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Chapter

Probability Modeling Taking into Account Nonlinear Processes of a Deformation and Fracture for the Equipment of Nuclear Power Plants

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Abstract

At the solution of integrated tasks of strength, safe life and service safety maintenance for the nuclear power plants (NPP) equipment with slow reactors—water-moderated power reactors (WMPR) of VVER type and channel-type graphite-moderated power reactors (GMPR) of RBMK type arise necessity of physical and mathematical modeling of nonlinear processes of a deformation, fracture and damage at nonlinear probability statement. First of all, it concerns deriving determined, statistical and probabilistic characteristics of mechanical properties of reactor materials. Expectations and variation factors of mechanical properties' characteristics obtained from experimental researches are inducted into the equations for the verification calculations at determination of static and cyclic strength margins with the use of nominal and local stresses and strains. For the improved determined and probability analysis of these margins modeling experimental researches of stress-strain states of the analyzed equipment are conducted. Special attention at such tests is given to concentration factors and variation factors of loading conditions. The final stage of estimation of basic normative and verification calculation accuracy at laboratory, modeling and test bench researches are full-scale pre-operational tests (cold-hot running-in) of pilot nuclear reactors with the use of the experimental mechanics methods. The conditions of safety service of the NPP equipment are estimated taking into account factors of reaching limiting states by criteria of risk of initiation of emergency situations.

Keywords: probability modeling, strength, deformation, damages, fracture, nuclear reactors, safety service, risk, structural materials

1. Introduction

The era of nuclear energy in the world started in 1954 by putting into service the first nuclear power plant (NPP)—the Obninsk NPP with a channel-type reactor and

power of 5 MW. Since then, leading countries of the world (the USSR-Russia, the USA, Great Britain, France, etc.) have come up with a whole spectrum of a new type of power supply—nuclear-powered.

By 2019, in the Russian Federation, 10 NPPs with 35 power-generating units with a total power of 29 GW are operational. In model of the NPP of Russia, there are 20 pressurized water reactors, including water-moderated power reactors (12 units of VVER-1000 type, 1 unit of VVER-1100 type, 2 units of VVER-1200 type, 5 units of VVER-400 type, and 1 unit of VVER-417 type). There are also 13 units of channel boiling water reactors of a high power of RBMK type (channel-type graphite-moderated power reactor—GMPR)—(10 units of RBMK-1000 type and 3 units of type EPG-6 type with power of 12 MW) and 2 units of fast-neutron reactor (FNPR) of BN type (BN-600 type and BN-800 type).

In 56 states of the world, more than 430 nuclear reactors with a total power 370 GW is now operated. The NPPs in the world produce about 11% of the consumable electric power. Leaders in this production are France (80%), South Korea (32%), and Ukraine (30%). In Russia, this share amounts to 16%. In the long term of 20–25 years, probably accretion of this share will be about 25%.

On changeover to reactors of power plants of first generations of 1960–1970 reactors of new third and fourth breeds come. And if the first reactors were considered as "nuclear boilers" and designed on norms of boiler fabrication for thermal power, up-to-date reactors develop on these details both on scientifically well-founded norms and on methods of national (Russia, the USA, Great Britain, France, and Germany) and international levels (IAEA).

From stands of classes of hazards detection for technosphere objects, nuclear reactors undoubtedly fall into critically (CRO) and strategically (SRO) relevant objects. These are facts that demand the profound combined analysis and a justification of all design and service solutions for all stages of their life cycle.

