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MODERNIZED LOSS-OF-COOLANT & BLACKOUT ACCIDENT MANAGEMENT STRATEGY AT NUCLEAR POWER PLANTS WITH WWER

В.І. Скалозубов, В.М. Спінов, Д.В. Спінов, Т.В. Габлая, В.Ю. Кочнєва, Ю.О. Комаров. **Модернізована стратегія управління аваріями з міжконтурними течами та повним тривалим знеструмленням ядерних енергоустановок з ВВЕР.** Проведений аналіз відомих результатів розрахункового моделювання кодом RELAP5/V.3.2 аварій з повним тривалим знеструмленням та течами другого контуру ядерних енергоустановок з ВВЕР показав, що проектні стратегії управління такими аваріями проектними пасивними системами безпеки не забезпечують необхідні умови безпеки щодо максимально допустимої температури оболонок твелів, щодо мінімально допустимого рівня теплоносія в реакторі і живильної води в парогенераторах. Представлено модернізовану стратегію управління аварією з повним знеструмленням та течами другого контуру на основі перспективних систем пасивного відводу тепла і підтримки необхідного рівня теплоносія в реакторі і живильної води в парогенераторах. Для обґрунтування модернізованої стратегії управління аваріями з повним знеструмленням та течами другого контуру розроблена консервативна теплогідродинамічна модель. Основні консервативні припущення моделі: приймається повна тривала відмова (на 72 години) усіх електронасосів систем безпеки і моделюється максимальна міжконтурна теча (еквівалентна відриву кришки колектора парогенератора). Аналіз результатів розрахункових обґрунтувань показав, що модернізована стратегія управління аваріями з повним тривалим знеструмленням та міжконтурними течами або розривами паропроводів/трубопроводів основної живильної води забезпечує необхідні умови безпеки щодо максимально допустимої температури оболонок твелів, щодо мінімально допустимого рівня теплоносія і живильної води. Представлені результати розрахункового моделювання стратегій управління аваріями з повним тривалим знеструмленням ядерних енергоустановок можуть бути використані для модернізації і вдосконалення симптомно-орієнтрованих аварійних інструкцій та посібників з управління важкими аваріями на ядерних енергоустановках із реакторами типа ВВЕР. Застосування отриманих результатів розрахункового моделювання стратегій управління аваріями з повним тривалим знеструмленням у загальному випадку не обґрунтовано для інших типів реакторної установки. У цьому випадку необхідна розробка розрахункових моделей управління аваріями з повним тривалим знеструмленням, що враховують специфіку конструкційно-технічних характеристик та умов експлуатації систем, важливих для безпеки ядерних енергоустановок.

Ключові слова: управління аваріями, теча реакторного контуру, знеструмлення

V. Skalozubov, V. Spinov, D. Spinov, T. Gablaya, V. Kochnyeva, Yu. Komarov. **Modernized Loss of Coolant & Blackout Accident Management Strategy at Nuclear Power Plants with WWER.** The analysis of the known results of RELAP5/V.3.2 simulation for loss of coolant & blackout accidents at WWER nuclear power plants showed that the design accident management strategies with design passive safety systems do not provide the necessary safety conditions for the maximum permissible temperature of fuel claddings, the minimum permissible level of coolant in the reactor and feed water in the steam generators. This work presents the modernized loss of coolant & blackout accident management strategy based on promising heat removal passive systems, reactor level control systems and steam generator feed water level control systems. A conservative thermohydrodynamic model was developed to substantiate the modernized loss of coolant & blackout accident management strategy. The main conservative assumptions of the model: a complete long-term failure (for 72 hours) of all electric pumps of the safety systems is accepted and the maximum interloop leak (equivalent to the steam generator collector cover lift-up) is modelled. The analysis of the calculation results showed that the modernized loss of coolant & blackout accident management strategy provides the necessary safety conditions for the maximum allowable temperature of the fuel claddings, for the minimum acceptable level of coolant and feed water. The presented results of computational modelling of blackout accident management strategies for nuclear power plants can be used to modernize and improve symptom-informed emergency instructions and guidelines for the severe accident management at nuclear power plants with WWER. Application of the results of computational modelling of blackout accident management strategies is generally not substantiated for other types of reactor facilities. In this case, it is necessary to develop calculated models for blackout accident management taking into account the specifics of the structural and technical characteristics and operating conditions for safety related systems of nuclear power plants.

