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THE OPTIMAL CORRECTING THE POWER VALUE OF A NUCLEAR POWER PLANT POWER UNIT REACTOR IN THE EVENT OF EQUIPMENT FAILURES

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The paper considers the problem of adjusting the power of a nuclear power plant power unit reactor for those cases when equipment failure occurs. In such circumstances, sometimes it is sufficient to reduce the reactor power, while maintaining the probabilistic level of safe operation of the power unit. The rational number of the reactor power is determined by solving the problem of minimizing the risk criterion in the integral root mean square context. In order to demonstrate the efficiency of the proposed approach, a numerical example is considered. The approach outlined in the paper is focused on improving the power unit control in case of equipment failures.

Keywords: *nuclear power plant, power unit, safety, risk, power, failure.*

Introduction

Nowadays, the community of scientific and engineering specialists in the field of nuclear energy has accepted that one of the conditions for the safe operation of nuclear power units is the assessment of the probability of a design basis radiation accident $P(A)$, which should not exceed the maximum permissible value P^*: $P(A)\le P^*$, $P^*=(10^{-5}–10^{-4})$ year^{-1}. This condition, given, in particular, in [1, 2], equals to an expert assessment. It is not a proven requirement that could apply to design basis accidents and could guarantee the safety of the power unit.

Condition $P(A)\le P^*$ has a numerical confirmation. Thus, the results of the analysis of radiation risk from possible accidents at the Kursk (RBMK (high-power channel-type reactor) reactors) and Rostov (VVER (water-water power reactor) reactors) NPPs are given in [3]. The estimates are given as follows.

For the Kursk NPP, the probability of beyond design basis accidents is estimated by the value 4.0×10^{-8} year^{-1}; and for design basis accidents, the probability (frequency) of accidents is 10^{-4} year^{-1}.

For the Rostov NPP, the probability of beyond design basis accidents is estimated by the value 3.8×10^{-8} year^{-1}; for design basis accidents probability of accidents is 5.6×10^{-5} year^{-1}.

Information on design basis radiation accidents (Table 1) is given in [4], with reference to sources.

Table 1. Design basis radiation accidents

Facility name	Year of the accident	INES level	Losses over 10 years, billion $	Probability, 1/year
Chornobyl NPP-4	1986	7	600	1.29×10^{-4}
Fukushima-1, 2, 3	2011	7	100	1.58×10^{-3}
Kyshtym	1957	6	50	4.18×10^{-3}
TMI-2	1979	5	20	1.51×10^{-2}
Windscale	1957	5	10	3.98×10^{-2}

The values of the estimates were determined by conducting a probabilistic safety analysis. Such an analysis has already been carried out for many power units. It shows [4] that the probability of design basis accident can be estimated, on average, by the value 10^{-4} year^{-1}. However, such an estimate, at least due to the uncertainty of part of the data, does not form the maximum permissible level of the security parameter P^*, violation of which, even in a short period of time, requires the reactor shutdown. It can only be assumed that the "stricter" the requirement for P^* is, for example, P^*=(10^{-6}–10^{-5}) year^{-1}, the more likely a higher level of NPP safety can be expected. Also, the level of safety is positively affected by an increase in the number of applied criteria, for example, the use of risk criterion along with the criterion P^*.

An analysis of accidents at nuclear power plants made it possible to note the following trend in [5]: accidents with large consequences have a relatively low probability of occurrence, and accidents with small consequences have a relatively high probability of occurrence.

Risk assessments are associated with the solution of two main problems – calculation of the probabilities of radiation accidents and determination of the consequences of these accidents.

The probability of accidents is usually calculated using the method of events "tree" and failures "tree" [1, 2, 3, 5].

It consists in creating a set of scenarios for the occurrence and development of possible accidents with an assessment of their frequencies.

Determination of the scope and types of consequences of radiation accidents is a complex and lengthy process. Some results of research in this direction are given in [3, 4, 5].

At the stage of energy generation, the fulfillment of the requirements for ensuring the power unit safety conditions in case of equipment and systems failures is achieved in various ways, in particular, by adjusting the reactor power.

Mathematical problem statement

During the problem statement, the following main notations, assumptions and conditions are set.

1. τ_0=0 – time for the power unit to reach the nominal mode; τ – current time; t – power unit shutdown time.

2. The following reliability indicators are used: the probability of failure and the probability of failure-free operation over time τ; probability of failure on demand.