In the proximal (till 2020), midrange (till 2030), and kept away (till 2050) prospects, the evolution of nuclear energetics will be carried out on the basis of operating, built, and designed nuclear power plants. Basis of the fundamental and application analysis of strength, life time, reliability, and safety of operation of NPP elements with reactors of VVER, RBMK, and BN types (**Figure 1**) in regular both emergency situations are the equations and criteria linear and nonlinear mechanics of deformation and fracture [1–11]. They contain in home and foreign strength standards and are used as at design, so at manufacture and operation of working in
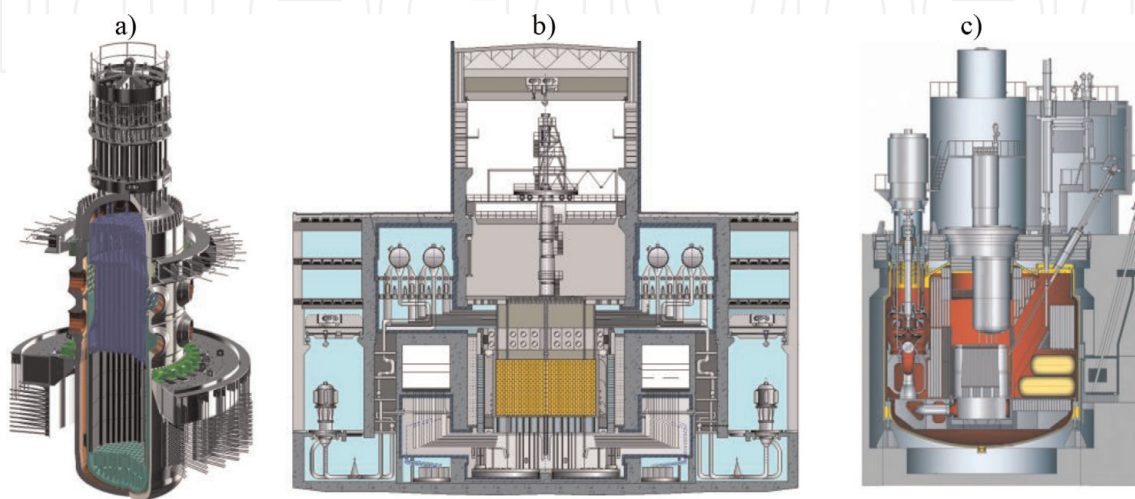

Figure 1.
The Russian reactors of VVER (water-moderated power reactor) (a), RBMK (channel graphite-moderated power reactor) (b) and BN (fast-neutron reactor) (c) types.

extremely conditions, a high-loaded power-generating plants with use physical and mathematical modeling [1, 12–17].

Results of traditional researches and a standardization of strength and life time of NPP in the determined statement in Russia and abroad are both initial scientific baseline of normative documents on design and actual baseline of making of perspective methods of a reliability estimate, survivability, initiation, and evolution of accidents and disasters by risk criteria, and also of makings of new principles, technologies, and engineering complexes ensuring safety service of NPP. These are conditions that are scientifically grounded to prevent initiation of the emergency and disastrous situations and also to minimize probable losses at their initiation at all stages of life cycle. Such situations within the limits of usual normative approaches and methods, as a rule, remained the least investigated from the scientific and application points of view owing to the complication, small predictability, and recurrence. At the same time, survivability of power-generating units in emergency situations and risk analysis of probable aftereffects should become weighable arguments in favor of building of nuclear units with a life expectancy from 60 to 100 years.

The analysis of sources, the reasons, and aftereffects of the heavy disasters occurring during installations of nuclear energetics display both their likeness and essential difference. Accidents known to the world on NPP with radioactivity ejection in a circumambient manner in the USA (the NPP "Three Mile Island (TMI)"—**Figure 2**), in the USSR (the Chernobyl NPP (CNPP)—**Figure 3**), and in Japan (the NPP "Fukushima-1—**Figure 4**) were the heaviest [3, 6, 8, 11, 18].

A common after effect of NPP accidents and disasters was that direct and indirect economical losses from them reached tens and hundreds of billions of USD. For their forestalling and preventing in the subsequent, the principal changes were made to designer, technological, and service solutions. Heavy emergency situations for NPP service arose earlier at the time of damage to their equipment, such as runners, steam plants, main coolant pumps, heat exchanger pipes, gate valves, and legs of reactor internals [11, 17].

