ISSN 2076-2429 (print)
ISSN 2223-3814 (online)

UDC 629.031

V. Skalozubov[1]**,** DSc, Prof.,
V. Spinov[1]**,** PhD,
D. Spinov[2]**,**
T. Gablaya[1]**,**
V. Kochnyeva[1]**,**
Yu. Komarov[1]**,** DSc
[1] Odessa National Polytechnic University, 1 Shevchenko Ave., Odessa, Ukraine, 65044; e-mail: gntcod@te.net.ua
[2] Intersectoral Center for Basic Research in the field of energy and ecology of NAS of Ukraine by ONPU, 1 Shevchenko Ave., Odessa, Ukraine, 65044

SUBSTANTIATION OF STRATEGIES FOR THE MANAGEMENT OF BLACKOUT ACCIDENT AT NUCLEAR POWER PLANTS WITH WWER-TYPE REACTORS

В.І. Скалозубов, В.М. Спінов, Д.В. Спінов, Т.В. Габлая, В.Ю. Кочнєва, Ю.О. Комаров. **Обґрунтування стратегій управління аваріями з повним тривалим знеструмленням на ядерних енергоустановках з реакторами типа ВВЕР.** Розроблено консервативну теплогідродинамічну модель аварії на ядерній енергоустановці з реакторами типа ВВЕР для проектної та модернізованої стратегії управління аваріями з повним тривалим знеструмленням. Проектна стратегія управління аварією з повним тривалим знеструмленням заснована на пасивних системах безпеки (які не потребують енергопостачання): система компенсації тиску та рівня теплоносія в реакторі, а також система пароскидальних пристроїв другого контуру. Модернізована стратегія управління аварією з повним тривалим знеструмленням заснована на додатковому застосуванні перспективних пасивних систем безпеки: система аварійного живильного насоса з пароприводом від парогенератора і система пасивного відводу тепла природною циркуляцією через парогенератор. Проведено розрахункове моделювання для двох стратегій управління аваріями з повним тривалим знеструмленням: проектної, що здійснюється пасивними системами безпеки компенсатора тиску і пароскидальних пристроїв другого контуру, та модернізованої, що здійснюється пасивними системами безпеки компенсатора тиску, пароскидальних пристроїв, аварійного живильного насоса з пароприводом від парогенератора і пасивного відводу тепла від активної зони реактора через парогенератор. У результаті розрахункового аналізу на основі розробленої консервативної теплогідродинамічної моделі встановлено, що проектна стратегія управління аваріями з повним тривалим знеструмленням не забезпечує успішного виконання функцій безпеки щодо відведення залишкових тепловиділень від активної зони реактора і щодо підтримки необхідного рівня живильної води в парогенераторах. Умови безпеки щодо максимальної температури оболонок тепловиділяючих елементів і максимально припустимого рівня живильної води в парогенераторі порушені при проектній стратегії управління аварією з повним тривалим знеструмленням. Модернізована стратегія управління аваріями з повним тривалим знеструмленням забезпечує успішне виконання функцій і умов безпеки. Результати моделювання аварії з повним тривалим знеструмленням, представлені в цій роботі, можуть бути використані для вдосконалення стратегій управління аваріями в експлуатаційній документації з управлінню аваріями та симптомно-орієнтованих аварійних інструкціях для ядерних енергоустановок з реакторами типу ВВЕР.
Ключові слова: стратегія управління аварією, ядерна енергоустановка, повне тривале знеструмлення

