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MODELING OF ULTRA-FAST ACCIDENT COOLING OF THE WALL OF THE REACTOR VESSEL WWER-1000

Abstract: *Zaporizhzhia Nuclear Power Plant (NPP) is under temporary occupation. There are reasonable suspicions regarding compliance with all the norms and standards of Ukraine and the IAEA recommendations. Deliberate sabotage measures are possible, the result of which will lead to an emergency cooling mode of the WWER-1000 reactor vessel. The reactor vessel resource determines the service life of the entire NPP unit. The complex and dangerous situation that has developed at the Zaporizhzhya NPP, due to possible emergency modes with ultra-rapid cooling of the reactors, may lead to their failure, and therefore to the failure of the Zaporizhzhya NPP units. The emergency mode of ultrafast cooling of the reactor at a speed of 1800 degrees per hour is considered with a numerical solution of the problem of unsteady heat conductivity. The effect of this mode on the cyclic damage of the metal of the reactor vessel and on its brittle strength was evaluated. An appropriate prognosis of its radiation lifetime was made. For this, the worst postulated critical crack, for which the predicted lifetime of the reactor is the smallest, is determined. The express methodology of the specified calculation assessment for the reactor vessels was tested, which contains all the necessary sections according to the norms and recommendations of the IAEA: thermal conductivity, stress state, cyclic damage, radiation lifetime. The methodology is relevant and useful for assessing the safety of reactors of the WWER type and negative consequences, including in the conditions of terrorist actions of the occupation administration.*
Keywords: Nuclear power plant, safety, water-water energy reactor, reactor pressure vessel, weld, reactor cooling accident mode, numerical methods, non-stationary thermal conductivity, strength, stress state, cyclic damage of metal, resistance to brittle fracture, radiation lifetime, express calculation methodology.

Introduction. Zaporizhzhia NPP, the largest in Europe, is under temporary occupation. There is information that there are almost no highly qualified Ukrainian personnel left at this nuclear plant. Who exactly replaced the Ukrainian personnel, which persons, according to which profession, experience and with what qualifications perform the functions of managing this large, extremely complex and responsible facility - it is not known definitively. There is some information about the presence of representatives of the foreign concern Rosatom, whose professional capabilities are questionable. In addition, there are well-founded suspicions about the possibility of the occupiers undermining safety-important elements and systems of the Zaporizhia NPP (primarily the current state of the systems and elements of units 4 and 5). In particular, there is a danger of intentional detonation of steam pipelines, the loss of which will lead to an accident cooling mode of the WWER-1000 reactor vessel. The lifetime of the reactor vessel determines the lifetime of the entire NPP unit. And among the NPP accident modes dangerous for the WWER-1000 reactor vessel, modes with its rapid cooling are the most dangerous for its integrity and radiation lifetime. Thus, the complex and dangerous situation that has developed at the Zaporizhia NPP, due to possible accident modes with ultra-fast cooling of the reactors, may lead to their loss, and thus, to the loss of units of the Zaporizhia NPP.

The latest research and publications. Starting from 2014, the authors have been developing the relevant special expert express methodology ([1], [2]). The goal is to have a reliable scientific-technical and engineering-methodological tool that would allow at any time, in a short period of time and without significant costs, to obtain verified and conservative results of assessments of emergency modes of nuclear power plants, as well as their consequences for the integrity, cyclic damage [2] and radiation resource [1] of the bodies of WWER reactors (primarily WWER-1000). Also, the authors adhere to the requirements of all norms and standards of Ukraine and IAEA recommendations.

The goal of the work is the development of express methodology to assess the most dangerous scenario for the reactor vessels. Non-stationary thermal conductivity. Considering the above, this study is devoted to modeling the accident mode with ultra-fast cooling of the reactor vessel. To achieve its goal, the most severe design and real accident modes, which negatively affect the technical condition of the WWER-1000 reactor vessel, were previously analyzed.

Main assumptions. Among such regimes, the attention of the authors was drawn to the mode of cooling the reactor body at a rate of 1800°C per hour, which took place on October 22, 1985 at Unit 1 of the South Ukrainian NPP [7]. In the future, for the convenience of teaching, this mode is designated as "22.10.85" (by the date of the event). Note that the WWER-1000 reactors vessels of the Zaporizhzhya NPP and the South Ukrainian NPP are similar and have the same design, main design dimensions and manufacturing materials (Figure 1).

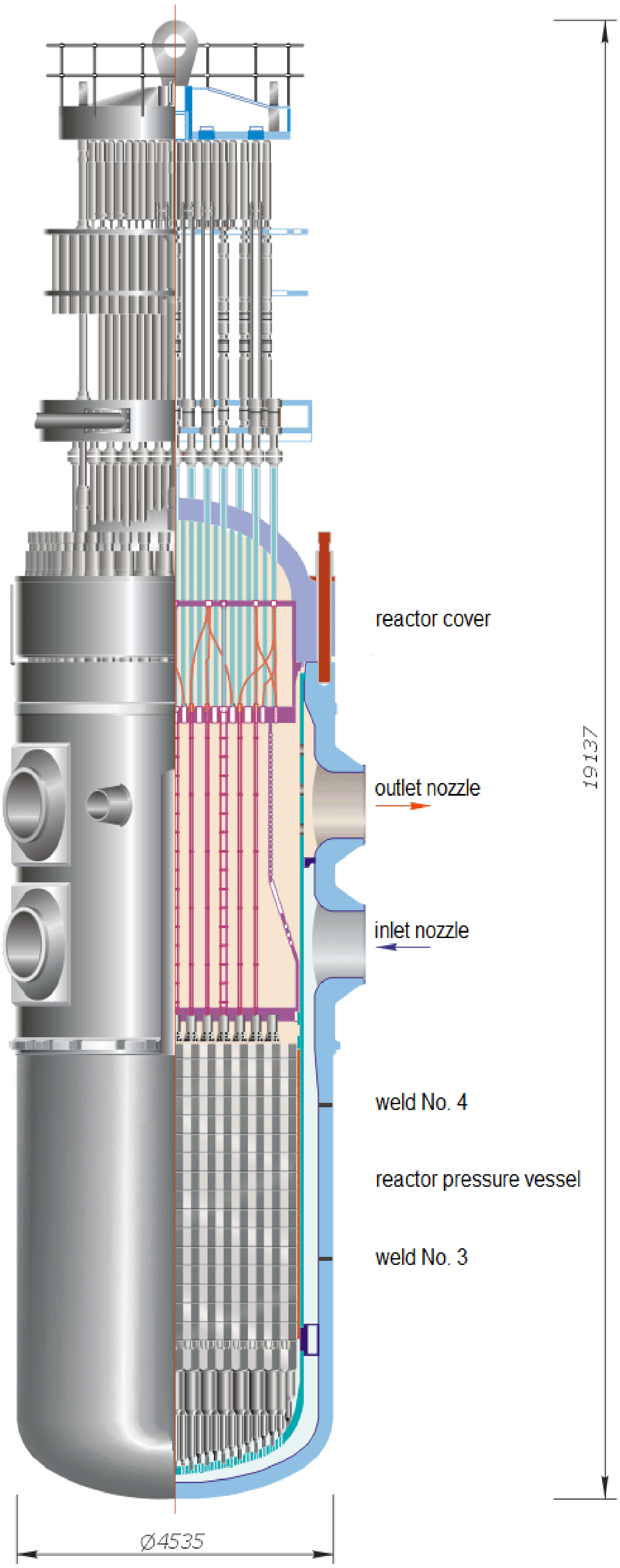

Figure 1. The WWER-1000 reactor and its relevant elements, which are important when considering accidents with its cooling

Given the lack of detailed technical data on the scenario of the flow of the "22.10.85" mode (results of measurements of coolant control devices or previously performed calculations), the cooling of the reactor vessel wall is modeled as follows.