The abovementioned NPP heavy accidents and disasters originated from unapproved impacts of human controllers, non-observance of technological discipline at emergency situation (TMI, CNPP), heavy-lift seismic loads, and a tsunami (Fukushima-1). Regular systems of the automatize guard of the NPP have been unreasonably disconnected (CNPP) or could not work in an emergency situation (TMI, Fukushima-1). Heavy emergency situations on turbine runners, steam plants,
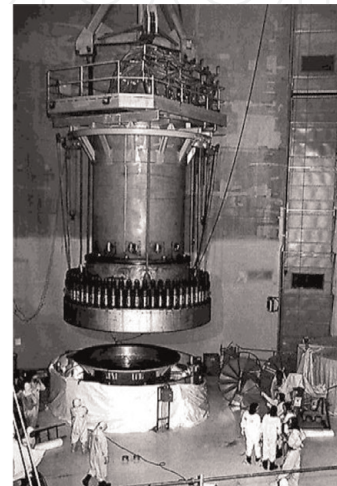
Figure 2.
The "Three Mile Island" NPP (TMI).

Figure 3.
The accident on the Chernobyl NPP (CNPP).

Figure 4.
The accident on the "Fukushima-1" NPP.

gate valves, and legs arose due to the lack of suitable technical diagnostics of these situations [11, 19], when faults in the form of cracks because of technological or operational fault attained of the limiting, intolerable sizes (10^2 to 1.5×10^3 mm), affecting 50–70% of carrying cross-section and creating sharp magnification of runner chattering. Thus, the analysis of such situations was not envisioned by normative calculations.

2. Combined researches of strength and life time

For installations of a nuclear energy in our country and abroad in the second half of twentieth century, the whole complex of fundamental and application developments [1–7, 11–14, 20–23] on the creation of normative strength calculations of the equipment and pipelines for nuclear power plants has been executed. Thus in our country special meaning had the solution of policy-making bodies that the scientific adviser of research developments on a justification of norms had been defined the Academy of Sciences of the USSR (The A.A. Blagonravov Institute for Machine Sciences—the IMASH), and the head development engineer of norms—the Ministry of medium machine building of the USSR (The N.A. Dollezhal Research and Development Institute of Power Engineering—NIKIET).

The same organizations making all prototype models of reactors for the NPP established the total statement about the strength before starting a reactor in service. Such norms developed both in the USSR [1, 12] and in the USA [14] subsequently were developed according to international standards set by the International Atomic Energy Agency—IAEA [13]. Compared to home norms of an NPP design [1, 12], basic sections on calculations, monitoring, probability safety assessment, and a justification of life time extension have been included.
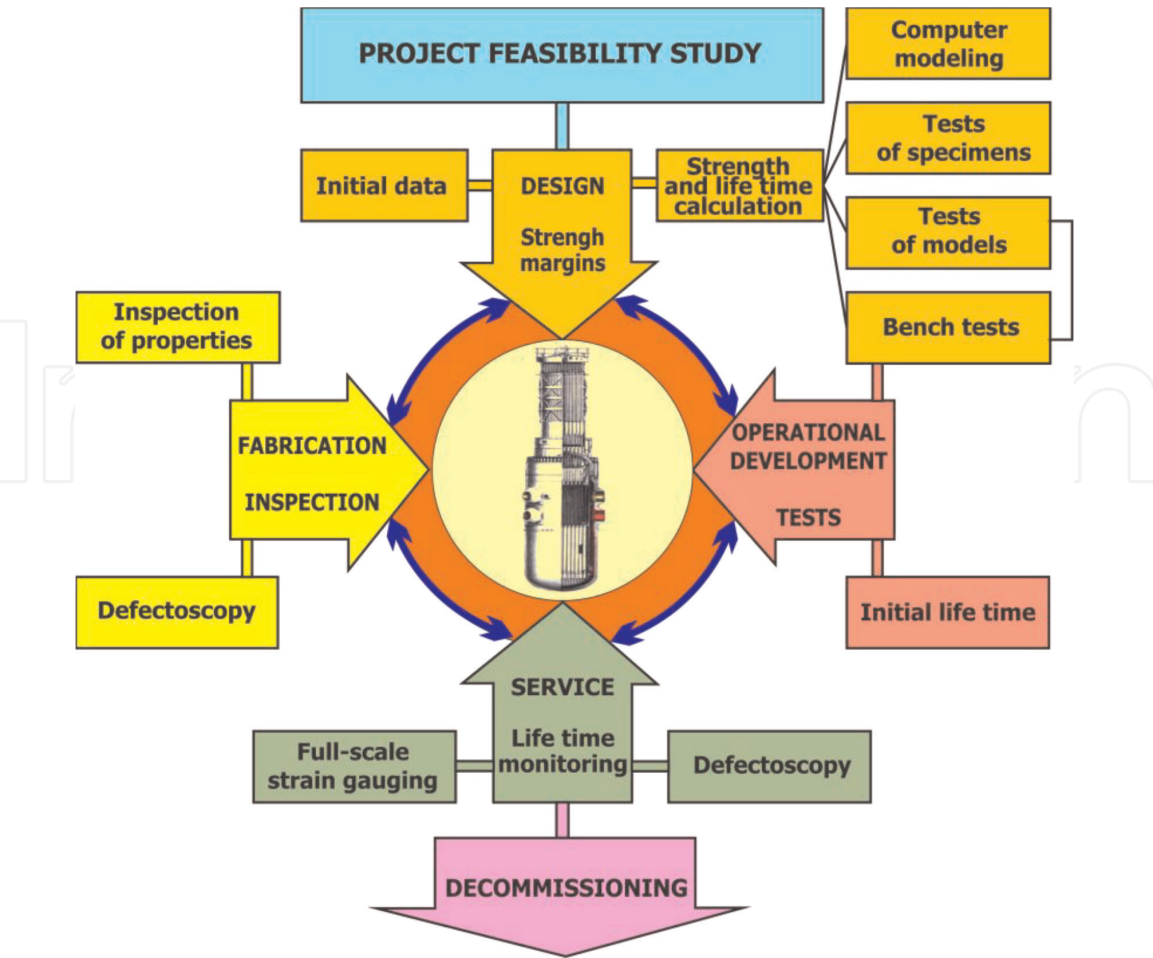
Figure 5.
The structure of the main task solution at making and service of the NPP equipment.