V. Skalozubov, V. Spinov, D. Spinov, T. Gablaya, V. Kochnyeva, Yu. Komarov. **Substantiation of strategies for the management of blackout accident at Nuclear Power Plants with WWER-type reactors.** A conservative thermohydrodynamic model of an accident at a nuclear power plant with WWER has been developed for a design and modernized blackout accident management strategy. The design blackout accident management strategy is based on passive safety systems (that do not require power supply): a pressurizer system, a reactor level control system, and a secondary steam relief system. The modernized blackout accident management strategy is based on the additional use of promising passive safety systems: a steam generator driven auxiliary feed pump and an afterheat removal passive system using natural circulation through a steam generator. Two strategies of blackout accident management are modelled: the first one is design blackout accident management strategy implemented by passive safety systems of pressurizer and the secondary steam relief system, and the second is modernized blackout accident management strategy that implemented by passive safety systems of pressurizer, by steam relief system, steam generator driven auxiliary feed pump and afterheat removal passive system through a steam generator. Based on the developed conservative thermohydrodynamic model, a calculation analysis has found that the design blackout accident management strategy does not ensure the successful safety functions to remove residual heat from the reactor core and to maintain the required level of feed water in steam generators. The design blackout accident management strategy violates safety conditions for the maximum temperature of the fuel claddings and the maximum level of feed water in the steam generator. A modernized blackout accident management strategy ensures the successful safety functions and conditions. The results of blackout accident modelling presented in this work can be used to improve accident management strategies in accident management operational documentation and symptom-informed accident regulations for nuclear power plants with WWERs.
Keywords: accident management strategy, nuclear power plant, blackout

DOI: 10.15276/opu.3.59.2019.06

Introduction. The initial accident event of blackout of nuclear power facilities (NPF) at the Fukushima-Daiichi NPP because of site's flooding by a tsunami became one of basic causes of the severe nuclear fuel damages and destructive steam-gas explosions [1]. One of the main lessons of the Fukushima accident is to develop and implement effective strategies for blackout accident management at NPF.

The modernized blackout accident management strategy (MAMS) for NPF with WWER can be based on additional use of the power-free passive safety systems: a steam generator driven auxiliary feed pump and an afterheat removal passive system through a steam generator.

The analysis of the design and modernized blackout accident management strategies determines relevance of the presented work.

Analysis of recent publications and problem statement. At present, there are no publications about blackout accident modelling at NPFs with WWER.

Reports of the operating organization of NNEGC "Energoatom" have analysed blackout accidents at NPF with WWER-1000 using RELAP 5/M3.2 Code. The results are the followings.

1. In the design mode, blackout accidents proceed without violating nuclear safety conditions if auxiliary power supply is restored quickly [2]. However, this work does not consider blackout accident modelling.

2. To provide nuclear safety conditions, operation of one system of emergency feed water pumps (EFWP) is enough [3]. However, this work does not consider blackout accident modelling.

3. In case of the design blackout accident management strategy (DAMS) with passive safety systems, nuclear safety conditions for the most admissible temperature of fuel claddings are violated [4]. However, this work does not consider modernization of blackout accident management strategies.

Purpose and objectives of the study. The purpose of the work is to substantiate the blackout accident management strategies at nuclear power plants with WWERs.

This purpose requires solving the following tasks:

1) Analysis of literature data on blackout accident modelling;

2) Development of a conservative thermohydrodynamic model of a blackout accident;

3) Analysis of the results and development of practical recommendations.

Conservative thermohydrodynamic model of blackout accident. The scenario of blackout accident:

Shutdowns of reactor, main coolant pump (MCP) and feed pumps of the steam generator (SG);

Closer of the main steam isolation valve (MSIV);

Blackout and failure of diesel generators;

Failure to restore power supply;

Failure of all electric pumps of safety systems;

Accident management by passive safety systems.

Design passive safety systems:

Pressurizer system in a reactor loop;

Secondary steam relief system (SRS).

Advanced passive safety systems:

Steam generator driven auxiliary feed pump (SGAFP);

Afterheat removal passive system through a steam generator (SG ARPS).

The rated scheme of systems at blackout accident is given in Fig. 1.

Main conservative assumptions:

To neglect the effect of a "run down" flow of a turbine feed pump on heat exchange conditions;

To consider nuclear fuel temperature in the central part of a fuel matrix as maximum.