3. General causes that can lead to a radiation accident (floods, aircraft crashes, loss of consumers, loss of power supply, etc.) are not taken into account.

4. The power unit is given in the form of two structures: security systems and other systems. Security systems S_m, m=1, 2, …, M fend off accidents. They are independent from each other and from other systems in a probabilistic context. Other systems D_n, n=1, 2, …, N are the systems of normal operation; systems important for safety; auxiliary systems. Each of the systems S_m and D_n can be in two states: serviceability and failure.

$h(\tau_n)$ – accident-initiating event that occurs when a system D_n fails at time τ_n.

The corrected reactor power after equipment and system failures can be obtained as follows.

Suppose that a complex event A_k, k=1, 2, …, K is formed by events that are determined by the full set of failure and serviceability states of each safety system S_m, m=1, 2, …, M over time τ. Set of all events A_k, k=1, 2, …, K forms a complete group. These events are incompatible. Number of all events is $K=2^M$. Combination of these events forms an event $A_1+A_2+\ldots+A_K$, which can be defined as the ability of a security system to fend off triggering events $h(\tau_n)$ n=1, 2, …, N, occurring over time τ_n, as a result of failure of systems D_n, n=1, 2, …, N. Event probability $h(\tau_n)$ is marked as $P(h(\tau_n))$. Events $h(\tau_n)$ and $A_1, A_2, \ldots, A_K$ are considered as independent events due to the constructive architecture of the power unit.

If the event $E(\tau_n)$ – accident triggered by event $h(\tau_n)$, then the probability of the accident $P(E(\tau_n))$ can be calculated by the total probability formula [6]:

$$P(E(\tau_n)) = P(h(\tau_n))\sum_{k=1}^{K} P(A_k)\cdot P(E(\tau_n)\setminus h\cdot A_k)\,. \quad (1)$$

Formula (1) can be refined by assuming each event A_k as a multiplication of events consisting of failure-free operation and failure of systems S_m, m=1, 2, …, M. For example, suppose that the event A_k is possible to be assumed as a multiplication of independent events: $A_k = S_1(\tau_n)\cdot S_2(\tau_n)\cdot\ldots\cdot\overline{S}_k(\tau_n)\cdot\ldots\cdot\overline{S}_{M-1}(\tau_n)\cdot\overline{S}_M(\tau_n)$, where $S_k(\tau_n)$ – failure-free system operation, $\overline{S}_k(\tau_n)$ – system failure over time τ_n.

Event probabilities $S_k(\tau_n)$ and $\overline{S}_k(\tau_n)$ over time τ_n can be determined by calculating their reliability indicators. That is, the probability of the event A_k, $P(A_k)$ over time τ_n, due to the independence of safety systems, can be defined as multiplication of the corresponding probabilities:

$$P(A_k) = P(S_1(\tau_n))\cdot P(S_2(\tau_n))\cdot\ldots\cdot P(\overline{S}_k(\tau_n))\cdot\ldots\cdot P(\overline{S}_{M-1}(\tau_n))\cdot P(\overline{S}_M(\tau_n))\,.$$

Extending this expression to an arbitrary time interval $\tau \in (0,t)$, we get that the probabilities $P(A_k(\tau_n))$ and $P(E(\tau_n))$ in formula (1) are functions of time τ, and, if necessary, their values can be calculated through reliability indicators of systems S_m m=1, 2, …, M.

The potential danger of accidents is characterized by risk. The risk value is determined by multiplication of the accident probability value and the accident consequences value (damage). The enumeration of the types of consequences of accidents, which may be several, is indicated in the text by the symbol $\xi=1, 2, \ldots, \xi^*$.

The values of the consequences of radiation accidents depend on many factors, including the reactor power: the lower the reactor power is, the smaller is the consequence value.

Each consequence of the accident can be correlated with the monetary equivalent of the costs expected to be expended to eliminate this consequence over time θ.

Suppose that $c^{\xi}_{\max}(\theta)$ – the maximum financial cost that can be expected to be spent over time θ for liquidation of consequence ξ at the maximum accident, and $c^{\xi}_{n}(\theta)$ – the amount of financial costs that are likely to be spent on liquidation of an accident consequence ξ over time θ in case of failure of system D_n, n=1, 2, …, N.