In a general three-dimensional formulation, the problem of non-stationary thermal conductivity is determined by the Fourier-Kirchhoff equation, which relates the change in temperature T of a volumetric body at any of its points (x, y, z) at a given time t:

$$\rho c\frac{\partial T}{\partial t}=\frac{\partial}{\partial x}\left(\lambda\frac{\partial T}{\partial x}\right)+\frac{\partial}{\partial y}\left(\lambda\frac{\partial T}{\partial y}\right)+\frac{\partial}{\partial z}\left(\lambda\frac{\partial T}{\partial z}\right)+Q_w(x,y,z,t,T), \qquad (1)$$

where ρ – density; c – specific heat capacity; λ – thermal conductivity coefficient; $Q_w(x, y, z, t, T)$ – the power of internal heat sources.

To assess the strength of the reactor body in a certain zone of its wall and the acceptable accuracy of this express assessment, it is enough to take into account the following assumptions: during emergency cooling of the wall of the reactor body, the temperature of the metal changes along the thickness of the wall (in the radial direction), and in the other two perpendicular directions (along the ring and vertically) there is no temperature gradient. Then the three-dimensional equation of non-stationary thermal conductivity (1) reduces to a one-dimensional equation:

$$\rho c\frac{\partial T}{\partial t}=\lambda\frac{\partial^2 T}{\partial x^2},\ 0<x<\delta, \qquad (2)$$

where δ – the thickness of the wall of the reactor body.

The currently known emergency cooling rate of the inner surface of the reactor vessel wall is equal to *1800°C/hour = 0.5°C/sec*.

It is assumed that, taking into account this speed, the temperature of the inner surface of the reactor wall at the beginning of the mode decreases from *290°C* to *100°C* in *380 seconds* and in the further development of the mode is kept within *100°C* until the disappearance of the temperature gradient along the thickness of the reactor wall at the end of the mode. The corresponding duration of the mode is calculated and is *15,000 seconds*.

Taking this emergency scenario into account in the express assessment requires appropriate management of the calculation model. Therefore, in the numerical solution of the problem of non-stationary thermal conductivity (2), the boundary condition of the second kind was used for the inner wall of the reactor body:

$$x=0;\ -\lambda\frac{\partial^2 T}{\partial x^2}=q(t), t>0, \qquad (3)$$

where *q(t)* – the time-varying cooling flow under the condition $q(t) < 0$.

The outer surface of the wall of the reactor body $(x = \delta)$ is under the influence of the external environment (external thermal insulation).

Management of the calculation model of cooling is carried out by determining the law of the cooling flow *q(t), W/m²* on the inner wall of the reactor body:

$$q(t) = \begin{cases} 0 \le t \le 380\,;\ -97500; \\ 380 \le t \le 530\,;\ -97500 + 18750 \cdot \left(1 - cos\left(\frac{\pi\cdot(t-380)}{0{,}6\cdot t - 168}\right)\right); \\ 530 \le t \le 1000\,;\ 42{,}553191 \cdot t - 82553{,}19; \\ 1000 \le t \le 2500\,;\ 13{,}333333 \cdot t - 53333{,}33; \\ 2500 \le t \le 6000\,;\ 4{,}2857143 \cdot t - 30714{,}29; \\ 6000 \le t \le 15000\,;\ 0{,}5555556 \cdot t - 8333{,}333. \end{cases} \quad (4)$$

The *q(t)* graph shown in Figure 2 corresponds to this law.

The starting initial data for calculating the change in temperature fields in the wall of the reactor body during an emergency scenario are the law of change of the cooling flow *q(t)* (formula (4), Figure 2), wall thickness δ, initial temperature field in the wall for the mode of normal operation T_0, thermal insulation temperature T^{ins}, heat transfer coefficient on the outer surface of the wall κ, metal density ρ, metal thermal conductivity coefficient λ and its specific heat capacity c.

The calculation algorithm of the implicit four-point finite difference procedure is implemented, each iteration of which for the step *h (mm)* along the wall thickness from *0* to δ mm through the nodes $i = 1 \div N$ and for the step τ (sec) in the time of development of the emergency scenario from *0* to *15000 sec* after moments of time *n* the next.

The first running coefficients α_1 and β_1 are determined:

$$\alpha_1 = \frac{2a\tau}{h^2 + 2a\tau}; \quad (5)$$

$$\beta_1 = \frac{h^2}{h^2 + 2a\tau} T_1^n + \frac{2a\tau h q(t)}{\lambda(h^2 + 2a\tau)}.$$

where *a* – the coefficient of thermal conductivity:

$$a = \lambda/(\rho c).$$

The coefficients α_i and β_i of the running method are determined:

$$\alpha_i = \frac{A_i}{B_i - C_i \alpha_{i-1}};$$

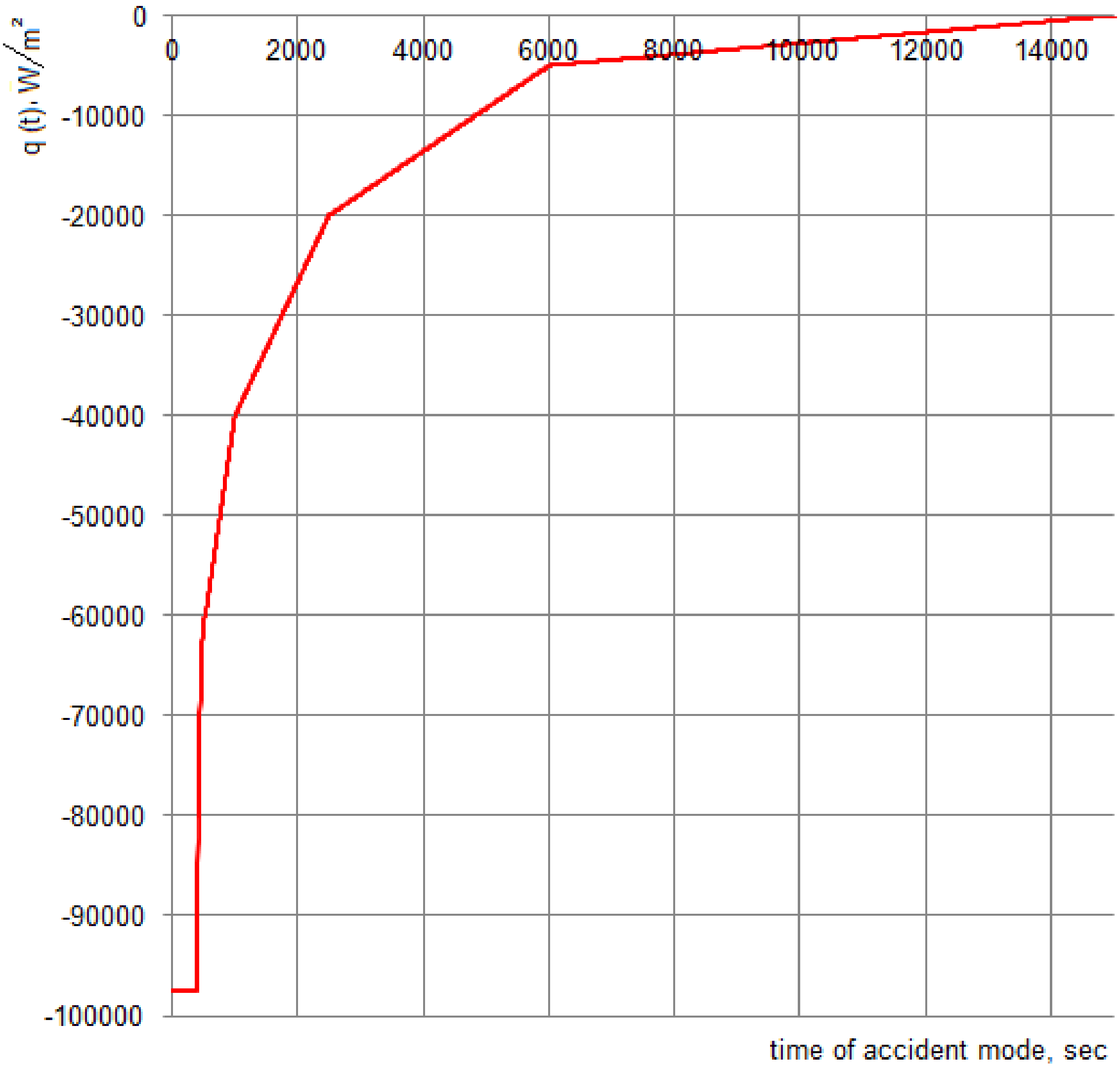

Figure 2. Graph *q(t)* that corresponds to the adopted law of modeling cooling flow *q(t)* for the "22.10.85" mode

$$\beta_i = \frac{C_i \beta_{i-1} - F_i}{B_i - C_i \alpha_{i-1}} \tag{6}$$

where: $A_i = \frac{\lambda}{h^2}$; $B_i = \frac{2\lambda}{h^2} + \frac{\rho c}{\tau}$; $C_i = \frac{\lambda}{h^2}$; $F_i = -\frac{\rho c}{\tau} T_i^n$.

The temperature of the metal on the outer surface of the wall of the reactor body T_N^{n+1} is determined:

$$T_N^{n+1} = \frac{\lambda h^2 T_N^n + 2a\tau\left(\lambda \beta_{N-1} + h\kappa T^{ins}\right)}{\lambda h^2 + 2a\tau\left(h\kappa + \lambda(1 - \alpha_{N-1})\right)}. \tag{7}$$

where κ – heat transfer coefficient; T^{ins} – the temperature of the external thermal insulation.

The temperature field in the wall at the moment of time *(n + 1)* is calculated, where the movement along nodes i along the thickness of the wall of the reactor body is performed from *(N - 1)* to *1*:

$$T_i^{n+1} = \alpha_i T_{i+1}^{n+1} + \beta_i.$$

For the "22.10.85" accident mode, the results of the calculation of non-stationary thermal conductivity through the wall of the reactor body at the level of weld No. 3 (Figure 1) were obtained, the scenario of the temperature gradient change for which is shown in the form of a surface in Figure 3.

Definition of a stress state. The scenario for the "22.10.85" mode of changes in annular and axial stresses in the wall of the reactor vessel at the level of weld No. 3, which corresponds to the above-mentioned temperature and pressure gradients in the reactor during operation, is performed in accordance with the Strength Norms [6]. At the same time, residual stresses from welding are also taken into account.

Sections of determination of the state of the metal and its prognosis. According to the requirements of the Strength Norms [6], the total brittleness temperature of the metal T_k of the reactor vessel under the action of radiation and cyclic loads is (without taking into account the effect of thermal aging):

$$T_k = T_{k0} + \Delta T_F + \Delta T_N,$$

where T_{k0} – initial brittleness temperature of the metal; ΔT_F – shift of brittleness temperature due to radiation load; ΔT_N – the shift in brittleness temperature due to cyclic damage.

The shift of the critical brittleness temperature from cyclic loads ΔT_N is determined according to clause 5.8.4.4 of the Strength norm [6] by the formula

$$\Delta T_N = 20 \cdot \sum_{i=1}^{m} \frac{N_i}{[N_0]_i},$$

where N_i – the number of load cycles in the i-th mode of operation; $[N_0]_i$ – the number of cycles allowed for the i-th operating mode; m – the number of modes.

The shift of the critical temperature of brittleness due to the influence of ionizing radiation ΔT_F is determined according to clause 5.8.4.5 [6] by the formula

$$\Delta T_F = A_F \cdot F^{\frac{1}{3}},$$

where A_F – radiation embrittlement coefficient $^{\circ}C$; F – transfer of neutrons with an energy not less than 0.5 MeV, reduced in 10^{22} neutrons/m^2.