The balance equations of masses and heat energy for the steam and coolant volumes in the reactor:

$$\frac{\mathrm{d}(\rho_{\mathrm{V}} V_{\mathrm{VR}})}{\mathrm{d}t} = G_{\mathrm{VR}}, \quad V_{\mathrm{R}} = V_{\mathrm{VR}} + V_{\mathrm{T}}, \tag{1}$$

$$\rho_{\mathrm{T}} \frac{\mathrm{d}V_{\mathrm{T}}}{\mathrm{d}t} = G_{\mathrm{GC}}(t) + G_{\mathrm{K}}(t) - G_{1}(t) - G_{\mathrm{VR}}, \tag{2}$$

 ISSN 2076-2429 (print)
ISSN 2223-3814 (online)

$$\frac{\mathrm{d}(\rho_V i_V V_{VR})}{\mathrm{d}t} = G_{VR} r_V\,, \tag{3}$$

$$\rho_T \frac{\mathrm{d}V_T i_T}{\mathrm{d}t} = F_0 R_{f0}^{-1}(T_f - T_0) + \alpha_T F_0 R_{f0}(T_0 - T_T) - F_{VG} R_{VG}^{-1}(T_T - T_L)\,. \tag{4}$$

The heat balance equation for the reactor and the secondary SG volume:

$$N_T(t) = F_0 R_{f0}^{-1}(T_f - T_0) + \alpha_T F_0 R_{f0}(T_0 - T_T) - F_{VG} R_{VG}^{-1}(T_T - T_L)\,. \tag{5}$$

The coolant equation in a reactor loop of length L_1 and throat area Π_1:

$$\frac{\mathrm{d}}{\mathrm{d}t}(G_1 + G_{GC} + G_K) = \frac{\Pi_1}{L_1}\left[P_{VR} - \frac{\xi_1}{\rho_T \Pi_1^2}(G_1 + G_{GC} + G_K)^2 + (\rho_T - \rho_V) g h_1\right], \tag{6}$$

where $G_{GC}(t)$, $G_1(t)$, $G_K(t)$ is a mass flow rate of MCP “run down”, at the entrance to a reactor loop and between the pressurizer and a reactor loop, respectively, ξ_1 is total coefficient of hydraulic resistance of a reactor loop, h_1 is height of a reactor loop.

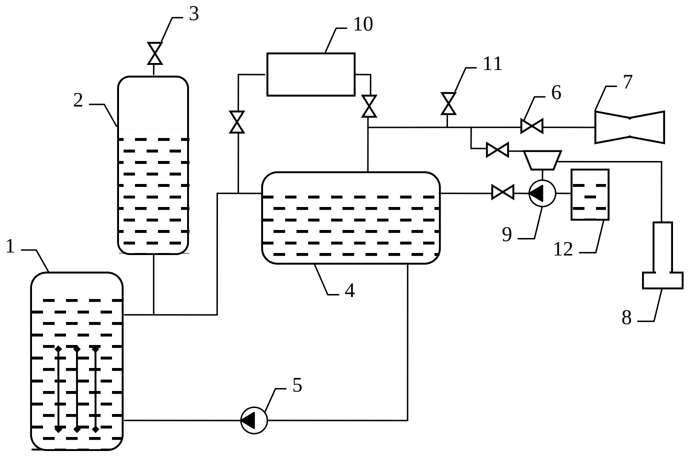

Fig. 1. The rated scheme of model of blackout accident: 1 – reactor, 2 – pressurizer, 3 – safety valves, 4 – SG, 5 – MCP, 6 – MSIV, 7 – turbine, 8 – deaerator, 9 – SGAFP, 10 – SG ARPS, 11 – SRS, 12 – feedwater tank

Time dependence $G_{GC}(t)$ can be determined from numerical approximation of results of calculated modelling [4, 5].

