The Reliability Of Steam-Operated Emergency Feed Pumps

Vladimir Skalozubov[0000-0003-2361-223X], Denis Pirkovskiy[0000-0002-6638-9499], Vladislav Spinov[0000-0001-7555-847X], Igor Kozlov[0000-0003-0435-6373], YuriyKomarov[0000-0002-4696-6551]

Odessa National Polytechnic University, 1, Shevchenko Av., Odessa, Ukraine, 65044

kozlov_i.l_@ukr.net

Abstract. An original method of calculation and experimental qualification of quality and reliability an emergency feed pump with a steam drive from a steam generator is proposed to improve the efficiency of accident management with complete loss of long-term power supply at nuclear power plants. The use of an emergency feed pump with a steam drive from a steam generator is an alternative approach to passive heat removal systems through a steam generator, which, due to design and technical limitations, cannot ensure sufficient performance of safety functions to remove residual heat generation from a reactor and maintain the required level in steam generators in case of accidents with complete loss of long-term power supply. On the basis of the obtained qualification results, the technical requirements for an alternative feed pump with a steam drive were determined to compensate for ensuring the safety functions in case of failure of the design emergency feed pumps.

Keywords: Qualification, Quality And Reliability, Feed Pumps, Nuclear Power Plant.

1. Introduction

A promising approach to managing emergency situations with a complete loss of electricity is the development of emergency feed pumps with a steam drive from the steam generator. The main advantage of this approach in relation to known systems is that there is a fundamental possibility of the complete implementation of the requirements for emergency electronic energy sources for the reactor and the required level in the steam generator; without a reactor. A promising approach to emergency management at nuclear power plants with a complete loss of electricity is an emergency feed pump with a steam drive from the steam generator. The advantages of this approach to existing system systems are the fundamental possibility of a complete resolution of problems related to system security and electronic radiation. However, the use of emergency pumps with steam drive requires a deep study of their reliability. One of these issues is the reliability qualification for an emergency steam pump. An original method is proposed for modeling the conditions for the occurrence of water hammer when starting a steam driven pump. Certain conditions for the prevention of water hammer and pressure amplitudes due to the characteristics of emergency feed pumps and steam generators in a nuclear power plant. The pressure in the piping system of the emergency feed pump depends on the pressure in the steam generator. The conditions for the occurrence of water hammer correspond to excess pressure during operation. The pressure of water hammers is determined by the conditions of transition of kinetic energy of flow inhibition due to the momentum of water hammers. The results can be used in the design of emergency feed pumps with a steam generator, subject to additional experimental qualifications. The purpose and objective of the proposed work is to analyze the conditions for the occurrence of water hammers at the start of the AEFP, which determines its relevance.

2. Literature Review

One of the main causes of severe accidents and destructive steam-gas explosions at Fukushima-Daiichi NPPs in 2011 was the complete loss of the long-term power supply (CLLPS) due to beyond-design earthquakes and tsunamis. CLLPS led to the failure of active safety systems using pumping equipment with electric drives. Passive safety systems were also not sufficiently effective to manage the emerged emergency processes [1- 5].

In the post-Fukushima period, the further development and improvement of the passive heat removal systems (PHRS) for managing accidents with AEDP was significantly intensified. PHRS is a closed circulation loop with a heat exchange surface for cooling and condensation outside the containment / containment of a nuclear power plant (NPP). Convection heat exchange in PHRS is carried out due to natural circulation due to the different density of the vapor-gas medium at the lifting section and the density of condensate at the lower section of the circulation circuit. The greatest development and implementation found PHRS in the containment (CO) NPP.

Analysis of existing and projected PHRSCO (for example, [6-10]) has shown their sufficient effectiveness for reducing the pressure and temperature of the vapor-gas medium in GO (including in case of accidents in CLLPS). However, PHRSCO does not provide for the performance of safety functions for removing residual heat from the reactor and maintaining the required feed-water level in the steam generator in case of accidents with PHRS.

A prerequisite for the use of an emergency feed pump with a steam drive from the steam generator is the qualification of the emergency feed pump with a steam drive for reliability and performance in accidents with complete long blackout. The criteria for qualifying the emergency feed pump with a steam drive in transient and operating modes were determined to ensure the successful implementation of safety functions to remove residual heat from the reactor and maintain the required level of feed water in the steam generator; as well as preventing critical hydrodynamic shock for reliability. Based on the developed conservative thermo-hydrodynamic model “reactor-steam generator - emergency feed pump with steam drive”, the structural and technical requirements for the emergency feed pump system with steam drive that meet the established criteria and qualification conditions are determined. [11-16]

3. Research Methodology

The solution of these issues is possible on the basis of PHRS reactor installation (PHRS RI) and PHRS via a steam generator (PHRSSD). In particular, air-cooled and water-cooled PHRS SDare being implemented for Chinese nuclear power plants with pressurized reactors CNPPPR and CPWR 1000. Currently, large European projects

are being considered for the development and implementation of a PHRS SDfor WWER-1000 nuclear power plants.
However, with regard to the existing and projected PHRS CO, the following should be noted.