Keywords: accident management, loss of coolant, blackout

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Introduction

Analysing known results of calculation modelling of secondary loss of coolant accidents (LOCA2) at WWER NPF with RELAP5/V.3.2 Code we can conclude the followings.

1. For initial accident events (IAE) with total loss of feedwater of the steam generator (LOCA21), safety conditions for the maximum admissible temperature of fuel claddings (1200 °C) are violated in $18.0\cdot10^3$ s of accident process, the minimum level of feedwater in the steam generator (1.35 m) is reached in $1.5\cdot10^3$ s, and full drainage of the steam generator is reached in $14.0\cdot10^3$ s of accident process. Operability of systems of the emergency and auxiliary feedwater pumps (EFWP/AFWP) is critical for safety during LOCA21.

2. Possible uncontrolled primary cooldown (recriticality) stopping after emptying of the damaged steam generator (780 s of process) is characteristic for IAE with steam line break in not isolated part between the steam generator (SG) and the main steam isolation valve (MSIV) outside containment (LOCA22). Besides, fast emptying of SG because of LOCA22 leads to increase in thermal stress in heat-exchange tubes and to their possible destruction. EFWP or AFWP also is critical for safety during LOCA22.

3. Irretrievable primary coolant loss is characteristic for interloop loss IAE (LOCA23). Calculation modelling of an interloop loss with the accepted maximum equivalent size of 100 mm (equivalent to SG collector cover lift-up) and jamming of quick-acting pressure reducing plant of air discharge to the atmosphere has revealed that safety conditions for the maximum admissible temperature of fuel claddings are violated in $17.8\cdot10^3$ s of accident process, and feedwater level in SG increases to 3.2 m in $1.5\cdot10^3$ s of accident process.

Electric pumps of an emergency core cooling system of the reactor (ECCS) and EFWP/AFWP are critical for safety conditions during LOCA23.

For blackout accidents with failure of all electric pumps of safety systems at NPF with WWER, it is recognized that full drainage of SG is in $6.0\cdot10^3$ s, and safety conditions for the maximum admissible temperature of fuel claddings are violated in $19.0\cdot10^3$ s of accident process.

One of lessons of great accident at the Fukushima-Daiichi NPP in 2011 is need to model and analyse accidents with joint IAEs and multiple failures of safety and control systems. As regards LOCA2, joint LOCA23 and blackout accidents are the most conservative as to the critical systems providing safety conditions. Known results of calculation modelling of IAEs of LOCA23 and blackout accident at WWER NPF with the RELAP5/V.3.2 Code showed that design accident management strategies (DAMS) do not provide required safety conditions. It defines relevance of substantiation of the modernized accident management strategies (MAMS) to manage accidents with IAE "LOCA2 + blackout" by the promising passive safety systems without long power supply.

Analysis of recent publications and problem statement

The work [1] presents results of loss-of-coolant accident modelling with ATHLET Code for nuclear power plants with WWER-1000. However, this work does not consider blackout accidents.

The work [2] presents summarized results of loss-of-coolant accident modelling. However, this work does not consider accident modelling for simultaneous initiation of loss of coolant and blackout.

The work [3] analyses accidents at NPPs with WWER. However, this work does not consider problems of simultaneous initiation of interloop loss of coolant and blackout.

The IAEA Report [4] presents the task program for investigating and analysing of NPP accidents. This report also does not address accidents in case of simultaneous initiation of loss of coolant and blackout.

The IAEA Final Report [5] summarizing results of NPP accident modelling also does not consider IAE of simultaneous initiation of loss of coolant and blackout.

The work [6] studies the maximum design basis accident at NPP with WWER-1000 to substantiate nuclear fuel diversification. This work does not consider loss-of-coolant & blackout accident modelling.

The work [7] analyses accidents of short-term power loss. This work does not address blackout accident modelling.

The work [8] analyses blackout accidents at NPP with WWER-1000, but does not taking into account possibility of simultaneous initiation of interloop loss of coolant.

The work [9] presents the accident management strategies at NPP with WWER. However, this work does not consider blackout accident management strategies.