Mass flow rate between pressurizer and a reactor loop:

$$G_K(t) = \begin{cases} \mu_K \Pi_{TT} \sqrt{2\rho_T \Delta P_{KT}} & \text{if} \quad \Delta P_{KT} = \Delta P_{VK} + g\rho_T(H_K + L) - P_{VR}; \\ -\mu_K \sqrt{2\rho_T \left|\Delta P_{KT}\right|} & \text{if} \quad \Delta P_{KT} < 0. \end{cases} \tag{7}$$

Mass and energy balance equations for pressurizer volumes:

$$(V_K - \rho_T \Pi_K H_K)\frac{\mathrm{d}\rho_V}{\mathrm{d}P_{VK}}\frac{\mathrm{d}P_{VK}}{\mathrm{d}t} - \rho_V \Pi_K \frac{\mathrm{d}H_K}{\mathrm{d}t} = -G_{iK}\,, \qquad \rho_T \Pi_K \frac{\mathrm{d}H_K}{\mathrm{d}t} = -G_K\,, \tag{8}$$

$$\frac{\mathrm{d}\left[\rho_V (V_K - \rho_T \Pi_K H_K) i_V\right]}{\mathrm{d}t} = -G_{iK} i_V\,, \tag{9}$$

where t is time of accident progress, ρ_V, ρ_T is density of steam and the coolant, respectively, V_{VR}, V_T is volume of steam and the coolant in the reactor, respectively, G_{VR}, G_K is flow rate of a steam generation in the reactor core ($G_{VR} = 0$ $T_T < T_{TS}$) and the coolant between pressurizer and a reactor loop, respectively, i_V, i_T is a specific enthalpy of steam and the coolant, respectively, r_V is the latent heat of steam generation, T_f, T_0, T_L is the maximum temperature of nuclear fuel, a fuel cladding and feedwater in SG, respectively, P_{VK}, P_{VR} is steam pressure in pressurizer and the reactor, respectively, F_0, F_{VG} is the total area of heat exchange of fuel elements and SG surface submerged in feedwater, respectively, α_T is heat transfer coefficient on a surface of fuel element [6], H_K is the coolant level in pressurizer, L is height of the connecting pipeline between pressurizer and a reactor loop, μ_K is a flow coefficient between pressurizer and a reactor loop, Π_{TT}, Π_K is a throat area of the coolant between pressurizer and a reactor loop and in pressurizer volume, respectively, V_K is the pressurizer volume, "free" of structures, $N(t)$ is the power of the reactor afterheat, g is gravity acceleration.

Mass flow rate via pilot-operated safety valves (POSV) of pressurizer:

$$G_{iK} = \begin{cases} \mu_{iK}\Pi_{iK}\sqrt{2\rho_V(P_{VK} - P_{g0})} \text{ if } P_{VK} \geq P_{max}; \\ 0 \text{ if } P_{VK} < P_{max}, \end{cases} \tag{10}$$

where μ_{iK} is a flow coefficient of the pressurizer POSV, Π_{iK} is the throat area of the safety valve, P_{G0}, P_{max} is pressure in containment and the maximum allowable pressure in pressurizer (criterion of opening of the pressurizer POSV), respectively.

Coefficients of thermal resistance of fuel element R_{f0} and interloop volume R_{VG}:

$$R_{f0} = \delta_f / \lambda_f + \delta_g / \lambda_g + \delta_0 / \lambda_0, \tag{11}$$

$$R_{VG} = \delta_{VG} / \lambda_{VG} + 1 / \lambda_{VG}, \tag{12}$$

where δ_f, δ_g, δ_0, δ_{VG} is thickness of a fuel matrix, a gas gap, a fuel cladding and heat-exchanging tubes of SG, respectively, λ_f, λ_g, λ_0, λ_{Vg} is a thermal conductivity of a fuel matrix, a gas gap, a fuel cladding and heat-exchanging tubes of SG, respectively, α_{VG} is heat transfer coefficient on an outer surface of heat-exchanging tubes of SG [7], V_R is the reactor volume, "free" of structures.