In this paper, the value of the ξ-type consequence from triggering events $h(\tau_n)$ is defined in dimensionless form:

$$g^{\xi}_{n}(\theta) = \frac{c^{\xi}_{n}(\theta)}{c^{\xi}_{\max}(\theta)} \cdot \frac{W_n}{W_0},\ n=1, 2, \ldots, N,\ \xi=1, 2, \ldots, \xi^*, \quad (2)$$

where W_n – reactor power at time τ_n; W_0 – rated power of the reactor.

Taking into account (2) the value of the ξ-type risk of accident $E(\tau)$ will be determined by the formula:

$$r^{\xi}(\theta,\tau) = g^{\xi}_{n}(\theta) \cdot P(E(\tau)) = \frac{c^{\xi}_{n}(\theta)}{c^{\xi}_{\max}(\theta)} \cdot \frac{W_n}{W_0} \cdot P(E(\tau)),\ n=1, 2, \ldots, N,\ \xi=1, 2, \ldots, \xi^*. \quad (3)$$

Since it is assumed that, on average, the probability of a radiation accident is $P(E(\tau)) \le P^*$ year^{-1}, then from (3) follows the risk estimate:

$$r^{\xi}(\theta,\tau_0) \le \frac{c^{\xi}_{n}(\theta)}{c^{\xi}_{\max}(\theta)} \cdot \frac{W_n}{W_0} \cdot P^* \ \text{year}^{-1},\ n=1, 2, \ldots, N,\ \xi=1, 2, \ldots, \xi^*. \quad (4)$$

Risk-based approaches to solving safety problems are associated with risk regulation. As already noted, due to the dependence of risk on a number of factors, the normalized risk can be determined in various ways.

For example, if we assume and fix the values $c^{\xi}_{n}(\theta)$ and $c^{\xi}_{\max}(\theta)$, and also take the values t=1 year, W_n=W_0, then $\forall \xi = 1,2,\ldots,\xi^*$ and $\forall n = 1,2,\ldots,N$ from (4) followed by an option of dimensionless normalized risk assessment:

$$r^{\xi}(\theta)_{\text{norm}} \le \frac{c^{\xi}_{n}(\theta)_{\text{fixed}}}{c^{\xi}_{\max}(\theta)_{\text{fixed}}} \cdot P^* \ \ \text{year}^{-1}. \quad (5)$$

In problems where the value of the normalized risk is used, the inequality $P(E(\tau)) \le r^{\xi}(\theta)_{\text{norm}}$ determines a higher level of requirements for the power unit safety than the inequality $P(E(\tau)) \le P^*$.

The value of the corrected reactor power in case of equipment and system failures can be determined by solving the following problem.

Problem. Suppose that at moments of time $\tau_1, \tau_2, \ldots, \tau_N$ triggering events $h_1 \vee h_2, \ldots h_N$, consisting in failures of systems $D_1, D_2, \ldots, D_N$ that cannot be restored during operation, occur. The type of accidents consequences (parameter ξ); and the value of the normalized risk – $r^{\xi}(\theta)_{\text{norm}}$ are selected.

It is required to find such values of the corrected reactor powers W_n, which for all time intervals (τ_n, t), n=1, 2, …, N would ensure the fulfillment of the following conditions:

1) new risks $r^{\xi}(\theta, \tau_n, t)$ must be different from previous risks over time (τ_n, t) at least in the integral mean square context;

2) new risks $r^{\xi}(\theta, \tau_n, t)$ should not be greater than the normalized risk $r^{\xi}(\theta)_{\text{norm}}$ over time (τ_n, t).

Solution. Applying the integral method of least squares [7] and taking into account that W_0, $c^{\xi}_{\max}(\theta)$ are constants, the first requirement of the problem can be written as:

$$\delta_n = \min_{0<W_n<W_{n-1}} \left(\frac{1}{W_0 c^{\xi}_{\max}(\theta)}\right)^2 \cdot \int_{\tau_n}^{t} \{c^{\xi}_{n-1}(\theta)\cdot W_{n-1}\cdot P(E(\tau_{n-1})) - c^{\xi}_{n}(\theta)\cdot W_n \cdot P(E(\tau_n))\}^2 d\tau, \qquad (6)$$