The work [10] proposes methods for modelling the conditions for water hammers during accidents in a pressurizer. However, these methods are not substantiated for blackout accident conditions.

The work [11] analyses the conditions for water hammers in active safety systems with electric pumps. However, the results of this work are not substantiated for passive safety systems in the blackout accident conditions.

The work [12] determines the conditions for water hammers in the transonic two-phase flows in the NPP armature. However, the results of this work do not cover the conditions for water hammers in other heat engineering equipment during blackout accidents.

The work [13] proposes an original approach to blackout accident management using a steam-driven emergency feed pump. However, the conditions for the effectiveness of such a pump for blackout accident management are not defined.

Purpose and objectives of the study

The purpose of the work is to substantiate the accident management strategy for combined initial events of blackout and loss of coolant at nuclear power plants with WWER.

The work tasks.

1. To develop the thermohydraulic accident model for combined initial events of blackout and loss of coolant (IAE “LOCA23 + blackout”).

2. To analyse results of calculated modelling and to substantiate the “LOCA23 + blackout” accident management strategy.

Conservative thermohydrodynamic accident model for IAE “LOCA23 + blackout”. The key diagram of system of promising passive safety systems for management of accident with IAE “LOCA2 + blackout” is presented in Fig. 1 and Fig. 2.

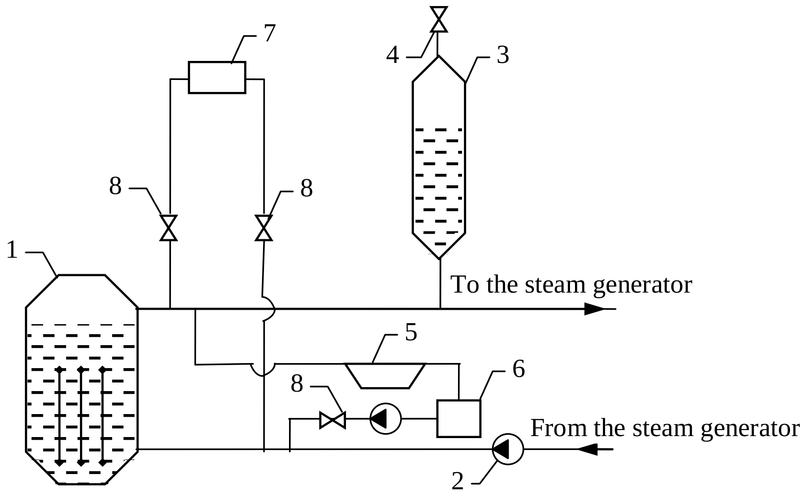

Fig. 1. ARPS R: 1 – reactor, 2 – main circulating pump (MCP), 3 – pressurizer, 4 – pressurizer safety valves, 5 – ARPS R1, 6 – hydraulic reservoirs of ECCS, 7 – ARPS R2, 8 – armature

The structure of system:

a) The afterheat removal passive system of the reactor (ARPS R) consisting of two independent subsystems:

ARPS R1 – a subsystem of the reactor steam-driven emergency pump (SDEP);

ARPS R2 – a subsystem of a closed circuit of natural circulation;

b) The afterheat removal passive system of SG (ARPS SG) consisting of two subsystems:

ARPS SG1 – a subsystem of the SG steam-driven auxiliary feedwater pump (SDAFP);

ARPS SG2 – a subsystem of a closed circuit of natural circulation.

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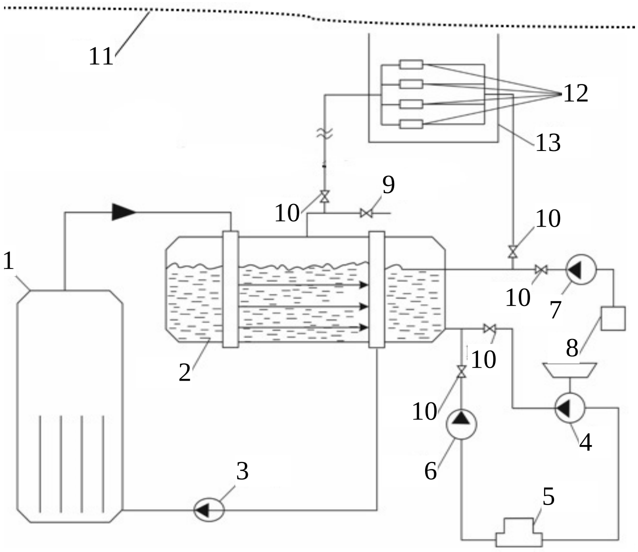