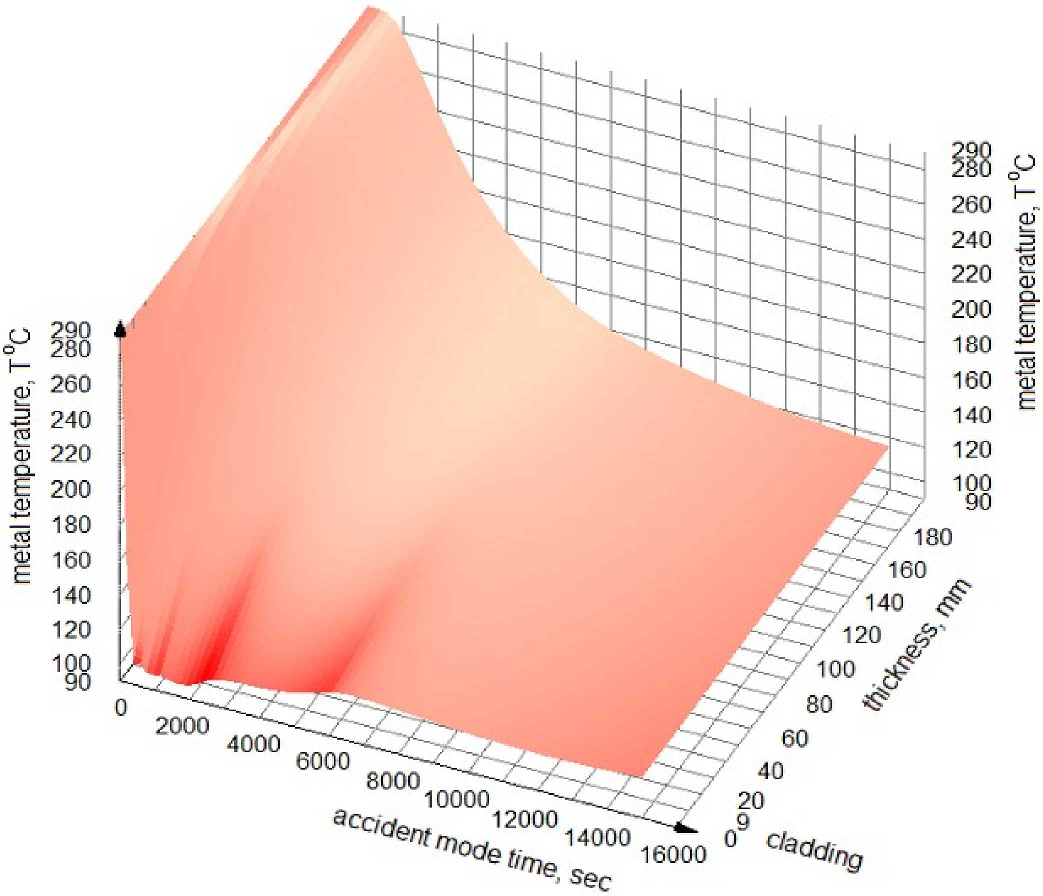

Figure 3: The scenario of the development in time (seconds) of the temperature gradient through the wall of the WWER-1000 reactor vessel at the level of weld No. 3 during the accident mode "22.10.85"

Thus, it is necessary to evaluate the cyclic damage and resistance to brittle fracture for the reactor vessel, as well as their joint effect on the strength and lifetime of the reactor vessel.

Assessment of cyclic metal damage. Let's evaluate the cyclic damage of the metal of the reactor vessel at the level of its weld No. 3, taking into account operational loads and the "22.10.85" regime. This calculated estimate is made in accordance with the requirements of section 5.6 of the Strength Norms [6].

The reduced stresses (σ_L) and conditional elastic reduced stresses (σ_F) are determined according to the requirements of section 5.3 of the Strength Norms [6]. Therefore, in order to determine the reduced stresses, it was necessary to determine the three main stresses. In addition, taking into account that 95.5% of the wall thickness is occupied by the base metal and only 4.5% by the surfacing, to determine the elastic state of the metal of the wall of the reactor vessel at the level of weld No. 3, the normative formulas of Table P3.17 of Appendix 3 of the Norm [6] were used of strength for a single-layer thick-walled cylinder, namely, for the calculation of radial

σ_r, annular σ_θ and axial σ_z stresses.

The results of determining the cyclic damage of the metal of the reactor vessel at the level of its weld No. 3 for the "22.10.85" mode are given with different scales along the vertical axis in Figures 4 (up to 60 mm wall thickness) and 5 (from 60 mm to 200 mm thickness walls).

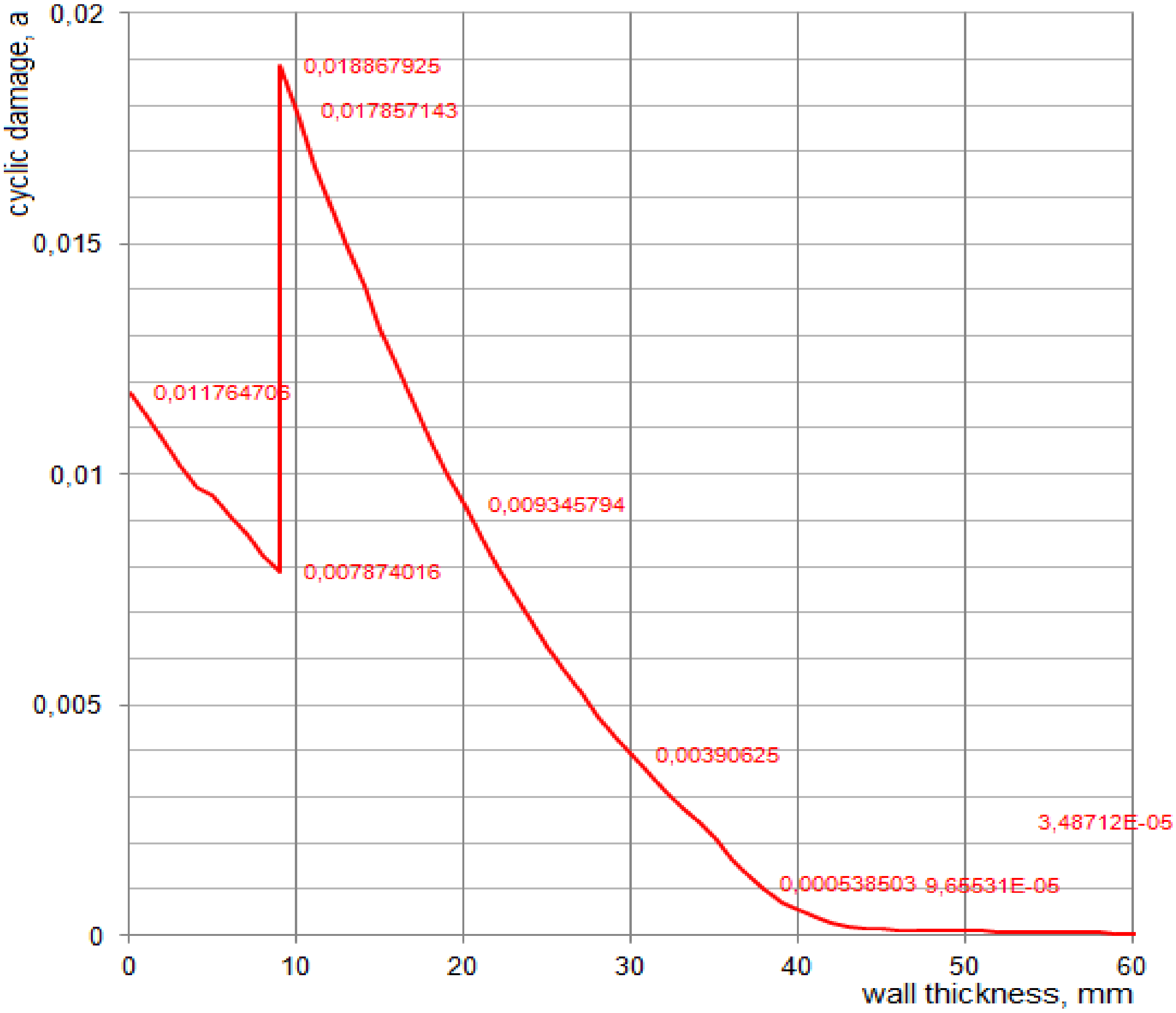

Figure 4: Graph of cyclic damage a of the metal of the WWER-1000 reactor vessel at the level of weld No. 3 for the normal operation and "22.10.85" modes on the wall thickness section from the inner surface to a depth of 60 mm (on a scale convenient for perception)