To find the minimum δ_n according to the desired parameter W_n, it is needed to find the corresponding derivative and equate it to zero. By differentiating expression (6) with respect to the parameter W_n, we get:

$$-\frac{1}{2}\frac{d\delta_n}{dW_n} = -\frac{1}{2}\left(\frac{1}{W_0 c^{\xi}_{\max}(\theta)}\right)^2 \cdot \int_{\tau_n}^{t} \{c^{\xi}_{n-1}(\theta)\cdot W_{n-1}\cdot P(E(\tau_{n-1})) - W_n \cdot c^{\xi}_{n}(\theta)\cdot P(E(\tau_n))\}\cdot c^{\xi}_{n}(\theta)\cdot P(E(\tau_n)) d\tau = 0 \qquad (7)$$

Since $W_0 c^{\xi}_{\max}(\theta) \neq 0$, $c^{\xi}_{n}(\theta)\cdot P(E(\tau_n)) \neq 0$, then from (7) it follows that the integrand is equal to zero, and then the value of the corrected reactor power W_n will be equal to:

$$W_n = W_{n-1}\cdot \int_{\tau_n}^{t} c^{\xi}_{n-1}(\theta)\cdot P(E(\tau_{n-1}))d\tau \Big/ \int_{\tau_n}^{t} c^{\xi}_{n}(\theta)\cdot P(E(\tau_n))d\tau. \qquad (8)$$

Substituting the found value W_n into formula (3), the new risk function $r^{\xi}(\theta, \tau_n)$ on the time interval (τ_n, t) is obtained.

If the value of the new risk function $r^{\xi}(\theta, \tau_n, t)$ does not exceed the value of normalized risk $r^{\xi}(\theta)_{\text{norm}}$ in the time interval (τ_n, t), then the operation of the power unit in this time interval will not contradict the requirement of its safety according to the risk criterion.

If the value of the new risk $r^{\xi}(\theta, \tau_n, t)$ exceeds the risk value $r^{\xi}(\theta)_{\text{norm}}$ at some point τ_z, then it is necessary to continue adjusting the reactor power in the time interval (τ_n, t), based on point τ_z.

Thus, solution (8) gives a solution to the problem and describes the sequence of reactor power adjustment after equipment and systems failures without restoring their operability.

Example. As an example, the following problem is considered: adjusting the power value of the power unit reactor with a VVER in case of failure to close one of the main safety valves of the pressure compensator pulse-safety device. The purpose of the pressure compensator pulse-safety device systems is to discharge overpressure steam from the pressure compensator into the pressure relief tank. If the main pressure relief valve, which opened on demand, does not close, then this can lead to damage of the reactor core – an accident.

To solve the problem, the following data are taken.

1. Event $h(\tau_1)$ – refusal of the main pressure relief valve to close at the time τ_1. $P(h(\tau_1))$=0.05 request^{-1} – the probability of this event according to the data [1].

2. W_0=3000 MW – rated thermal power of the reactor.

3. τ_0=0 h – time for the power unit to reach the nominal mode; τ_1=4000 h – time of failure of the main pressure relief valve to close;

4. t=8000 h – shutdown time of the power unit for refueling;

5. $c^{\xi}_{0}(\theta) = c^{\xi}_{1}(\theta)$ – possible costs of eliminating damage from an accident over time θ after a failure.

6. Emergency core cooling systems (ECCS) and reactor protection system (RPS) form that part of the safety system, which is designed, in particular, to fend off the failure of the main pressure relief valve to close [8].

7. A_1, A_2, A_3, A_4 – a list of possible states of the safety system, which in this example consists of ECCS and RPS. The probabilities of correct operation and the probabilities of system failure are taken from the data [5].

A_1 – serviceable ECCS system and serviceable RPS system, P_{ECCS}=0.9 request^{-1} and P_{RPS}=0.9 request^{-1};

A_2 – serviceable ECCS system and failure state of the RPS system, P_{ECCS}=0.9 request^{-1} and Q_{RPS}=0.1 request^{-1};

A_3 – failure state of the ECCS system and serviceable RPS system, Q_{ECCS}=0.1 request^{-1} and P_{RPS}=0.9 request^{-1};

A_4 – failure state of the ECCS system and failure state of the RPS system, Q_{ECCS}=0.1 request^{-1} and Q_{RPS}=0.1 request^{-1}.

8. $P^*=10^{-4}$ year^{-1} – maximum permissible value of the accident probability.

9. $E(\tau_1)$ – accident. An event that consists of the failure of the safety system (ECCS and RPS) to fend off the triggering event $h(\tau_1)$; $P(E(\tau_1))$ – the probability of this event.