Fig. 2. Standard scheme of ARPS SG of WWER: 1 – reactor, 2 – SG, 3 – MCP, 4 – turbine feed pump, 5 – deaerator, 6 – AFWP, 7 – EFWP, 8 – EFWP water storage tank, 9 – MSIV, 10 – stop valves, 11 – containment, 12 – ARPS SG condensers, 13 – water storage tank (for water cooling of ARPS SG condensers)

Main conservative assumptions:

The maximum interloop loss conservative to steam line breaks is modelled;

Complete long-term failure (not less than 72 h) of all electric pumps of safety systems is accepted;

Temperature of nuclear fuel T_F in the central part of a fuel matrix is accepted as the maximum admissible for nuclear safety conditions (2800 °C);

Effect of “run down” flow of a turbine feed pump and the coolant level in pressurizer on accident process is not considered.

The balance equations of masses and heat energy for the reactor volume V_R free of internals:

$$\frac{d(\rho_{VR}V_{VR})}{dt} = G_{TV} - G_{V1R} - G_{2R}, \quad V_R = V_{VR} + V_T, \tag{1}$$

$$\rho_T \frac{dV_T}{dt} = G_K + G_{GP} + G_{1R} + G_{2R} + G_{GE} - G_{LOC}, \tag{2}$$

$$\frac{d(\rho_{VR}V_{VR}i_{VR})}{dt} = G_{TV}r_V - (G_{1R} + G_{2R})i_{VR}, \tag{3}$$

$$\rho_T \frac{d(V_T i_T)}{dt} = (G_K + G_{GP} - G_{LOC})i_T + G_{1R}i_{1R} + G_{2R}i_{2R} - G_{TV}r_V + G_{GE}i_{GE}. \tag{4}$$

The heat balance equation for ARPS R2:

$$G_{2R}r_C = \alpha_0 F_{2R}(T_{VR} - T_{0S}) - C_P G_{2R}(T_{CS} - T_{2R}). \tag{5}$$

The balance equations of masses and heat energy for SG volume V_G "free" of internals:

$$\frac{d(\rho_{VG}V_{VG})}{dt} = G_{LV} - G_{V1G} - G_{2G}; \qquad V_G = V_{VG} + V_L, \tag{6}$$

$$\rho_L \frac{dV_L}{dt} = G_{1G} + G_{2G} - G_{LV} + G_{LOC}, \tag{7}$$

$$\frac{d(\rho_{VG}V_{VG}i_{VG})}{dt} = G_{LV}r_V - (G_{V1G} + G_{2G})i_{VG}, \tag{8}$$

$$\rho_L \frac{d(V_L i_L)}{dt} = G_{1G} i_{1G} + G_{2G} i_{2G} - G_{LV} r_V . \quad (9)$$

The heat balance equation for ARPS SG2:

$$G_{2G} r_C = \alpha_0 F_{2G} (T_{VG} - T_{0S}) - C_P G_{2G} (T_{CS} - T_{2G}) . \quad (10)$$

The heat balance equation between the reactor and SG:

$$N_T(t) = F_1 R_T^{-1} (T_F - T_0) + F_2 R_{VG}^{-1} (T_0 - T_L) + G_{LOC} i_T . \quad (11)$$