Evaluation of brittle strength and prognosis of radiation lifetime. When evaluating the brittle strength and lifetime of the reactor vessel, $\boldsymbol{a}$ set of postulated annular and axial cracks (9 defects each) with the following parameters was considered:

– depth a: 30 mm, 40 mm and 50 mm;
– depth to half-length ratio a/c: 0,2; 0,6 and 1,0,

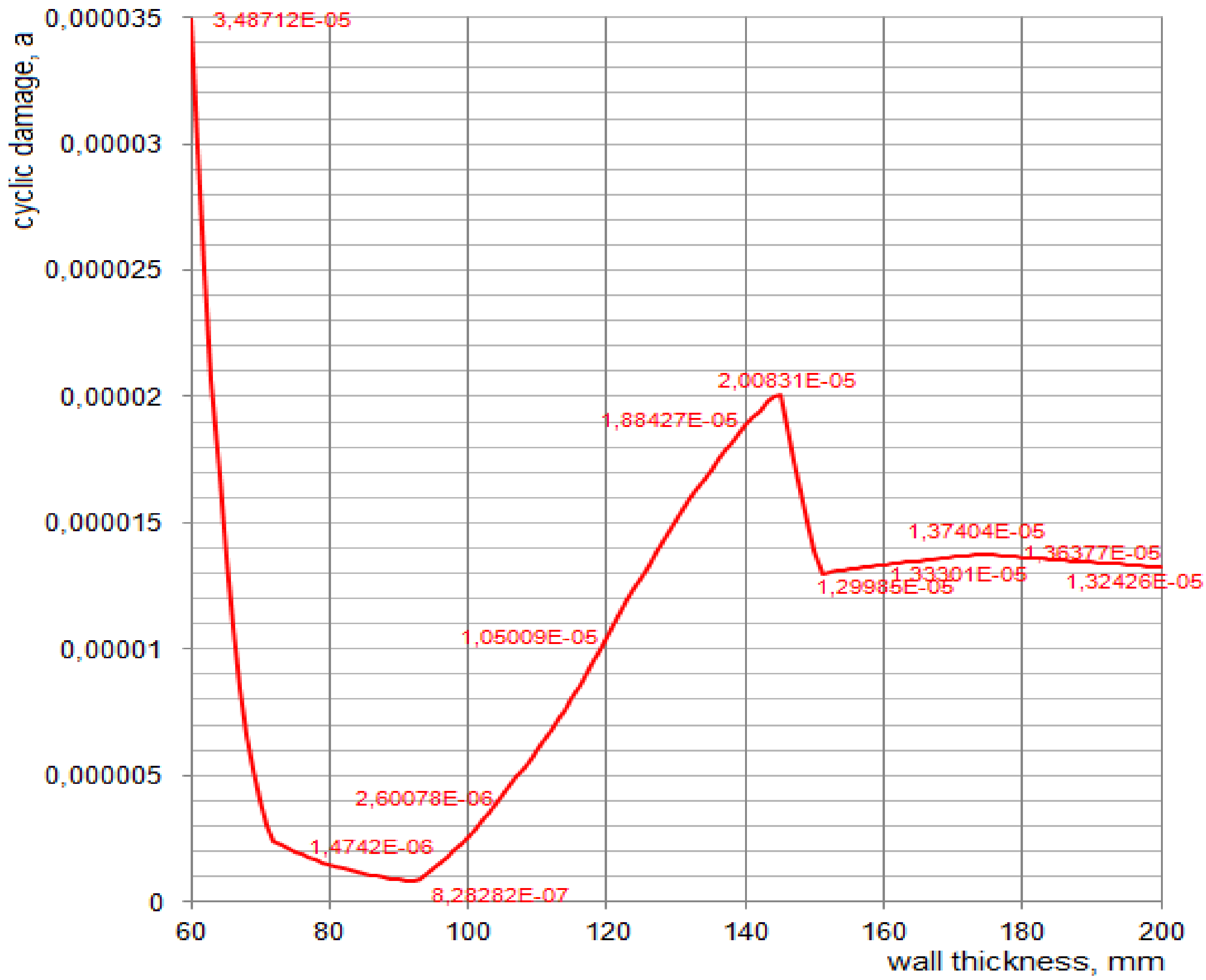

Figure 5: Graph of cyclic damage a of the metal of the WWER-1000 reactor vessel at the level of weld No. 3 for normal operation and "22.10.85" modes in the wall thickness section from a depth of 60 mm to the outer surface (on a scale convenient for perception).

which is within the regulatory limits in clause 5.8.5.2, and in item 4 of clause 5.8.7.2 of the Strength Norms [6], and in clause 6.3 of the IAEA recommendations [4].

The calculated assessment is performed by two methods according to:

– section 5.8, in particular - item 5.8.7.2 Strength standards [6];

– chapter 7, in particular, clause 7.3 of the IAEA recommendations [5].

The stress intensity factor of the first kind K_I was calculated by the method of weight functions according to [4].

When evaluation the lifetime of the reactor vessel, it is taken into account that:

– the operating time of the reactor of unit No. 1 of the South Ukrainian NPP, determined in years, is 2 (two) years more than calculated in the fuel campaigns;

– there is an additional shift of the critical fragility temperature ΔT_N from the operating modes and emergency mode "22.10.85".

The worst result for the considered set of postulated annular cracks corresponds to an annular crack with a depth of $a = 30\,mm$ and a ratio of $a/c = 0{,}2$. For her, the lifetime of the reactor vessel is equal to 5.59 years.

The worst result for the considered set of postulated axial cracks corresponds to an axial crack with a depth of a = 50 mm and a ratio of a/c = 0,2. For it, the lifetime of the reactor vessel is 2.91 years.

The results of calculating the resistance to brittle fracture of the reactor body according to the normative criterion for the specified critical case (axial crack with a depth of a = 50 mm with a ratio of a/c = 0,2) are presented in Figure 6 (obtained according to the IAEA recommendations [5]).

The result of the assessment of the resource/lifetime of the reactor body – 2.91 years – for the specified critical case is shown in Figure 7. The specified lifetime will be exhausted at a metal depth of $15.45\,mm$ at a temperature of 116.72 °C at 1710 seconds of the accident mode "22.10.85".