10. Values $P(E(\tau_1)\setminus h\cdot A_1)=0$, since with a working system of ECCS and RPS an accident will not occur; $P(E(\tau_1)\setminus h\cdot A_2)=P(E(\tau_1)\setminus h\cdot A_3)=0.9$ – according to [5]; $P(E(\tau_1)\setminus h\cdot A_4)=1.0$, since the failure of the ECCS and RPS systems leads to an accident.

The value of the accident $P(E(\tau_1))$ probability can be calculated by the total probability formula (1). The terms in this formula (1) will be equal to:

$$P(E(\tau_1)\setminus h\cdot A_1)=P(h)\cdot P_{\text{ECCS}}\cdot P_{\text{RPS}}\cdot P(E(\tau_1)\setminus hA_1)=0.095\cdot 0.95\cdot 0.95\cdot 0=0;$$

$$P(E(\tau_1)\setminus h\cdot A_2)=P(h)\cdot P_{\text{ECCS}}\cdot Q_{\text{RPS}}\cdot P(E(\tau_1)\setminus hA_2)=0.095\cdot 0.95\cdot 0.1\cdot 0.9\approx 0.00812;$$

$$P(E(\tau_1)\setminus h\cdot A_3)=P(h)\cdot Q_{\text{ECCS}}\cdot P_{\text{RPS}}\cdot P(E(\tau_1)\setminus hA_3)=0.095\cdot 0.05\cdot 0.95\cdot 0.9\approx 0.00812;$$

$$P(E(\tau_1)\setminus h\cdot A_4)=P(h)\cdot Q_{\text{ECCS}}\cdot Q_{\text{RPS}}\cdot P(E(\tau_1)\setminus hA_4)=0.095\cdot 0.05\cdot 0.05\cdot 1.0\approx 0.00024.$$

Adding these values, we get that the accident probability is $P(E(\tau_1))\approx 0.01648$.

Then, substituting these results into formula (8), the corrected reactor power is obtained:

$$W_1=3000\cdot\int_{4000}^{8000}P^*d\tau\Big/\int_{4000}^{8000}0.01648d\tau\approx 4000\cdot 10^{-4}/4000\cdot 0.01648\approx 3.658\ \text{MW}.$$

Thus, in case of failure to close one main pressure relief valve, the value of the corrected reactor power will take on the value: $W_1\approx 3.66$ MW.

By changing the input data, in particular P^*, it is possible to obtain other results. For example, if we allow $P^*=10^{-3}$ request^{-1}, then we get: $W_1\approx 36.2$ MW.

The stronger the requirement for the maximum allowable probabilities P^* of radiation accidents at the power unit (P^* decreases) is, the lower the value of the corrected reactor power should be; the weaker the requirement for the maximum permissible probabilities P^* of radiation accidents (P^* increases) is, the greater the value of the corrected reactor power should be. This conclusion follows from formula (8) and corresponds to the meaning of the power unit safety.

Conclusion

The reactor power value is one of the power unit safety control parameters. In a number of operational cases, when equipment and systems fail, the reactor power adjusting makes it possible to fend off the risk of an accident. The value of the corrected reactor power can be determined by solving a mathematical problem – minimizing the value of risk change in the integral mean square context.

Calculation of the probabilities of correct operation and failures of the power unit equipment and systems can be performed by applying the method of the events and failures "tree" [1] or the method of the systems structural reliability calculation [9].

The numerical result of solving the problem given in the example indicates the possibility of using the method proposed in the paper for determining the value of the corrected reactor power in case of failures of non-recoverable equipment and systems of the power unit.

This method can be useful for improving the management of energy generation by power units. Appropriate software, focused on the application of the method in practice, can be included as an independent software package in the information and analytical systems of nuclear power plants.

References

1. Bakhmetyev, A. M., Samoylov, O. B., & Usynin, G. B. (1988). *Metody otsenki i obespecheniya bezopasnosti YaEU* [Metody otsenki i obespecheniya bezopasnosti NPP]. Moscow: Energoatomizdat, 136 p. (in Russian).
2. Rahn, F. J., Adamantiades, A. G., Kenton, J. E., & Braun, C. (1984). A guide to nuclear power technology: A resource for decision making. New York: Wiley, 986 p.
3. Islamov, R. T., Derevyankin, A. A., Zhukov, I. V., Berberova, M. A., Glukhov, I. V., & Islamov, D. R. (2011). Risk assessment of nuclear power plants. *Atomic Energy*, vol. 109, pp. 375–379. https://doi.org/10.1007/s10512-011-9371-y.