Long-term experience of home nuclear branch organizations and the academic institutes has allowed to form (**Figure 5**) the schematic diagram of the combined solution of tasks in view:

- The determined and statistical researches of deformation and fracture processes of laboratory specimens (with groups from 3–10 to 100–200 specimens of one steel)

- Model tests of the metallic specimens imitating most important parts (for example, studs of threaded connections with a diameter from 24 to 110 mm) and also nonmetallic specimens of studs with a diameter from 60 to 210 mm

- Tests of the modeling reactor vessels fabricated of nonmetallic materials in scale 1:10 and from metallic materials in scale 1:5

- Full-scale prestarting and starting tests of reactor prototype models of VVER, RBMK, and BN types

In considered norms, there are two cores sections: calculation of principal dimensions predominantly by criteria of a static strength and the verification calculations on a different combination of limiting states at low-cycle and high-cycle, long-term, vibration, seismic loads with initiation of static, cyclic, brittle, corrosion fracture, and also cyclic forming and radiation damage.
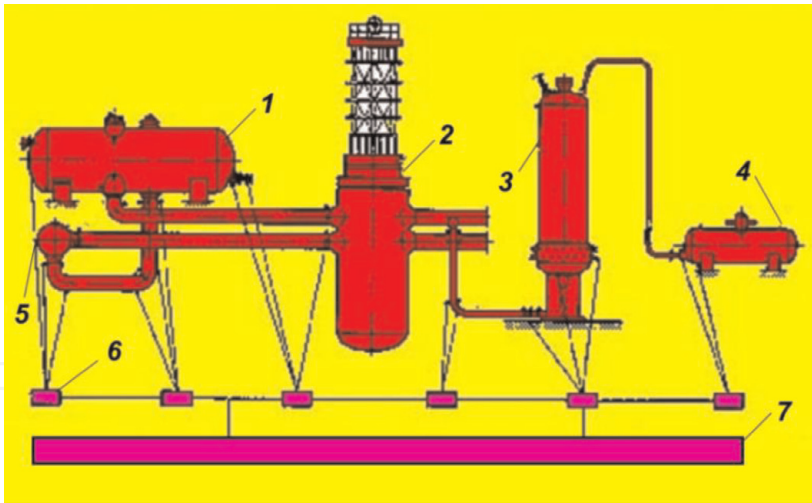
Figure 6.
The flow chart of strain-gauging of power equipment: 1—steam generator, 2—reactor, 3—pressure compensator, 4—bubble tank, 5—a main coolant pump, 6—commutators, and 7—registering apparatuses.

