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QUALIFICATION OF AFTERHEAT REMOVAL PASSIVE SYSTEM FROM REACTOR TO MANAGE BLACKOUT ACCIDENTS

В.И. Скалозубов, В.М. Спинов, Д.В. Спинов, Т.В. Габлая, В.Ю. Кочнева, К.В. Скалозубов. **Кваліфікація системи пасивного відводу тепла від реакторної установки для управління аваріями з повним тривалим знеструмленням.** Критеріями та умовами кваліфікації працездатності і надійності пропонованої в роботі системи пасивного відводу тепла від реактора для управління аваріями з повним тривалим знеструмленням є критерії умови ядерної безпеки по максимально допустимим температурам ядерного палива і оболонок тепловиділяючих елементів; по напору тиску і витраті теплоносія аварійним насосом з пароприводом і за габаритними обмеженням пасивної системи відводу тепла природною циркуляцією. Розроблено консервативна теплогідродинамічна модель кваліфікації системи пасивного відводу тепла від реактора для управління аваріями із повним тривалим знеструмленням. В результаті розрахункового моделювання, за запропонованою консервативною моделлю встановлено, що проектна стратегія управління аварією з повним тривалим знеструмленням не забезпечує умови ядерної безпеки. Модернізована стратегія управління аварією системою пасивного відведення тепла від реактора забезпечує умови ядерної безпеки при досить консервативних припущеннях. Відповідно до експериментальних даних О.В. Корольова, працездатність аварійного насоса з пароприводом забезпечена при тиску в реакторі більше 0,3 МПа. При менших тисках, функції безпеки по охолодженню активної зони і підтримки рівня теплоносія в реакторі, забезпечуються кваліфікованою підсистемою пасивного відведення тепла природною циркуляцією. Отримані в роботі результати можна використовувати для модернізації ядерних енергетичних установок з метою підвищення ефективності управління аваріями з повним тривалим знеструмленням, а також для вдосконалення симптомно-орієнтованих аварійних інструкцій і керівництв з управління важкими аваріями з пошкодженням ядерного палива. Пропоновану систему управління аваріями з повним тривалим знеструмленням можна доповнити пасивними системами безпеки з відводом пари через парогенератори ядерних енергоустановок з ректорами типа ВВЕР. Пропонована пасивна система є ефективною лише для аварій з повним тривалим знеструмленням та великими течами реакторного контуру (в том числі і для максимальної проектної аварії з розривом реакторного контуру). Результати, представлені в цій роботі, використовуються в учбовому процесі для підготовки, перепідготовки та підвищення кваліфікації фахівців ядерної енергетики України.
Ключові слова: кваліфікація, пасивні системи безпеки, аварії з повним знеструмленням, реакторна установка

V. Skalozubov, V. Spinov, D. Spinov, T. Gablaya, V. Kochnyeva, K. Skalozubov. **Qualification of the afterheat removal passive system from the reactor to manage blackout accidents.** Nuclear safety criteria and conditions for the maximum admissible temperatures of nuclear fuel and fuel claddings, for the pressure and coolant flow of the steam-driven emergency pump and for dimensions of using natural circulation are qualification criteria and conditions for operability and reliability of the presented afterheat removal passive system from the reactor to manage blackout accidents. The conservative heathydrodynamic model of qualification of afterheat removal passive system from the reactor is developed for blackout accident management. Calculation modelling with the presented conservative model has recognized that design blackout accident management strategy does not ensure nuclear safety conditions. The modernized accident management strategy with afterheat removal passive system from the reactor provides nuclear safety conditions. According to Prof. Korolev's experiments, operability of the steam-driven emergency pump is provided when the reactor pressure is more than 0.3 MPa. For smaller pressure, the afterheat removal passive subsystem using natural circulation provides safety functions. The results of this work can be used to modernize nuclear power plants with the view of increasing the efficiency of blackout accident management, and to improve symptom-informed instructions and guidelines for the accident management of severe fuel damages. The proposed blackout accident management system can be supplemented with steam removal passive safety systems through steam generators of nuclear power plants with WWERs. The proposed passive system is effective only for blackout accidents and large loss-of-coolant accidents in the reactor (including the maximum design depressurization accident). The presented results are used for training, retraining and advanced training of specialists in Ukrainian nuclear energy.
Keywords: qualification, passive safety system, blackout accident, reactor plant

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Introduction. Blackout accident is a basic cause of catastrophic consequences at Fukushima-Daiichi NPP. Therefore, world nuclear power faced the problem to develop and implement effective blackout accident management strategies in the reactor. Development and implementation of the promising passive safety systems that do not need long-term power supply is one of the directions of solving the above problem.