According to the data of the South Ukrainian NPP, at the end of the 38th fuel campaign (the last 40th year of operation according to the requirements of the passport and the reactor design), the reserve for fulfilling the conditions of brittle strength at the critical temperature of brittleness (the condition looks like $T_k < T_k^a$) for the worst design mode for the weld No. 3 is equal to 2.0°C for the radiation working time of the metal of the reactor vessel $37.3 \cdot 10^{18}$ neutrons/cm^2.

According to the results of the calculated assessment of the influence of the "22.10.85" regime and other regimes on the shift of the critical brittleness temperature (taking into account the shift of ΔT_N from fatigue cyclic damage), the following was found out (Figure 8):

1. According to the data analysis of the South Ukrainian NPP, the emergency critical brittleness temperature T_k^a for the worst design mode is probably equal to 51.5°C. Then the temperature T_k (without a margin of 2°C) is equal to 49.5°C;
2. Taking into account the shift $\Delta T_N = 20°C \cdot a = 20 \cdot 0.0308 = 0.62°C$ from common cyclic loads reduces the estimated value of the lifetime/operating time of the reactor by $1.1 = 43.4 - 42.3$ years.

Conclusions. The results of the computational modeling demonstrate the very negative consequences of ultra-fast accident cooling of the wall of the WWER-1000 reactor vessel. The design regimes correspond to the reactor's design lifetime of 40 years, when the "22.10.85" regime considered here corresponds to its radiation lifetime of less than 3 years; in addition to the above, the general cyclic damage of the metal of the reactor vessel, in the case of the implementation of its ultra-fast cooling regime like "22.10.85", contributes to the reduction of the lifetime/term of safe operation of the WWER-1000 reactor vessel by more than 1 year. Regarding the assessment methodology used here, the following can be noted.

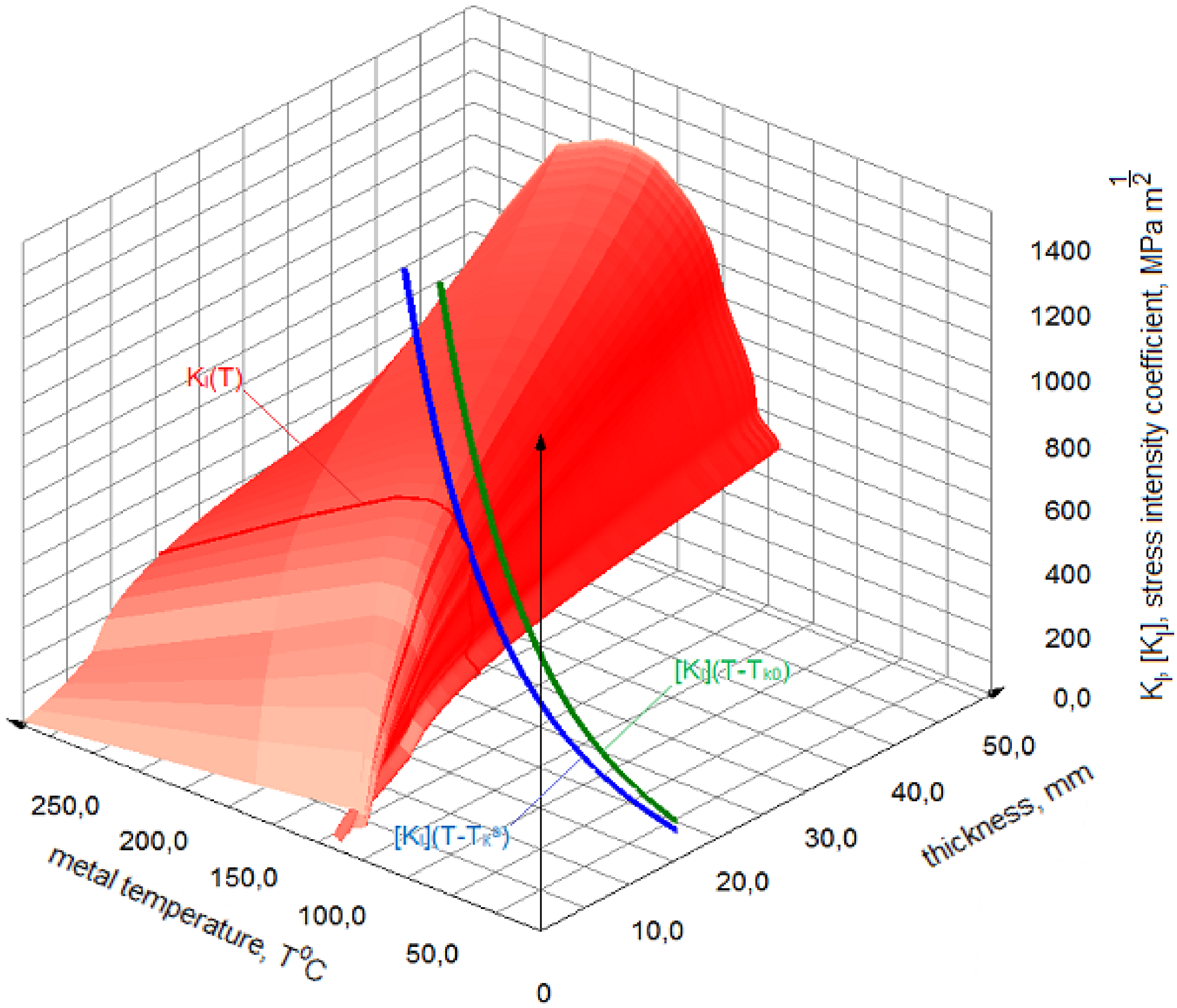

Figure 6: Graphs KI(T) and [KI](T), obtained according to the recommendations of the IAEA [5], which correspond to the emergency mode "22.10.85", the initial and limit state of the metal of weld No. 3 WWER-1000:

red surface – graph KI(T) for an axial semi-elliptical crack (a/c=0.2 and a=50 mm) for accident mode "22.10.85";

green curve – initial state of the metal of weld No. 3;

blue curve – the limit state of the metal of weld No. 3 at a depth of 15.45 mm at 1710 seconds of accident mode "22.10.85"

The used engineering and scientific and technical calculation methods correspond to the current norms, and their results are conservative and correct. This engineering and analytical express assessment was performed by the authors during one month of estimated working time. At the same time, as general experience shows, it is traditionally necessary to spend from six months to one year of working time on such a calculation using numerical methods and large computer programs.