After transformation of the equations (1) – (9) we will receive:

$$\frac{dP_{VR}}{dt} = f_1(P_{VR}, V_T, P_{VK}, H_K, G_1), \tag{13}$$

$$\frac{dV_T}{dt} = f_2(P_{VR}, V_T, P_{VK}, H_K, G_1), \tag{14}$$

$$\frac{dP_{VK}}{dt} = f_3(P_{VR}, V_T, P_{VK}, H_K, G_1), \tag{15}$$

$$\frac{dH_K}{dt} = f_4(P_{VR}, V_T, P_{VK}, H_K, G_1), \tag{16}$$

$$\frac{dG_1}{dt} = f_5(P_{VR}, V_T, P_{VK}, H_K, G_1). \tag{17}$$

Initial conditions:

$$P_{VR}(t=0) = P_{R0},\ V_T(t=0) = V_{T0},\ H_K(t=0) = H_{K0},\ G_1(t=0) = G_0,\ P_{VK}(t=0) = P_{R0} + \rho_T g(H_{K0} + L), \tag{18}$$

where G_0 is a quasistationary flow rate of the coolant at a rated power of the reactor N_0.

The equation of current temperature of fuel cladding in accident progress:

$$T_0(t=0) = \frac{N(t) + (\alpha_T F_0 - R_{VG}^{-1} F_{VG})T_T + R_{VG}^{-1} T_L(t)}{F_0(\alpha_T - R_{f0}^{-1})}, \tag{19}$$

$$T_0(t=0) = T_{00}, \qquad N(t=0) = N_0. \tag{20}$$

Power of afterheat $N(t)$ was determined from known experimental approximation [6] for the WWER nuclear fuel.

The mass and heat energy balance equations for SG volume:

 ISSN 2076-2429 (print)
ISSN 2223-3814 (online)

$$\frac{\mathrm{d}\rho_V V_{VG}}{\mathrm{d}t} = G_{LV} - G_{VA} - G_{VC} = \frac{\mathrm{d}\rho_V}{\mathrm{d}P_{VG}} \frac{\mathrm{d}P_{VG}}{\mathrm{dt}} + \rho_V \frac{\mathrm{d}V_{VG}}{\mathrm{dt}}, \quad (21)$$

$$\rho_L \frac{\mathrm{d}V_{LG}}{\mathrm{d}t} = G_A + G_{VC} - G_{LV}, \quad V_G = V_{VG} + V_{LG}, \quad (22)$$

$$\rho_L \frac{\mathrm{d}V_{LG} i_L}{\mathrm{d}t} = F_{VG} R_{VG}^{-1} (T_0 - T_L) + G_A i_{LA} - G_{LV} r_V + G_{VC} i_{LC}, \quad (23)$$

$$\frac{\mathrm{d}(\rho_V V_{VG} i_V)}{\mathrm{d}t} = G_{LV} r_V - (G_{VA} + G_{VC}) i_L. \quad (24)$$

Motion equations in a steam drive of SGAFP and SG ARPS, respectively:

$$G_{VA} = \mu_V \Pi_{VA} \sqrt{2\rho_V (P_{VG} - P_0)}, \quad (25)$$

$$G_{VC} = \Pi_C \sqrt{(\rho_L - \rho_V) \rho h_C / \xi_C}. \quad (26)$$

The equation of a heat balance for a heat-exchange surface of SG ARPS:

$$G_{VC} [r_C + C_P (T_{LS} - T_{LC})] = \alpha_C F_C (T_{LS} - T_{G0}), \quad (27)$$

where ρ_V, ρ_L is steam density and feedwater of SG, respectively, G_{LV}, G_{VA}, G_{VC} is a steam mass flow rate in SG volume, in a steam drive of SGAFP and in SG ARPS, respectively, P_{VG} is steam pressure in SG, V_G, V_{VG}, V_{LG} is the total volume of SG, "free" of structures, of steam and feedwater, respectively, i_L, i_{LA}, i_{LC} is a specific enthalpy of feedwater in SG volume, in hydraulic reservoirs of EFWP and at the exit from SG ARPS, respectively, G_A =35kg/s is the rated flow of SGAFP feedwater corresponding to a rated flow of EFWP, μ_V is steam flow coefficient in a steam drive, Π_{VA}, Π_C is a throat area of a steam drive of SGAFP and SG ARPS, respectively, ξ_C is total coefficient of hydraulic resistance of SG ARPS, C_P is the specific heat capacity of condensate, r_C is the latent heat of condensation, T_{LS}, T_{LC}, T_{G0} is temperature of saturation at condensation, condensate at the exit from the SG ARPS and environment in containment, respectively, α_C is heat transfer coefficient on a heat-exchange surface of SG ARPS.