In the capacity of the most responsible and dangerous NPP components, nuclear reactor vessels, pipelines, pumps, steam generators, reactors, and machine halls have been accepted (**Figure 6**).

In an NPP with water-moderated power reactors (VVER) in the capacity of the major critical parts, it is possible to consider also the basic attachment fittings of reactor covers such as studs. Thus, the computational-experimental analysis of stress-strain states, strengths, and life times of a connection joint of reactor covers is conducted by improved methods in more detail (**Figure 7**).

For reactor installations of home production, such analysis was fulfilled [2–4, 11, 15, 16] jointly by the academic institutes, head branch research, and designer organizations on all prototype models of reactors in our country and abroad (Bulgaria, Finland, Hungary, Czech, and China) with application of the foremost methods: model researches of covers, studs, pressing rings on models from stress-optical and metallic materials, full-scale researches on reactors at preoperational tests on all regimes (including emergency), and also at an initial stage (till 1–3 years) of service.

In particular, the fifth unit of the Kozloduy NPP (Bulgaria) has been developed and implemented [15] after a most complicated program of full-scale researches by methods of a strain measurement, a thermometry, a vibrometry for all components of a primary loop with 1000 measuring points of local stresses, pressure pulsations, and temperatures (**Figure 8**).
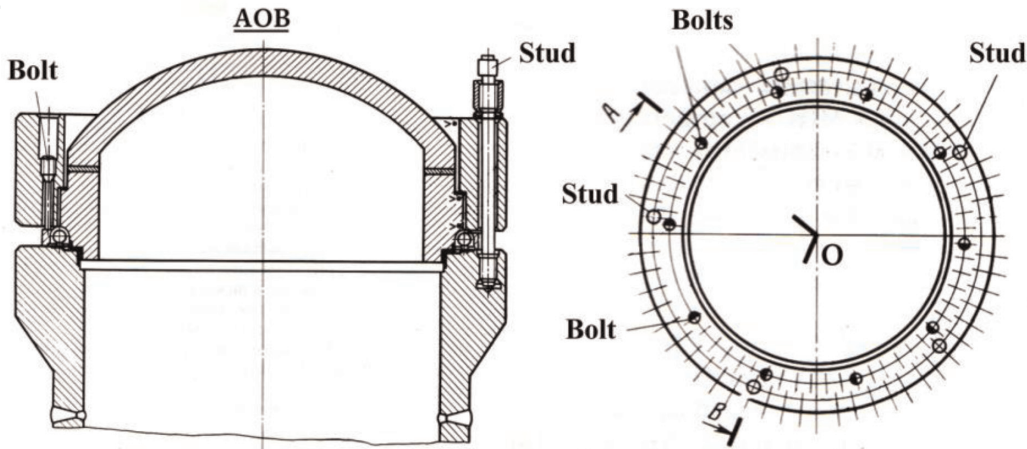