The afterheat removal passive system (ARPS R) can be a perspective passive safety system of a reactor. It consists of two subsystems: the core cooling system with steam-driven emergency pumps (ARPS R1) and afterheat removal passive system from a reactor using natural circulation (ARPS R2).

According to the current nuclear laws, development and implementation of ARPS R defines need for the corresponding qualification. In our case, the qualification is to substantiate operability and reliability of ARPS R for effective blackout accident management.

Qualification of ARPS R defines relevance of the represented work.

Analysis of publications. The work [1] presents calculation modelling of design and beyond design basis accidents at nuclear power plants with WWER. However, modelling of blackout accidents at nuclear power plants are studied insufficiently. The work [2] considers modelling of severe nuclear fuel damage. However, this work dose not consider severe accidents as a result of an initial emergency event of blackout. The work [3] analyses nuclear safety of diversification of nuclear fuel for the maximum design accident of a double-ended break of reactor coolant pipe. This work also does not consider blackout accidents.

The work [4] researches behaviour of nuclear fuel during accidents. However, this work also does not consider blackout accidents. The work [5] analyses thermomechanical behaviour of fuel claddings during loss-of-coolant accidents. This work also does not consider blackout conditions.

The work [6] considers short-time or partial power failures at nuclear power plants with WWER. Therefore, results of these calculation substantiations cannot be used for blackout accidents. As a result of the calculation modelling of blackout accidents, the work [7] has recognized that design blackout accident management strategy does not ensure nuclear safety conditions for the maximum admissible temperature of fuel claddings. Therefore, definition of more effective blackout accident management strategy is topical issue.

Complex use of the promising passive safety systems that do not need long-term power supply is effective blackout accident management strategy. ARPS R can be one of such passive safety systems. Qualification of ARPS R with alternative conservative methods of heathydrodynamic modelling of blackout accidents.

Main Objective and tasks of Work. Qualification of operability and reliability of afterheat removal passive system of the reactor (ARPS R) for blackout accident management is the main objective of work.

Main tasks of work:

– definition of criteria and conditions for ARPS R qualification to manage blackout accidents;

– development of conservative heat hydrodynamic model of system “Reactor – Steam generator – ARPS R”;

– the analysis of results of calculation modelling and the conclusion on ARPS R qualification.

Conservative Heathydrodynamic Model of System “Reactor – Steam generator – ARPS R”. The conservative heathydrodynamic model of system “Reactor – Steam generator – ARPS R” is developed for two blackout accident management strategies:

– Design blackout accident management strategy (DAMS) with pressurizer system in a reactor and secondary system of steam relief valves (SRV).

– Modernized blackout accident management strategy (MAMS) with systems of ARPS R, pressurizer, SRV.

Main conservative assumptions:

a) complete failure of all electric pumps of safety systems and lack of possibility of recovery of auxiliary power supply of the power unit is accepted;

b) effect of “run down” flow of the stopped turbine feed pump on feed conditions of the steam generator is not considered;

c) feedwater temperature in volume of the steam generator is accepted equal to the saturation boiling temperature (T_{ls});

d) temperature of nuclear fuel (T_F) of the central part of a fuel matrix of the fuel elements is accepted maximum for a rated power operation of the reactor.

The calculation scheme of system “Reactor – Steam generator – ARPS R” is given below.