1. PHRS SDin case of accidents with CLLPS does not provide sufficient safety functions to remove residual heat from the reactor and maintain the required level in the SD. Thus, according to experts of the IAEA SPOT, WWER SDs provide for the removal of heat from the reactor up to 2% of the rated power. Therefore PHRSSDs should be considered as an auxiliary systems for accident management with CLLPS.

2. Improving the efficiency of PHRS SDin accidents with CLLPS is associated with significant technical difficulties. So, to fully compensate for the failure of an emergency electric feed pump (EEFP), which supplies SD feeds with a flow rate of 100 t / h, the total heat exchange surface of the PHRS SDover 3.0 105 m2, located at a height of several hundred meters above the protective shell of CO, is required.

Therefore, it is necessary to search for alternative approaches to accident management with PHRS. One of these approaches is considered in the works of Professor A.V. Korolev [11-15] also proposes to direct the steam generated during the SD accident to the steam drive of the alternative emergency feed pump (AEFP), which feeds the SD from the hydraulic storage tanks of the EEFP water supply. The main advantages of this approach are as follows.

1. The principal technical possibility of fully compensating for the failure of EEFP to perform the safety functions of removing residual heat generation and maintaining the required level in the SD.

2. The system AEFP is located inside the CO, which significantly increases the level of security in relation to the PHRS SD under external extreme influences.

However, the development and implementation of AEFP requires appropriate qualifications. In accordance with the IAEA terminology, qualification of systems important to the safety of nuclear power plants means a calculated / experimental / calculated and experimental justification for the performance and reliability of the safety functions. These issues are devoted to the proposed work, which determines its relevance.

3.1. Basic provisions of the qualification method of an emergency nutritional pump with steam drive. The defining characteristics of the AEFP qualification are the pressure of the pump pressure ΔP_p developed by the steam drive and dependent on the inlet pressure to the steam drive P_v (pressure in the steam generator), as well as the inertia indicator of the pressure characteristics:

$$I_p = \frac{\mathrm{d}\Delta P_p(P_v)}{\mathrm{d}P_v} \tag{1}$$

Therefore, the main task of qualification of AEFP is to determine the required (critical) head and inertia index to ensure adequate characteristics of the design EEFP.

The equation of flow in the system AEFP with a quasistationary pressure P_v in SD:

$$\frac{L}{\Pi}\frac{\mathrm{d}G}{\mathrm{d}t} = \Delta P_p(P_v) - \Delta P_0 - \frac{\xi}{\rho\Pi}G^2 \qquad (2)$$

where L, P - the length and area of the bore of the pipeline system AEFP, respectively; G is the mass flow rate; t is time; $\Delta P_0 = P_v - P_b$; P_b - pressure in hydraulic containers AEFP; ξ - total hydraulic resistance coefficient of the system; ρ is the density of the water cooler.

The design qualification model (2) leads to the solution of the problem:

$$\frac{\mathrm{d}G}{\mathrm{d}t} = a - bG^2 \qquad (3)$$

$$G(t=0) = 0 \qquad (4)$$

where $a = \frac{\Delta P_p - \Delta P_0}{L}\Pi$, $b = \frac{\xi}{\rho\Pi^2}$.

Solution (3), (4) has the form:

$$G = \sqrt{\frac{a}{b}}\,\frac{\exp(2\sqrt{ab}t) - 1}{\exp(2\sqrt{ab}t) + 1} \qquad (5)$$

The required (critical) pressure ΔP_{cr} can be determined from the condition G ($t = t_0$) = G_A in the formula (5):

$$G_A = \sqrt{\frac{a(\Delta P_{\mathrm{cr}})}{b}}\,\frac{\exp\left[2\sqrt{a(\Delta P_{\mathrm{cr}})b}t_0 - 1\right]}{\exp\left[2\sqrt{a(\Delta P_{\mathrm{cr}})b}t_0 + 1\right]}, \qquad (6)$$

where G_A is the EEFP design flow (about 100 t / h); t_0 is the start time of the AEFP.