Figure 7.
The scheme of strain-gauging of a threaded connection of an attachment fitting of a cover.
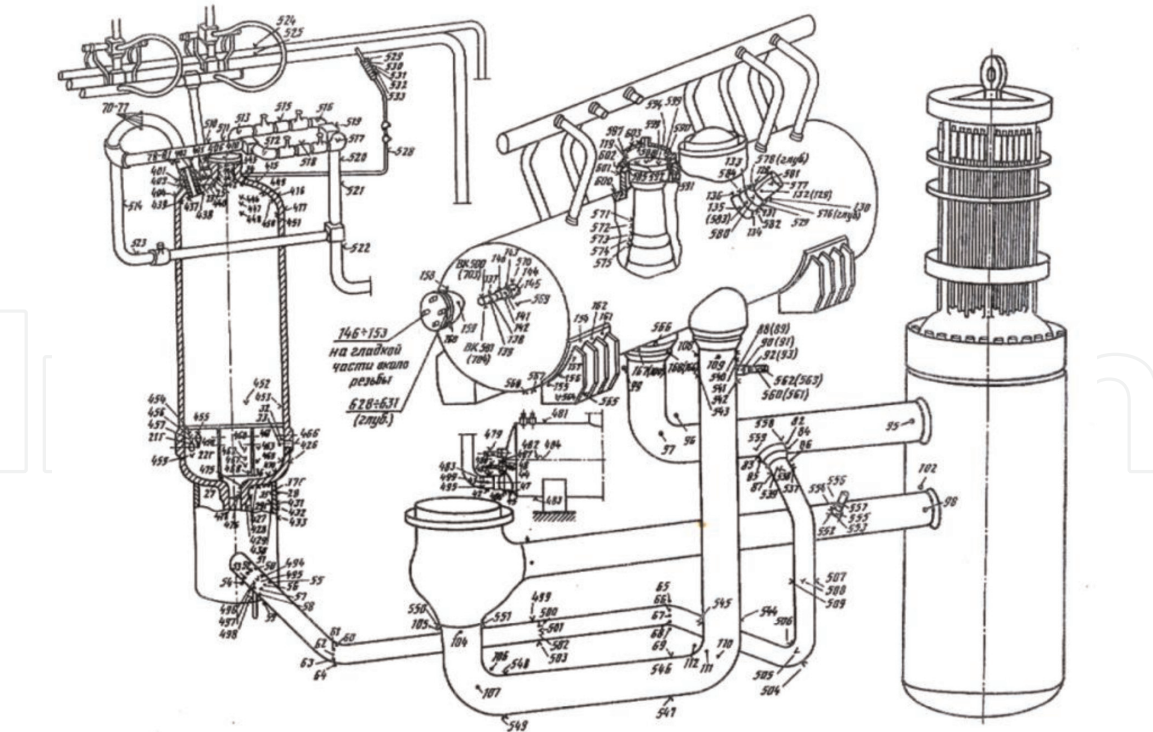
Figure 8.
Zones and points of placing of measuring gauges on the NPP equipment.

Modeling and full-scale researches have allowed to define detailed stress distributions on threads (**Figure 9**) and in a cover (**Figure 10**). These facts have given the chance to obtain real history of service impacts and nominal and local stresses on all parts of a reactor main joint.
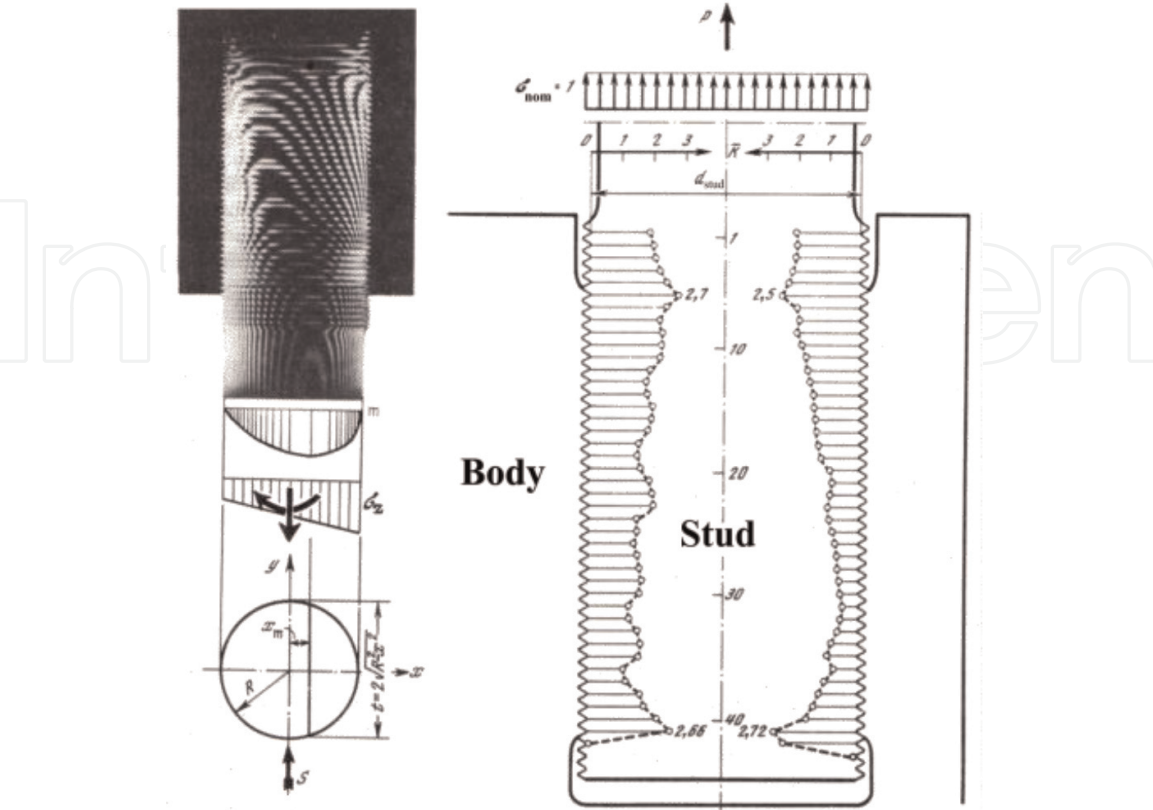