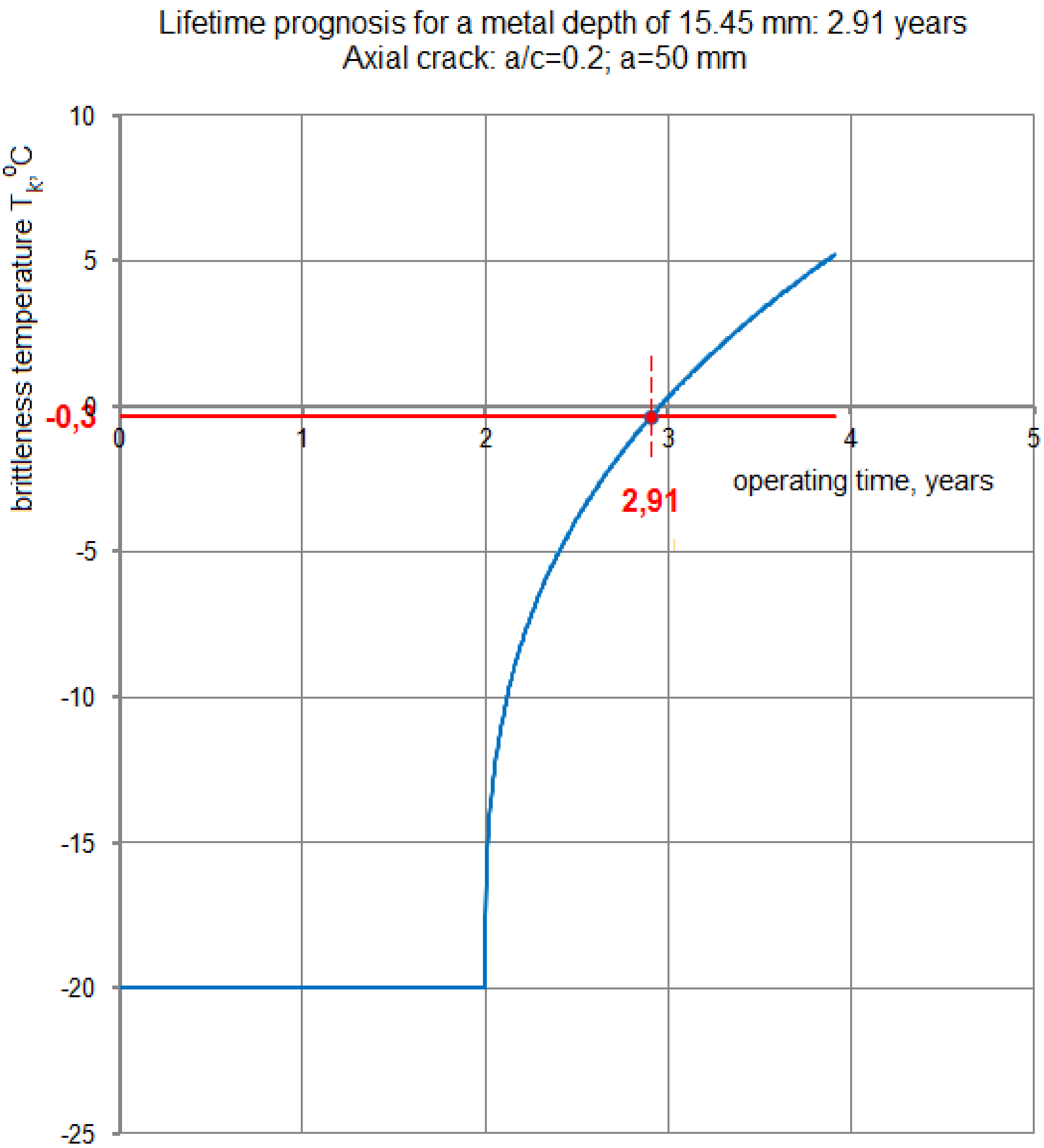

Figure 7: Graphs $T_k^a(F, a, t)$ and $T_k(F, a, t)$ of determining the radiation lifetime of the WWER-1000 reactor vessel for the "22.10.85" regime (taking into account the influence of radiation F and cyclic loads in time t (years)) for the metal of weld No. 3:
the red graph $T_k^a(F, a, t)$ corresponds to the red surface KI(T) in Figure 6;
the blue graph $T_k(F, a, t)$ corresponds to the transition of the initial state of the metal to the limit state (green and blue curves of Figure 6)

The results of calculations according to this express method can be clarified in case of obtaining additional data. In the extremely difficult modern conditions of war and in the tense working conditions of the nuclear industry of Ukraine, the use of such expert express assessments can be a useful and timely means for forecasting dangerous beyond-design consequences for nuclear power plants, including – after the intervention and terrorist actions of the occupiers, which is now possible at the site of the Zaporizhzhya NPP.