Criteria and conditions for the ARPS R qualification meet to criteria and conditions for nuclear safety:

$$T_0 < T_{m0}\,;\quad T_F < T_{mF}\ , \tag{1}$$

where T_0, T_F is temperature of fuel claddings and nuclear fuel, respectively, T_{m0}, T_{mF} is the maximum admissible temperatures of fuel claddings and nuclear fuel, respectively. For WWER: T_{m0} = 1200 °C and T_{mF} = 2800 °C.

ARPS R1 qualification condition for pressure head (ΔP_A) of the steam-driven emergency pump (SDEP):

$$\Delta P_A > P_{mR}\,, \tag{2}$$

where P_{mR} is the maximum allowable pressure in the reactor.

Qualification conditions for the ARPS R2 constructional and technical parameters:

$$F = F_{\max}\,;\ h = h_{\max}\,, \tag{3}$$

where $F_{\max}$, $h_{\max}$ is maximum heat-transfer area and ARPS R2 height (for the placement in containment), respectively.

Equations of a mass and heat balance:

$$\frac{d\rho_v V_v}{dt} = G_{Tv} - G_{v1} - G_{v2}\,, \tag{4}$$

$$\rho_T\frac{dV_T}{dt} = -G_{Tv} + G_K + G_{gp} + G_A + G_{v2}\,, \tag{5}$$

$$\frac{d\rho_v V_v i_v}{dt} = G_{Tv}\cdot r_v - (G_{v1} + G_{v2})\cdot i_v\,, \tag{6}$$

$$N(t) = F_1\cdot R_T^{-1}\cdot(T_F - T_0) + q_{Rv}(T_{ls})\cdot F_2\,, \tag{7}$$

$$\rho_T\frac{dV_T\cdot i_T}{dt} = -G_{Tv}\cdot r_v + G_K\cdot i_T + G_{gp}\cdot i_T + G_A\cdot i_{TA} + G_{v2}\cdot i_{T2}\,, \tag{8}$$

$$G_{v2}\cdot r_c = \alpha_0\cdot F_{\max}(T_v - T_{0s}) + C_p\cdot G_{v2}\cdot(T_{cs} - T_2)\,. \tag{9}$$

Motion equations in systems of pressurizer, a steam-drive of SDEP and ARPS R2 in quasistationary approach:

$$G_K = \begin{cases} \mu_K F_K\sqrt{2\rho_T(P_K - P_R)}, & P_K \ge P_R; \\ -\mu_K F_K\sqrt{2\rho_T(P_K - P_R)}, & P_K < P_R, \end{cases} \tag{10}$$

at

$$G_{v1} = \mu_v F_v\sqrt{2\rho_v(P_R - P_{SA})}\,, \tag{11}$$

$$G_{v2} = \Pi\cdot\sqrt{\frac{\rho(\rho_v - \rho_T)g\cdot h_{\max}}{\xi_2}}\,, \tag{12}$$

where ρ_v, ρ_T is a steam and coolant density, respectively, V_v, V_T is the steam and coolant volume in the reactor, respectively, t is accident time, G_{Tv} is a steam generation flow in the reactor, G_{v1}, G_{v2} is steam

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flow consumption in the ARPS R1 and ARPS R2, respectively, G_K, G_{gp}, G_A is flow of coolant in pressurizer, of "run down" of the main circulating pump and SDEP, respectively, i_v, i_T is specific enthalpy of steam and the coolant in the reactor, respectively, r_v, r_c is latent heat of steam generation and condensation, respectively, $N(t)$ is the afterheat power in the reactor, F_1, F_2 is the heat-transfer area in the core and a pipe surface of the steam generator, respectively, q_{Rv} is heat flux density when boiling feedwater in volume of the steam generator, T_v, T_{cs}, T_{T2} is temperature of steam, saturated condensate and condensate at the outlet of ARPS R2, respectively, C_p is the specific heat capacity of the coolant, i_{TA}, i_{t2} is the specific enthalpy of the coolant in hydraulic reservoirs of emergency core cooling systems of the reactor (ARPS R1) and at the outlet of ARPS R2, respectively, α_0 is heat-transfer coefficient on a surface of ARPS R2, T_{0s} is environment temperature in containment, P_R is pressure in the reactor, R_T is thermal resistance of fuel element:

$$R_T = \frac{\delta_F}{\lambda_F} + \frac{\delta_g}{\lambda_g} + \frac{\delta_0}{\lambda_0},$$

where δ_F, δ_g, δ_0 is thickness of a fuel matrix, gas gap and fuel cladding, respectively, λ_F, λ_g, λ_0 is heat conductivity of nuclear fuel, gas gap and fuel cladding, respectively.