Mass flow rates in subsystems in quasistationary approximation:

$$G_K = \Pi_T \sqrt{\rho_T (\rho_{VR} - \rho_T) g h_{1K} / \xi_{1K}} , \quad (12)$$

$$G_{V1R} = \mu_V \Pi_{VR} \sqrt{2\rho_{VR} (P_R - P_{SA})} , \quad (13)$$

$$G_{2R} = \Pi_{2R} \sqrt{\rho_T (\rho_T - \rho_{VG}) g h_{2R} / \xi_{2R}} , \quad (14)$$

$$G_{2G} = \Pi_{2G} \sqrt{\rho_L (\rho_L - \rho_{VG}) g h_{2G} / \xi_{2G}} , \quad (15)$$

$$G_{GE} = \begin{cases} \mu_{GE} \Pi_{GE} \sqrt{2\rho_T (P_{GE} - P_R)},\ P_{GE} > P_R,\ t < t_{GE}, \\ 0,\ P_{GE} \le P_R,\ t \ge t_{GE}, \end{cases} \quad (16)$$

$$G_{V1G} = \mu_V \Pi_{VG} \sqrt{2\rho_{VG} (P_G - P_D)} , \quad (17)$$

$$G_{1R} = \begin{cases} G_{SR},\ P_R > 0.3\ \text{MPa}, \\ 0,\ P_R \le 0.3\ \text{MPa}, \end{cases} \quad (18)$$

$$G_{1G} = \begin{cases} G_{AG},\ P_G > 0.3\ \text{MPa}, \\ 0,\ P_G \le 0.3\ \text{MPa}. \end{cases} \quad (19)$$

Coolant mass break flow rate:

$$G_{LOC} = \begin{cases} \mu_{LOC} \Pi_{LOC} \sqrt{2\rho_T (P_R - P_G)},\ G_{LOC} < G_{CR}, \\ G_{CR},\ G_{LOC} \ge G_{CR}. \end{cases} \quad (20)$$

Arrangement of ARPS R2 and ARPS SG2 in NPF containment dimensions these subsystems. Limit operability of steam-driven pumps is defined from A.V. Korolev's experimental data [13] and operating experience of a turbine feed pump.

After transformation of the equations (1) – (11) taking into account:

$$\frac{d\rho_V}{dt} = \frac{d\rho_V}{dP} \frac{dP}{dt}, \quad \frac{di_V}{dt} = \frac{di_V}{dP} \frac{dP}{dt},$$

we get system of the nonlinear equations for key parameters of safety conditions:

$$\frac{dP_{VR}}{dt} = f_1(P_{VR}, V_T, P_{VG}, V_L, i_T, i_L, T_0, N, G_{LOC}) , \quad (21)$$

$$\frac{dV_T}{dt} = f_2(P_{VR}, V_T, P_{VG}, V_L, i_T, i_L, T_0, N, G_{LOC}) , \quad (22)$$

$$\frac{dP_{VG}}{dt} = f_3(P_{VR}, V_T, P_{VG}, V_L, i_T, i_L, T_0, N, G_{LOC}) , \quad (23)$$

$$\frac{dV_L}{dt} = f_4(P_{VR}, V_T, P_{VG}, V_L, i_T, i_L, T_0, N, G_{LOC}) , \quad (24)$$

$$\frac{di_T}{dt} = f_5(P_{VR}, V_T, P_{VG}, V_L, i_T, i_L, T_0, N, G_{LOC}) , \quad (25)$$

$$\frac{di_L}{dt} = f_6(P_{VR}, V_T, P_{VG}, V_L, i_T, i_L, T_0, N, G_{LOC}) , \quad (26)$$

$$T_0(t) = f_7(P_{VR}, V_T, P_{VG}, V_L, i_T, i_L, T_0, N, G_{LOC}) . \quad (27)$$

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Initial conditions:

$$P_{\mathrm{VR}}(t=0)=P_{\mathrm{VR0}},\ V_{\mathrm{T}}(t=0)=V_{\mathrm{T0}},\ P_{\mathrm{VG}}(t=0)=P_{\mathrm{VG0}},\ V_{\mathrm{L}}(t=0)=V_{\mathrm{L0}},$$
$$i_{\mathrm{T}}(t=0)=i_{\mathrm{T0}},\ i_{\mathrm{L}}(t=0)=i_{\mathrm{L0}},\ T_0(t=0)=T_{00}\,. \quad (28)$$

Combined equations (21) – (28) was integrated by a numerical Runge-Kutta method.