Thermohydraulic and constructional and technical parameters of the systems/equipment were defined according to [2 – 5, 7, 8].

After transformations, combined equations (21) – (27):

$$\frac{\mathrm{d}P_{VG}}{\mathrm{dt}} = f_6 (P_{VG}, H_{LG}, i_L), \quad (28)$$

$$\frac{\mathrm{d}H_{LG}}{\mathrm{dt}} = f_7 (P_{VG}, H_{LG}, i_L), \quad (29)$$

$$\frac{\mathrm{di}_L}{\mathrm{dt}} = f_8 (P_{VG}, H_{LG}, i_L). \quad (30)$$

At initial conditions:

$$P_{VG}(t=0) = P_{VG0}, \quad H_{LG} = V_{LG} / F_{LG}(t=0) = H_{LG0}, \quad i_{LG}(t=0) = i_{LG0}, \quad (31)$$

where G_{LG} is the throat area of feedwater in SG.

Analysis of results of calculated modelling. Conditions for successful safety functions of HR SF and SG SF:

$$T_0 < T_{lim}, \quad H_{LG} > H_{min}, \quad (32)$$

where T_{lim} is the maximum admissible temperature of fuel cladding, H_{min} is minimum admissible level of feedwater in SG. For WWER T_{lim}=1474 K, H_{min} =1.35m [4].

For blackout accidents the key parameters to fulfil safety functions:

Pressure in the reactor P_{VR} and SG P_{VG}, Level of coolant in the reactor H_T and feedwater H_{LG}.

Solutions of combined equations of conservative thermohydrodynamic model (13) – (18), (28) – (31) are received with a known Runge-Kutta method. Results of calculated modelling of the key parameters and conditions for HR SF and SG SF are given in Fig. 2 – 4.

Two strategies of blackout accident management are modelled:

DAMS is carried out by passive safety systems of pressurizer and the secondary SRS,

MAMS is carried out by passive safety systems of pressurizer, SRS, SGAFP and ARPS.

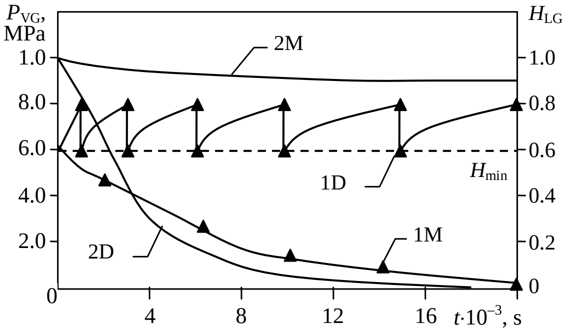

Fig. 3. Pressure P_{VR} and feedwater level H_{LG}=H_{LG}/H_{LG0} in SG: 1D – P_{VR}/DAMS, 1M – P_{VR}/MAMS, 2D – H_{LG}/DAMS; 1M – H_{LG}/MAMS

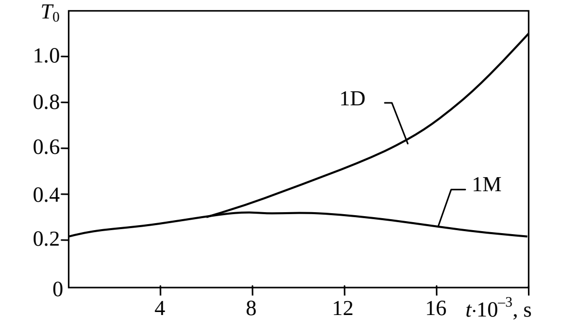

Fig. 4. Maximum temperature of fuel claddings T_0=T_0/T_{lim} in the reactor: 1D – DAMS, 1M – MAMS

Table 1 analyses results of calculated modelling of the key parameters and conditions for HR SF and SG SF for DAMS and MAMS in different times of the accident.