After transformation of combined equations (4) – (9) taking into account $\frac{d\rho_v}{dt} = \frac{d\rho_v}{dP_R} \cdot \frac{dP_R}{dt}$, we will receive:

$$\frac{dP_R}{dt} = f_1(P_R, V_T, T_0, T_{T2}), \tag{13}$$

$$\frac{dV_T}{dt} = f_2(P_R, V_T, T_0, T_{T2}), \tag{14}$$

$$T_{T2}(t) = T_{cs} - \frac{G_{v2} \cdot r_c - \alpha_0 \cdot F_{\max}(T_v - T_{0s})}{C_p \cdot G_{v2}}, \tag{15}$$

$$T_0(t) = T_F - R_T \cdot \frac{N(t) - q_{Rv} \cdot F_2}{F_1}. \tag{16}$$

At initial conditions:

$$P_R(t=0) = P_{R0}, \quad V_T(t=0) = V_{T0}, \quad N(t=0) = N_0. \tag{17}$$

Analysis of Qualification Results.

The combined equations (13) – (17) are decided with a known numerical method of Runge-Kutta.

The main results of calculation modelling of blackout accident are given in Table.

The main results of calculation modelling of blackout accident

№	Events	Accident time, 10^{-3} s	
		DAMS	MAMS
1	Pressure decrease in the reactor	before 1.0	before 1.0
2	Pressure increase up to the maximum admissible values (start of pilot-operated relief valves of the pressurizer)	after 2.0	–
3	Decrease in the coolant level below a core level	14.0	–
4	Fuel cladding temperature 1200 ^{0}C	20.0	–

At the initial stage of accident (before $1.0 \cdot 10^3$ s of accident), the reactor pressure decreases owing to reduction of a pressure head of the stopped main circulating pump (MCP). The coolant level is keeping with pressurizer system and a "run down" flow of the stopped MCP.

Pressure reduction in the reactor intensifies processes of steam generation in a core and the corresponding pressure increase. For DAMS, the reactor pressure reaches the maximum admissible values (start of pilot-operated relief valves of the pressurizer) in $2.0 \cdot 10^3$ s of an accident. The coolant level in the reactor decreases below a core level in $14.0 \cdot 10^3$ s.

For MAMS, ARPS R providing safety functions to cool a core and to keep the coolant level compensates pressure increase and reduction of the coolant level in the reactor.

For DAMS with blackout, temperature of a fuel cladding reaches 1200 ^{0}C in 20.0 s of an accident if the afterheat power of the reactor less than 1% of the rated power of the reactor. For MAMS with blackout, temperature of a fuel cladding does not exceed 500 ^{0}C within 72 hours of an accident.

Thus, the modernized accident management strategy with ARPS R (unlike design strategy) provides necessary nuclear safety conditions (1).

The ARPS R1 qualification conditions are provided when the coolant flow and pressure head of SDEP are corresponding to similar constructional and technical parameters of the electric high-pressure pumps of the emergency core cooling system. The reactor pressure less than 0.3 MPa is the lower bound of SDEP operability. When the reactor pressure less than 0.3 MPa, safety functions to cool a core and to keep the coolant level are provided by ARPS R2 subject to qualification condition (3).

Conclusions. Nuclear safety criteria and conditions for the maximum admissible temperatures of nuclear fuel and fuel claddings, for the pressure and coolant flow of the steam-driven emergency pump and for dimensions of afterheat removal passive system using natural circulation are qualification criteria and conditions for operability and reliability of the presented afterheat removal passive system from the reactor to manage blackout accidents.