In the equations (1) – (28): t is accident time, ρ_{VR}, ρ_{VG}, ρ_{T}, ρ_{L} is density of steam in the reactor and SG, of the coolant and feedwater, respectively, V_{VR}, V_{VG}, V_{T}, V_{L} is volume of steam in the reactor and SG, of the coolant in the reactor and feedwater in SG, respectively, i_{VR}, i_{VG}, i_{T}, i_{L}, i_{1R}, i_{2R}, i_{GE} is a specific enthalpy of steam in the reactor and SG, of the coolant and feedwater of SG, of the coolant in EFWP tanks (ARPS R1) and feedwater at the outlet of ARPS R2, of the coolant in ECCS hydraulic reservoirs, of feedwater at the inlet of ARPS SG1 and at the outlet of ARPS SG2, respectively, G_{TV}, G_{LV}, G_{V1R}, G_{V1G}, G_{1R}, G_{1G}, G_{2R}, G_{2G}, G_{K}, G_{GE}, G_{GP}, G_{SR}, G_{AG}, G_{CR} is a mass flow rate of steam generation in the reactor and SG, for steam drive of SDEP and SDAFP, from ARPS R1 and ARPS SG1, from ARPS R2 and ARPS SG2, of primary natural circulation, from ECCS hydraulic reservoirs, of MCP "run down", a rated capacity of electric pumps of ECCS and EFWP, a critical flow under the transonic flow conditions, respectively, r_{V}, r_{C} is heat of steam generation and condensation, respectively, T_{VR}, T_{VG}, T_{CS}, T_{2R}, T_{2G}, T_{F}, T_0, T_{0S} is temperature of steam in the reactor and SG, of steam condensation, of condensate at the outlet of ARPS R2 and ARPS SG2, of nuclear fuel in the central part of a fuel matrix and fuel cladding, of environment in containment, respectively, P_{R}, P_{G}, P_{SA}, P_{GE}, P_{D} is pressure in the reactor, SG, ECCS hydraulic reservoirs and the deaerator, respectively, α_0 is heat transfer coefficient on an outer heat transfer surface of ARPS R2 and ARPS SG2, F_{2R}, F_{2G} is the area of an outer heat transfer surface of ARPS R2 and ARPS SG2, respectively, C_{P} is the specific heat capacity of water, Π_{T}, Π_{VR}, Π_{VG}, Π_{GE}, Π_{2R}, Π_{2G}, Π_{LOC} is the throat area of primary coolant, a steam drive of SDEP and SDAFP, ECCS hydraulic reservoirs, ARPS R2 and ARPS SG2, interloop loss, respectively, F_1, F_2 is the total heat transfer area in the reactor and SG, respectively, g *is* acceleration due to gravity, h_{1K}, h_{2R}, h_{2G} is primary, ARPS R2 and ARPS SG2 pressure head, respectively, ξ_{1K}, ξ_{2R}, ξ_{2G} is primary, ARPS R2 and ARPS SG2 total drag coefficient, respectively, μ_{V}, μ_{GE}, μ_{LOC} is a flow coefficient in steam drives, ECCS hydraulic reservoirs and in a leak, respectively, t_{GE} is time for emptying ECCS hydraulic reservoirs, T_1 is feedwater temperature in SG.

Thermal resistance of fuel element and interloop volume:

$$R_{\mathrm{T}}=\delta_{\mathrm{F}}/\lambda_{\mathrm{F}}+\delta_{\mathrm{GA}}/\lambda_{\mathrm{GA}}+\delta_0/\lambda_0\,,$$
$$R_{\mathrm{VG}}=1/\alpha_1+\delta_{\mathrm{T0}}/\lambda_{\mathrm{T0}}+1/\alpha_2\,,$$

where δ_{F}, δ_{GA}, δ_0, δ_{T0} is thickness of a fuel matrix, gas gap, a fuel cladding and SG heat exchange pipes, respectively, λ_{F}, λ_{GA}, λ_0, λ_{T0} is a thermal conductivity of fuel, a gas gap, a fuel cladding and SG heat exchange pipes, respectively, α_1, α_2 is heat transfer coefficient to the coolant in the reactor and on a surface of SG heat exchange pipes, respectively.

