change the load regime is made according to these values comparison result. Also this unit, as a result of solving the optimization problem, is used to calculate a new level of reactor power.

– Unit control board (UCB), located in a special room and intended for centralized automated control of technological processes, implemented by operational personnel and automation.

General information about the active zone status in the form of parameters and control recommendations is presented on UCB monitors. It serves for reactor launching, bringing to rated power, turbine launching, to synchronize the generator with power system, unit control in the normal and emergency modes of operation, as well as the planned and emergency stop of the reactor and turbine;

– The control panel (PC) making part of the UCB is intended to control and monitor the change in reactor power, respective devices operation and respective processes run. The control panel consists of a information collection unit, a control unit, a signal unit, etc.

– The executive mechanism (EM) in the proposed control system is embodied with a solenoid (electromagnetic) valve. The level of its opening depends on the results of comparing the shells damage parameter values at FA and the calculation of reactor power new level that are carried out in the comparison unit CU. Degree of valve opening affects the amount of boron concentrate added to the nutritional water of the ACZ. Thus, the control of reactor's reactivity by the method of changing the boric acid concentration in the first circuit coolant is chosen. This is reasoned because the level boric acid of concentration in the coolant significantly affects the reactivity due to the corresponding reactivity coefficient [5]. Concentration of boric acid in the coolant is varied either by the coolant addition to the reactor circuit with boric acid solution introduced, either by adding pure water to the reactor circuit. Although this method of reactor power control is rather slow compared with the method of power regulation by the IRC, however, at such system for nuclear fuel properties control, there is no need for a rapid change in the reactor power level.

Conclusions. It is advisable to develop and implement for the NPP Energoatom NAEC the WWER-1000 reactors nuclear fuel properties automated control system which optimizes the reactor power level. In this automated control system, it is necessary to use a criterion for fuel elements' properties control effectiveness, including damage to fuel rod shells under normal operating conditions.

Such an innovative nuclear fuel properties control system can be created on the basis of the method for fuel cladding shells operating life estimating under WWER-1000 reactor normal operating conditions taking into account creep as the main process of deformation damage accumulation to shells. In a promising nuclear fuel properties automated control system operated through optimizing the reactor power level, in order to correctly model the fuel cladding envelopes stresses and deformations development necessary is to take into account the change in shell damage depending on the full history of the nuclear fuel loading, determined by the reactor parameters change throughout the fuel campaign.

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АВТОМАТИЗОВАНА СИСТЕМА КЕРУВАННЯ ВЛАСТИВОСТЯМИ ЯДЕРНОГО ПАЛИВА ВВЕР-1000 ЗА РАХУНОК ОПТИМІЗАЦІЇ ПОТУЖНОСТІ РЕАКТОРА

Описано використання методу оцінки ресурсу оболонок твелів реактора типу ВВЕР-1000, що враховує повзучість як основний процес накопичення пошкодженості оболонок, у створенні автоматизованої системи керування властивостями ядерного палива за рахунок оптимізації режиму навантаження реактора. Для коректного розрахунку еволюції напруг і деформацій в оболонках твелів показана необхідність моделювання зміни пошкодження оболонок з урахуванням повної історії навантаження реактора. Розроблено схему автоматизованої системи керування властивостями ядерного палива за рахунок оптимізації режиму навантаження реактора. Детально розглянуті елементи запропонованої системи керування.

***Ключові слова:** автоматизована система керування, властивості ядерного палива, реактор ВВЕР-1000, оптимізація потужності, ЕВТП метод.*

АВТОМАТИЗИРОВАННАЯ СИСТЕМА УПРАВЛЕНИЯ СВОЙСТВАМИ ЯДЕРНОГО ТОПЛИВА ВВЭР-1000 ЗА СЧЕТ ОПТИМИЗАЦИИ МОЩНОСТИ РЕАКТОРА

Описано использование метода оценки ресурса оболочек твэлов реактора типа ВВЭР-1000, учитывающий ползучесть как основной процесс накопления поврежденности оболочек, при создании автоматизированной системы управления свойствами ядерного топлива за счет оптимизации режима нагрузки реактора. Для корректного расчета эволюции напряжений и деформаций в оболочках твэлов показана необходимость моделирования изменения повреждения оболочек с учетом полной истории нагрузки реактора. Разработана схема автоматизированной системы управления свойствами ядерного топлива за счет оптимизации режима нагрузки реактора. Подробно рассмотрены элементы предложенной системы управления.

***Ключевые слова:** автоматизированная система управления, свойства ядерного топлива, реактор ВВЭР-1000, оптимизация мощности, ЭВТП метод.*