The maximum allowable start time of the pump t_{0m} can be determined from the condition of complete boil-off of the feedwater of the steam generator along the 2nd circuit with the mass M from the moment of the beginning of the accident:

$$\mathrm{M}(i_v - i_l) = \alpha F t_{0\mathrm{m}}, \qquad (7)$$

where i_v, i_l is the specific (per unit mass) enthalpy of vapor and liquid, respectively; α is the coefficient of inter-contour heat transfer; F is the total surface area of inter-loop heat exchange.

Then from the heat balance equation (7) follows:

$$t_{0\mathrm{m}} = \frac{\mathrm{M}(i_v - i_l)}{\kappa F} \tag{8}$$

In the general case, the solution of equation (6) with respect to ΔP_{cr} can be obtained by numerical methods. In the particular case provided

$$G_A \sqrt{\frac{a}{b}} << 1 \tag{9}$$

the solution (6) for the required head ΔP_{cr} of the AEFP with the quasistationary pressure SD has the form:

$$\Delta P_{\mathrm{cr}} = \Delta P_0 + \frac{G_A L}{\Pi t_0} \tag{10}$$

Based on the results obtained, the following technical requirements for the AEFP steam line can be formulated:

$$\Delta P_p(P_v) \geq \Delta P_{\mathrm{cr}}\,,\, I_p \geq 0\,,\, t_0 << t_{0m} \tag{11}$$

Other technical requirements are related to the need to prevent overflow of SDs in the 2nd circuit. Overfilling of SDs can lead to hydraulic impacts on the casing and SD internals, as well as to the failure of the AEFP steam line directly. Therefore, the valve of the AEFP system must be adjustable for feedwater level meters.

The requirements of the technical conditions for the control valve AEFP are to provide the necessary level in the SD:

$$h_{\min} \leq h(\xi) \leq h_{\max}\,, \tag{12}$$

where h, h_{min}, h_{max} - current, minimum and maximum allowable level of feed water in SD, respectively.

Under conditions of significant non-stationarity of pressure in the SD in the course of an accident, the equation of flow in the AEFP system is:

$$\frac{L}{\Pi}\frac{\mathrm{d}G}{\mathrm{d}t} = \Delta P_p(t=0) + \int_0^t I_p \frac{\mathrm{d}P_v}{\mathrm{d}\tau}\mathrm{d}\tau - P_v(t) + P_v - \frac{\xi(t)}{\rho \Pi^2} G^2 \tag{13}$$

In the general case, the solution (13) with the initial condition (4) can be obtained by numerical methods, taking into account the results of the simulation of thermo-hydrodynamic processes in the SG for specific emergency conditions.

4. Results

The key issue of qualification of AEFP is to determine the real characteristics:

pressure characteristics of the steam drive ΔP_p;

the inertia indicator of the pressure characteristics I_p (P_v);

time and conditions for starting the pump;

minimum permissible pressure in the steam drive, ensuring the efficiency of the pump, etc.

These defining qualification parameters can be obtained empirically at appropriate experimental facilities. However, experimental installations must comply with the requirements of ensuring the thermal-hydrodynamic similarity of processes in full-scale and experimental installations [16].

In the criterial form, the equation of motion (2) has the form:

$$\frac{d\mathbf{G}}{d\mathbf{t}} = \mathbf{K}_1 - \mathbf{K}_2\mathbf{G}^2 \tag{14}$$

where $\mathbf{G} = G / G_A$, $\mathbf{t} = t / t_0$, $\mathbf{K}_2 = \frac{\xi G_A t_0}{\rho \Pi L}$

The conditions for the similarity of thermo-hydrodynamic processes in full-scale and experimental installations in this case are the similarity of the similarity criteria K_1 and K_2:

$$\mathbf{K}_1 \equiv \text{idem}, \quad \mathbf{K}_2 \equiv \text{idem}, \tag{15}$$

5. Conclusions

1. The analysis carried out in the work showed that passive heat removal systems (without electrically driven pumps) due to structural and technical limitations cannot ensure sufficient safety functions for removing residual heat generation in the reactor and maintaining the required level in steam generators in case of accidents with complete loss of long-term power supply.

2. An alternative approach to solving these issues may be the use of an emergency feed pump with a steam actuator from a steam generator. However, the use of such a pump requires an appropriate computational and experimental qualification of efficiency / reliability for the conditions of accidents with complete loss of long-term power supply.

3. A method for qualifying an emergency feed pump with a steam actuator from a steam generator is proposed. Dependences are obtained for determining the required (critical) pump head with a steam drive from the pressure in the steam generator and

the design and technical parameters of an alternative system, as well as for the maximum allowable time for the pump to start.

4. On the basis of the obtained results of design qualification, the requirements of technical conditions for an emergency feed pump with a steam drive are determined to ensure the safety functions for removing the residual heat generated by the reactor and maintaining the required level of feed water in the steam generator.