Table 1

Results of modelling of the key parameters of safety functions

Time	Accident management strategy	Behaviour of the key parameters of safety functions	Availability of safety function
0 – $2\cdot10^3$ s	DAMS	Pressure in the reactor decreases because of MCP shutdown. Coolant level in the reactor is keep due to a "run down" flow of MCP and decrease in level in pressurizer. Steam pressure in SG increases up to actuation conditions for SRS. Feedwater level in SG significantly decreases because of an intensification of steam generation up to maximum values	SG SF is not available
	MAMS	Pressure in the reactor decreases; coolant level in the reactor is keep. Steam pressure and feedwater level in SG is keep due to SG feed by SG ARPS and SGAFP without SRS actuation. Feedwater level in SG decreases by 10% from initial one and it is more than maximum permissible values	HR SF and SG SF are available
Beyond $2\cdot10^3$ s	DAMS	Pressure in the reactor and SG increases up to the maximum admissible values (periodic actuation of the pressurizer POSVs and SRS valves). Coolant level in the reactor is zero in $14.0\cdot10^3$ s of the accident; and feedwater level in SG – in $17.0\cdot10^3$ s. Temperature of fuel claddings sharply increases after $7.0\cdot10^3$ s and reaches maximum permissible values (1473 K) in $19.5\cdot10^3$ s	HR SF and SG SF are not available
	MAMS	Pressure in the reactor increases up to "hot shutdown" conditions without actuation of the pressurizer POSVs. Coolant level in the reactor decreases by 30% from initial one before $10.0\cdot10^3$s of the accident and then it is constant. Pressure in SG falls down to 0.3 MPa in $20.0\cdot10^3$s of the accident. Feedwater level is constant (10% lower from initial one). Maximum temperature of fuel claddings increases up to 360°C in $6.5\cdot10^3$s and then it fall down to its initial value. Causes of violation of nuclear safety conditions of are absent before 72[nd]h of the accident	HR SF and SG SF are available until 72[nd]h of the accident

ISSN 2076-2429 (print)
ISSN 2223-3814 (online)

The analysis of results of calculated modelling based on conservative thermohydrodynamic model showed that DAMS does not ensure successful safety functions of HR SF and SG SF and safety conditions (32). MAMS provides successful safety functions of HR SF and SG SF and safety conditions (32).

Conclusions

1. A conservative thermohydrodynamic model of an accident at a nuclear power plant with WWER-type reactors has been developed for a design and modernized blackout accident management strategy. The design blackout accident management strategy is based on passive safety systems (that do not require power supply): a pressurizer system, a reactor level control system, and a secondary steam relief system. The modernized blackout accident management strategy is based on the additional use of promising passive safety systems: a steam generator driven auxiliary feed pump and an afterheat removal passive system using natural circulation through a steam generator.

2. Based on the developed conservative thermohydrodynamic model, a calculation analysis has found that the design blackout accident management strategy does not ensure the successful safety functions to remove residual heat from the reactor core and to maintain the required level of feed water in steam generators. The design blackout accident management strategy violates safety conditions for the maximum temperature of the fuel claddings and the maximum level of feed water in the steam generator.

A modernized blackout accident management strategy ensures the successful safety functions and conditions.