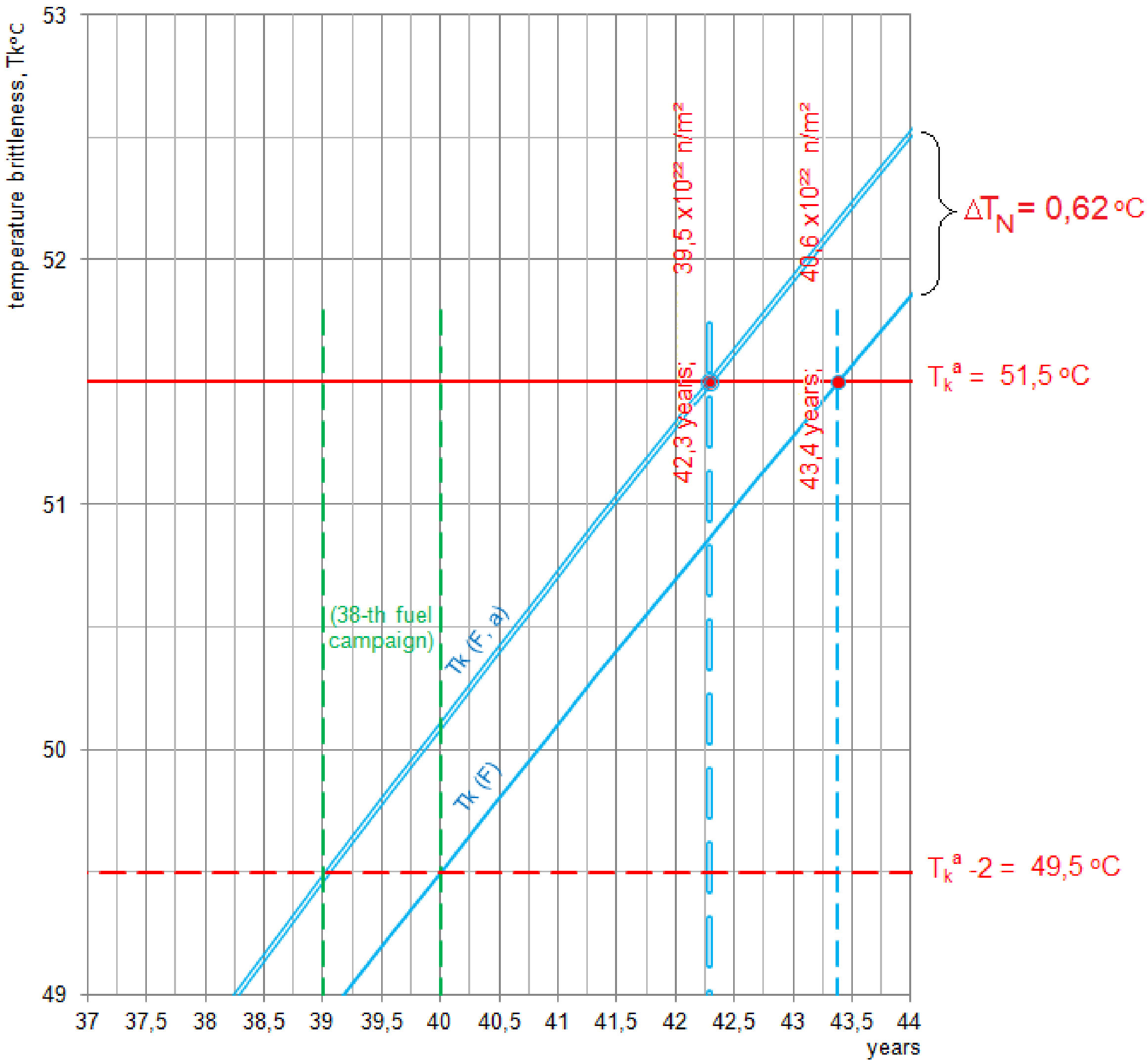

Figure 8: Negative impact on the resource of the WWER-1000 reactor vessel (weld No. 3) of combined radiation and cyclic loads taking into account the emergency mode "22.10.85" at the limit of the design designated lifetime of 40 years. Reduction of the period of safe operation of the WWER-1000 by 1.1 years.

In case of implementation of accident modes with ultra-fast cooling of the reactor at the Zaporizhzhya NPP, it is necessary to collect relevant data for the final assessment of the safety of the reactor vessel and, based on their results, to implement appropriate technical compensatory measures to ensure the safety of the nuclear installation. First of all, this concerns units 4 and 5 of the Zaporizhzhia NPP, where the occupation administration could implement extra-design modes and accident situations.

References

1. Popov V.V., Tryhub O.S., Mileikovskyi V. Ekspertne ekspres-otsiniuvannia vplyvu teplomasoobminnykh protsesiv na zalyshkovyi resurs korpusa reaktora WWER-1000 cherez okrykhkennia yoho metalu / V.V.Popov, O.S.Tryhub, V.Mileikovskyi. - Zbirnyk "Ventyliatsiia, osvitlennia ta teplohazopostachannia". – Kyiv, KNUBA, 2022. – Vyp.41. – s.39-49. DOI: https://doi.org/10.32347/2409-2606.2022.41.39-49.

2. Popov V.V., Tryhub O.S., Mileikovskyi V. Ekspertne ekspres-otsiniuvannia vplyvu teplomasoobminnykh protsesiv na zalyshkovyi resurs korpusa reaktora WWER-1000 cherez tsyklichnu poshkodzhuvanist / V.V.Popov, O.S.Tryhub, V.Mileikovskyi. "Ventyliatsiia, osvitlennia ta teplohazopostachannia". Kyiv, KNU-BA, 2021. Vyp.39. P.6-28. DOI: https://doi.org/10.32347/2409-2606.2021.39.6-28.

3. Chyrko L. I. “Comparison of Ukrainian and Russian approaches to determining the parameters of metal embrittlement of reactor vessels”. Abstracts of reports of the XIX annual scientific conference of the Institute of Nuclear Research of the National Academy of Sciences of Ukraine, January 24-27, Kyiv, Ukraine. 2012. P. 95-96.

4. Glinka G. “Development of weight functions and computer integration procedures for calculating stress intensity factors around cracks subjected to complex stress fields”. Progress Report: Stress and Fatigue-Fracture Design, Petersburg, Ontario, Canada. 1996. No. 1. 108 p.

5. IAEA-EBP-WWER-08 (Rev. 1). “Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants”. Revision 1. A publication of the extrabudgetary programme on the safety of WWER and RBMK nuclear power plants. Vienna, Austria. International Atomic Energy Agency. 2006. 65 p.

6. PNAE G-7-002-86. “Strength calculation standards for equipment and pipelines of nuclear power plants“ with amendments of 07 January 1987. Moscow: Energoatomizdat, 1989, 525 p.

7. Simonov E.Ya. “Extending the service life of NPP reactor units that have reached the end of their lifespan and constructing new NPPs is a dangerous technical gamble”. Atom.org.ua, Atomic energy in Ukraine and the world. Retrieved from: https://atom.org.ua/prodlenie-sroka-ekspluatatsii-reakto.htm (accessed 30.11.2023).

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МОДЕЛЮВАННЯ НАДШВИДКОГО АВАРІЙНОГО ОХОЛОДЖЕННЯ СТІНКИ КОРПУСУ РЕАКТОРА ВВЕР-100.

Анотація. *Найбільша в Європі Запорізька АЕС знаходиться в тимчасовій окупації. Є обґрунтовані підозри щодо дотримання всіх норм і стандартів України та рекомендацій МАГАТЕ щодо управління цим великим та надзвичайно складним та відповідальним об'єктом. Цілком ймовірно, що мали місце неодноразові порушення режимів роботи АЕС, а також можливі навмисні диверсійні заходи, в результаті яких буде введено режим аварійного охолодження корпусу реактора ВВЕР-1000. Ресурс корпусу реактора визначає термін служби всього блоку АЕС. А серед режимів аварії АЕС, небезпечних для корпусу реактора ВВЕР-1000, найбільш небезпечними для його цілісності та радіаційного ресурсу є режими з його швидким охолодженням. Таким чином, складна та небезпечна ситуація, що склалася на Запорізькій АЕС, через можливі аварійні режими з надшвидким охолодженням реакторів, може призвести до їх виходу з ладу, а отже, і блоків Запорізької АЕС. Ці питання розглядаються в роботі, а саме розглядається аварійний режим надшвидкого охолодження реактора ВВЕР-1000 зі швидкістю 1800 градусів на годину з чисельним рішенням задачі нестаціонарної теплопровідності методом скінчених різниць з використанням неявних різницевих схем. Оцінено вплив цього режиму на циклічне пошкодження металу корпусу реактора ВВЕР-1000 та на його крихку міцність. Зроблено відповідний прогноз його радіаційного життя. Для цього визначається найгірша постульована критична тріщина, для якої прогнозований термін служби реактора є найменшим. Апробовано експрес-методику зазначеної розрахункової оцінки для корпусів ВВЕР-1000, яка містить усі необхідні розділи згідно з нормами та рекомендаціями МАГАТЕ: теплопровідність, напружений стан,*

циклічні пошкодження, радіаційний ресурс. Методика є актуальною та корисною для оцінки безпеки реакторів типу ВВЕР та негативних наслідків, у тому числі в умовах терористичних дій окупаційної адміністрації (на прикладі Запорізької АЕС).

Ключові слова: Атомна електростанція, безпека, водо-водяний енергетичний реактор, корпус реактора, зварний шов, аварійний режим охолодження реактора, чисельні методи, нестаціонарна теплопровідність, міцність, напружений стан, циклічні пошкодження металу, стійкість до крихкого руйнування, радіаційний термін служби, експрес-методика розрахунку.