5. Determined the need for experimental qualification in relation to the defining parameters of the performance / reliability of an emergency feedwater pump with a steam-drive based on the criteria of thermal-hydrodynamic similarity of full-scale and experimental installations

References

1. International Fact Finding Expert Mission of the Fukushima-Daiichi NPP Accident Following the Great East Japan Earthquake and Tsunami, IAEA Mission Report, 160 p., (2011)
2. Fukushima Nuclear Accident Analysis Report [Electronic resource] / Tokyo Electric Power Company. — Tokyo: TEPCO, http://www.tepco.co.jp/en/press/corp-com/release/betu12_e/images/120620e0104.pdf, (2012)
3. Gauntt Randall. Fukushima Daiichi Accident Study Report / Randall Gauntt, Donald Kalinich, Jeff Cardoni [et al.] // Sandia National Laboratories. —298 p., (2012)
4. Kozlov I., Skalozubov V., Oborsky G.: Development of methods for the reassessment of nuclear safety, taking into account the lessons of a major accident at nuclear power plants Fukushima-Daiichi, LAP LAMBERT Academic Publishing, (2014).
5. Skalozubov V., Kozlov I., Chulkin O.: Revision of nuclear power plants safety systems’ routine testing assigned periodicity during the design extention period, Problems of atomic science and technology №5(111), pp. 53-56 (2017)
6. Naffea H., Gerliga V., Shevelev D., Balashevsky A.: Assessing the effectiveness of passive heat removal from the containment of a VVER RP reactor under prolonged blackout conditions, Nuclear and Radiation Safety 2 (58), pp. 27-31 (2013)
7. Naffaa, Kh M., D. V. Shevielov, and A. S. Balashevskyi. "Modeling Calculation and Assessment of Effectiveness of Containment Heat Removal Passive System in Case of Severe Accident on NPP with WWER-1000." Global nuclear security № 3 (8) (2013)
8. Balashevskyi, A. S., V. A. Gerliga, and N. I. Vlasenko. "Calculation substantiation of passive system of pressure reduction in PWR containment under condition of leakage accident with continuous blackout of power-generating unit. "Global nuclear security № 2 (7) (2013)
9. Naffea H., Dubkovsky V.: Classification of systems for passive removal of residual heat from protective shells of nuclear reactors, Pratsi Odessa Polytechnic University 1 (43), pp. 104-112. (2014)
10. M.P. Vyshemirsky, O. I. Zhabin, S. A. Ostapchuk. Analiz podpitki parogeneratora ot mobil’noy nasosnoy ystanovky pri polnom obestochivanii energobloka s reaktornoy ystanovkoy VVER-1000/V-320. [Analysis of steam generator feed from a mobile pumping unit with a complete blackout of a power unit with a VVER-1000 / V-320 reactor]. Nuclear and radiation safety, 4(72), pp. 25-31, (2016)
11. Korolev A., Derevianko O.: Composite design of a turbine driver of a pumping unit for backup feeding of steam generators of nuclear power plants, Pratsi Odessa Polytechnical University 1 (43), pp. 93-97.(2014)

12. Korolyov, O. V., O. V. Derevyanko, and O. Yu Pogosov. "Twin-rotor combined turbine driver with a transmission for the nuclear power plant equipment emergency water supply system." Pratsi Odessa Polytechnic University 2, pp. 88-91. (2014)
13. Korolev A., Derevianko O.: Backup feeding of steam generators of nuclear power plants in terms of electrical power supply of a power unit, Nuclear and radiation safety 2 (62),pp. 10-12.(2014)
14. O.V. Derevyanko, A.V.Korolev, A.Yu. Pogosov. Rotornye elementy kombinirovannyh turbonasosnyh agregatov dlya avtomatyzirovannoy sistemy avariynoy sistemy avariynoy podpitki teplomassoobmennogo oborudovaniya AES. [Rotor elements of combined turbopump units for an automated emergency feed system for heat and mass transfer equipment of nuclear power plants] Nuclear and radiation safety. 4(63), pp. 31-35, (2014)
15. A.V. Korolev, O.V. Derevyanko, A.Yu. Pogosov. Novye apparaty podpitki teplomassoobmennogo oborudovaniya v sisteme upravleniya energoblokom. [New devices for feeding heat and mass transfer equipment in the power unit control system]. Energy saving. Energetic. Energy audit.№8 (126), pp. 28-34, (2014)
16. Mazurenko A., Skalozubov V., Pirkovskiy D., Chulkin O.: Analysis of the applicability of the results of experimental studies of hydrodynamics to the pumping systems of thermal and nuclear power plants, Nuclear power engineering and environment 1 (9),pp. 49-52 (2017)