The conservative heathydrodynamic model of qualification of afterheat removal passive system from the reactor is developed for blackout accident management. Calculation modelling with the presented conservative model has recognized that design blackout accident management strategy does not ensure nuclear safety conditions. The modernized accident management strategy with afterheat removal passive system from the reactor provides nuclear safety conditions.

According to Prof. Korolev's experiments, operability of the steam-driven emergency pump is provided when the reactor pressure is more than 0.3 MPa. For smaller pressure, the afterheat removal passive subsystem using natural circulation provides safety functions.

Література

1. Antropov V., Bukrinsky A., Shvyryaev Yu. Development of Methodology and List of BDBA for WWER-1000 for Quantitative Analysis. SAM-99 Information Exchange Forum on "Severe Accident Management", 18 – 22 October 1999, Obninsk, Russia.
2. Precursors to Potential Severe Core Damage Accidents: 1992. US. NRC. NUREG/CR-4674; ORNL/NOAC-232, V. 17. 1993.
3. Analysis of nuclear safety in diversification of Westinghouse fuel assemblies at WWER-1000. / V.I. Skalozubov, I. L. Kozlov, Yu. A. Komarov, O. A. Chulkin, O. I. Piontkovskyi. *Nuclear Physics and Atomic Energy*. 2019. Vol. 20, issue 2. P. 159–163. DOI: https://doi.org/10.15407/jnpae2019.02.159.
4. Исследование поведения топлива легководных реакторов в аварийных условиях. ФГУП «ГНЦ РФ НИИАР». «ВНИИНМ». РНЦ «Курчатовский институт». 7-я конф. по реакторному материаловедению. Демитровград, 8-12 сентября 2003.
5. Bibilashvili Yu.K., Sokolov N.B., Andreeva-Andrievskaya L.N. et al. Thermomechanical properties of zirconium-based alloys oxidized claddings in LOCA simulating conditions. Proc. IAEA Technical Committee Meeting «Fuel behavior under transient and LOCA conditions». (Halden, Norway, 10-14 September) 2001. P. 186–208.
6. Расчет теплогидравлических параметров для всех режимов эксплуатации оборудования РУ энергоблока № 3 ОП ЗАЭС. EP01/2016.100.ОД.1. Т.1. 2016. 566 с.
7. Корректировка и обновление ВАБ энергоблока № 5 ЗАЭС. EP25 – 2004.210.ОД.2 – Обесточивание энергоблока с отказом дизель – генераторов. Приложение G2.1. 2004. 365 с.

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ISSN 2223-3814 (online)

References

1. Antropov, V., Bukrinsky, A., & Shvyryaev, Yu. (1999). *Development of Methodology and List of BDBA for WWER-1000 for Quantitative Analysis.* SAM-99 Information Exchange Forum on "Severe Accident Management", 18 – 22 October 1999, Obninsk, Russia.
2. *Precursors to Potential Severe Core Damage Accidents: 1992*. (1993). US. NRC. NUREG/CR-4674; ORNL/NOAC-232, V. 17.
3. 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. DOI: https://doi.org/10.15407/jnpae2019.02.159.
4. *Investigation of the behavior of fuel of light-water reactors in emergency conditions.* (2003). FSUE "SSC RU RIIAR". "VNIINM". RRC "Kurchatov Institute". 7th Conference. on reactor materials science. Demitrovgrad, September 8-12.
5. Bibilashvili, Yu.K., Sokolov, N.B., & Andreeva-Andrievskaya, L.N. et al. (2001). *Thermomechanical properties of zirconium-based alloys oxidized claddings in LOCA simulating conditions*. Proc. IAEA Technical Committee Meeting «Fuel behavior under transient and LOCA conditions». Halden (Norway), 10-14 September 2001. 186–208.
6. *Calculation of thermohydraulic parameters for all operating modes of the equipment of the Reactor of power unit No. 3 of ZNPP.* (2016). ER01 / 2016.100. OD.1. T.1. 566 p.
7. *Correction and updating of PSA of power unit No. 5 of ZNPP.* (2004). EP25 - 2004.210.OD.2. Power failure of the power unit with diesel generator failure. Appendix G2.1. 365 p.

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