Література

1. IAEA International Fact Expert Mission of the Fukushima-Daiichi NPP Accident Following the Great East Japan Earthquake and Tsunami. IAEA Mission Report, 2011. 160 p.
2. Проект углубленного анализа безопасности энергоблока № 5 ЗАЭС. Заключительный отчет по анализу критериев успеха систем. Расчетное обоснование № 100 52DL 12R. Энергодар : ГП НАЭК «Энергоатом», 2000. 536 с.
3. EP25-2004.210.JL.2. Потеря электроснабжения собственных нужд. Корректировка и обновление ВАБ энергоблока № 5 ЗАЭС. Расчетное обоснование критериев успеха. Приложение G1.1. Энергодар : ГП НАЭК «Энергоатом», 2004. 365 с.
4. EP01/2016.100.ОД.1-Т.1. Расчет теплогидравлических параметров для всех режимов эксплуатации оборудования РУ энергоблока № 3 ОП ЗАЭС. Энергодар : ГП НАЭК «Энергоатом», 2016. 566 с.
5. EP25-2004.210.ОД.2. Корректировка и обновление ВАБ энергоблока № 5 ЗАЭС. Обесточивание энергоблока с отказом дизель-генераторов. Приложение G2.1. Энергодар : ГП НАЭК «Энергоатом», 2004. 365 с.
6. Хизанцян А.М., Оганесян Л.С. Остаточные энерговыделения при аварийных остановах реакторов типа ВВЭР. *Известия НАН РА и ГИУА LX-1*. 2007. С. 100–106.
7. Кириллов П.Л., Юрьев Ю.С., Бобков В.П. Справочник по теплогидравлическим расчетам ядерных энергоустановок. Москва : Энергоатомиздат, 1990. 360 с.
8. Skalozubov V.I., Kozlov I.L., Komarov Yu.A., Chulkin O.A., Piontkovskyi O.I. Analysis of Nuclear Safety in Diversification of Westinghouse Fuel Assemblies at WWER-1000. *Nuclear Physics and Atomic Energy*. 2019. 20-2. P. 159–163.

References

1. *IAEA International Fact Expert Mission of the Fukushima-Daiichi NPP Accident Following The Great East Japan Earthquake and Tsunami* (2011). IAEA Mission Report, 160.
2. *Draft In-Depth Safety Analysis of Zaporizhzhya NPP-5.* (2000). Final Report on the Analysis of System Success Criteria. Calculation Justification 100 52DL 12R. Energodar: NNEGC Energoatom, 536.
3. EP25-2004.210.JL.2. *Loss of Auxilairy Power Supply. Correction and Updating of PSA of Zaporizhzhya NPP-5. Calculation Justification of System Success Criteria.* Annex G1.1 (2004). Energodar: NNEGC Energoatom, 365.
4. EP01/2016.100.ОД.1-Т.1. *Calculation of Thermohydraulic Parameters for All Operating Modes of Reactor Equipment at Zaporizhzhya NPP-3* (2016). Energodar: NNEGC Energoatom, 566.

5. EP25-2004.210.ОД.2. *Correction and Updating of PSA of Zaporizhzhya NPP-5. Power Unit Blackout Followed by a Failure of Diesel Generators.* Annex G2.1. (2004). Energodar: NNEGC Energoatom, 365.
6. Khizantsyan, A.M., & Hovhannisyan, L.S. (2007). Shutdown Heat at Emergency Hot Stand Reactors of WWER Type. *Izvestiya NAN RA i GIUA LX-1*, 100–106.
7. Kirillov, P.L., Yurev, Yu.S., & Bobkov, V.P. (1990) *Handbook of Thermal-Hydraulic Calculations*. 2nd Ed. Moskva: Energoatomizdat.
8. Skalozubov, V.I., Kozlov, I.L., Komarov, Yu.A., Chulkin, O.A., & Piontkovskyi, O.I. (2019). Analysis of Nuclear safety in Diversification of Westinghouse Fuel Assemblies at WWER 1000. *Nuclear Physics and Atomic Energy, 20-2*, 159–163.

Скалозубов Володимир Іванович; Skalozubov Volodymyr, ORCID: https://orcid.org/0000-0003-2361-223X
Спінов Владислав Михайлович; SpinovVladislav, ORCID: https://orcid.org/0000-0001-7555-847X
Спінов Дмитро Владиславович; Spinov Dmitro, ORCID: https://orcid.org/0000-0002-7888-2889
Габлая Таїсія Володимирівна; Gablaya Taisiya, ORCID: https://orcid.org/0000-0003-3184-5674
Кочнєва Валерія Юріївна; Kochneva Valeria, ORCID: https://orcid.org/0000-0001-7397-3573
Комаров Юрій Олексійович; Komarov Yuriy ORCID: https://orcid.org/0000-0002-4696-6551

Received October 12, 2019

Accepted November 26, 2019