4. Borysenko, V. I. (2012). *Pro deiaki zakonomirnosti naslidkiv avarii na AES* [About some regularities of consequences of accidents at nuclear power plants]. *Problemy bezpeky atomnykh elektrostantsii i Chornobylia – Problems of nuclear power plants' safety and of Chornobyl*, iss. 18, pp. 6–15 (in Ukrainian).
5. Weaver, L. (1980). *Risk ot avarii na AES s legkovodnymi reaktorami* [Risk from accidents at nuclear power plants with light water reactors]. In book: *Bezopasnost yadernoy energetiki* [Safety of Nuclear Energy] by Rast, J., Weaver, L. (eds.). Moscow: Atomizdat, pp. 114–133 (in Russian).
6. Venttsel, Ye. S. (1964). *Teoriya veroyatnostey* [Probability theory]. Moscow: Nauka, 576 p. (in Russian).
7. Demidovich, B. P., Maron, I. A., & Shuvalova, E. Z. (1967). *Chislennyye metody analiza* [Numerical methods of analysis]. Moscow: Nauka, 365 p. (in Russian).
8. Voronin, V. A. (1981). *Osobennosti ekspluatatsii i remonta AES* [Features of operation and repair of nuclear power plants]. Moscow: Energoizdat, 168 p. (in Russian).
9. Zevin, L. I. & Krol, H. H. (2019). Calculation of indicators of reliability of technical systems by the typical structural scheme method. *Journal of Mechanical Engineering – Problemy mashynobuduvannia*, vol. 22, no. 2, pp. 53–59. https://doi.org/10.15407/pmach2019.02.053.

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Оптимальне коригування величини потужності реактора енергоблока АЕС у разі відмови обладнання

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В статті розглянуто задачу про коригування потужності реактора енергоблоку АЕС для тих випадків, коли відбувається відмова обладнання. У таких обставинах іноді достатньо знизити потужність реактора, зберігаючи при цьому імовірнісний рівень безпечної експлуатації енергоблоку. Раціональна величина потужності реактора визначається шляхом розв'язання задачі мінімізації критерію ризику в інтегральному середньоквадратичному сенсі. З метою демонстрації працездатності запропонованого підходу розглянуто численний приклад. Викладений у статті підхід орієнтований на вдосконалення управління енергоблоком при відмови обладнання.

***Ключові слова**: АЕС, енергоблок, безпека, ризик, потужність, відмова.*

Література

1. Бахметьев А. М., Самойлов О. Б., Усынин Г. Б. Методы оценки и обеспечения безопасности ЯЭУ. М.: Энергоатомиздат, 1988. 136 с.
2. Справочник по ядерной энерготехнологии: пер. с англ. / Ф. Ран, А. Адамантиадес, Дж. Кентон, Ч. Браун. Под ред. В. А. Легасова. М.: Энергоатомиздат, 1989. 752 с.
3. Исламов Р. Т., Деревянкин А. А., Жуков И. В., Берберова М. А., Глухов И. В., Исламов Д. Р. Оценка риска для АЭС. *Атомная энергия*. 2010. Т. 109. № 6. С. 307–311.
4. Борисенко В. І. Про деякі закономірності наслідків аварій на АЕС. *Проблеми безпеки атомних електростанцій і Чорнобиля*. 2012. Вип. 18. С. 6–15.
5. Уивер Л. Риск от аварий на АЭС с легководными реакторами. Безопасность ядерной энергетики / под ред. Дж. Раста, Л. Уивера. М.: Атомиздат, 1980. С. 114–133.
6. Вентцель Е. С. Теория вероятностей. М.: Наука, 1964. 576 с.
7. Демидович Б. П., Марон И. А., Шувалова Э. З. Численные методы анализа. М.: Наука, 1967. 365 с.
8. Воронин В. А. Особенности эксплуатации и ремонта АЭС. М.: Энергоиздат, 1981. 168 с.
9. Zevin L. I., Krol H. H. Calculation of indicators of reliability of technical systems by the typical structural scheme method. *Journal of Mechanical Engineering – Problemy mashynobuduvannia*. 2019. Vol. 22. No. 2. P. 53–59. https://doi.org/10.15407/pmach2019.02.053.