Analysis of results of calculated modelling. The main results of calculation modelling of accident with interloop loss of D_u = 100 mm (equivalent to SG collector cover lift-up) and blackout ("LOCA23 + blackout") using the modernized accident management strategy for ARPS R and ARPS SG are given in Fig. 3 in dimensionless parameters:

$$\mathbf{T}_0=T_0/T_{\mathrm{lim}},\quad \mathbf{V}_{\mathrm{T}}=V_{\mathrm{T}}/V_{\mathrm{T0}},\quad \mathbf{V}_{\mathrm{L}}=V_{\mathrm{L}}/V_{\mathrm{G}},\quad \mathbf{P}_G=P_G/P_{\mathrm{lim}}\,, \quad (29)$$

where T_{lim} is the maximum admissible temperature of fuel cladding, P_{lim} is the maximum allowable pressure in SG (actuation of secondary steam relief valves).

At the initial moment of accident, the maximum temperature of fuel claddings increases at $\mathbf{T}_0<1$, and coolant level in the reactor decreases without drainage of a nuclear core owing to pressure decrease and the start of steam generation in the reactor, and also an interloop leak (see Fig. 3, *a*). After 100 s of accident, coolant level in the reactor increases, and temperature of fuel claddings decreases owing to operability the ARPS R and ARPS SG. Insignificant decrease in the coolant level at the final stage of accident is because of shutdown the ARPS R1 and ARPS SG1.

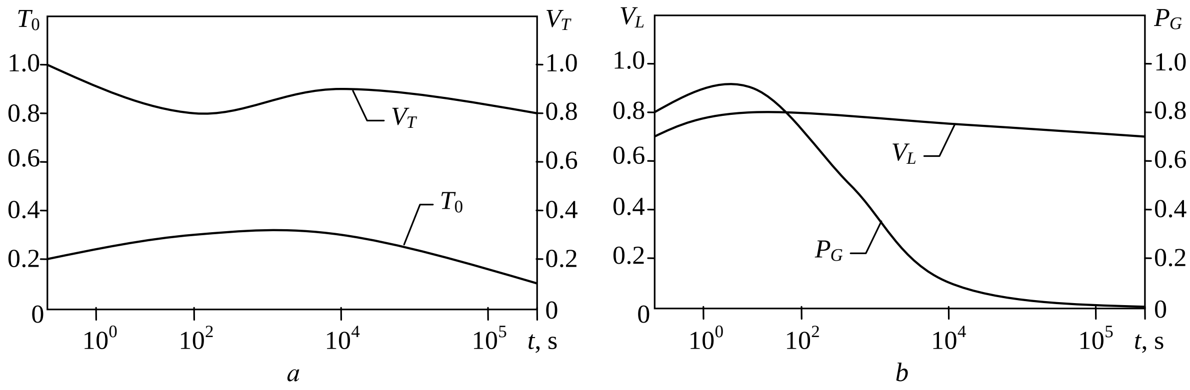

Fig. 3. MAMS with IAE "LOCA23 + blackout": maximum temperature of fuel claddings T_0 and the coolant level in the reactor V_T (a); feedwater level V_L and pressure P_G in SG (b)

At the initial moments of accident, feedwater level and pressure in SG increase (without actuation of steam relief valves and water hammers on the SG vessel). After 100 s of accident, feedwater level in SG is stabilized, and pressure in SG decreases owing to operability of SG ARPS.

Safety conditions for the maximum admissible temperature of fuel claddings and minimum admissible levels of the coolant and feedwater are provided for not less than 72 hours.

Conclusions

1. The analysis of the known results of RELAP5/V.3.2 simulation for loss of coolant & blackout accidents at WWER nuclear power plants showed that the design accident management strategies with design passive safety systems do not provide the necessary safety conditions for the maximum permissible temperature of fuel claddings, the minimum permissible level of coolant in the reactor and feedwater in the steam generators.

2. This work presents the modernized loss of coolant & blackout accident management strategy based on promising heat removal passive systems, reactor level control systems and steam generator feed water level control systems.

3. A conservative thermohydrodynamic model was developed to substantiate the modernized loss of coolant & blackout accident management strategy. The main conservative assumptions of the model: a complete long-term failure (for 72 hours) of all electric pumps of the safety systems is accepted and the maximum interloop leak (equivalent to the steam generator collector cover lift-up) is modelled.

4. The analysis of the calculation results showed that the modernized loss of coolant & blackout accident management strategy provides the necessary safety conditions for the maximum allowable temperature of the fuel claddings, for the minimum acceptable level of coolant and feedwater.

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