Figure 9.
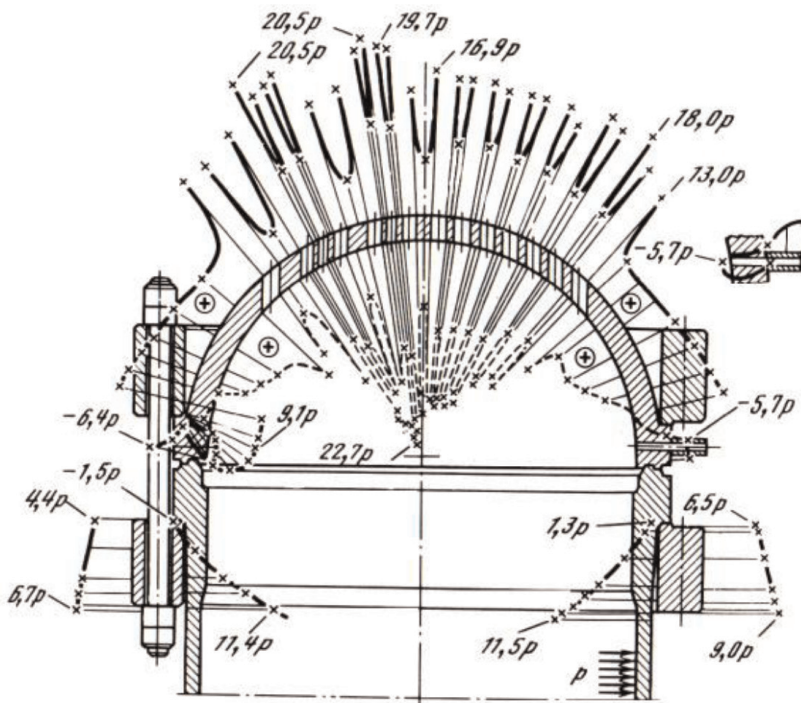*A stress loading of a stud attachment fitting of a reactor cover.*

Figure 10.
Stress distribution diagrams in a cover, flanges, and studs.

Computational and special experimental test bench researches of a dynamic stress loading and cyclical damages from seismic loads had a particular actuality.

On metallic modeling studs with a diameter from M12 to M110, data about life time on the basis of 10^4–10^5 cycles have been obtained. These data have allowed to justify improved margins on strength and life time of analyzed studs.

The principal great value in results these researches had that facts that the maximum accumulated damages (to 70%) arose in regimes multiple tightening and seal failure of caps (**Figure 11**). This fact has demanded work on special activities to decrease the indicated damages [15, 16].

Formation of development trends at the standardization instituting serviceability and safety of a nuclear (power-generating equipment went in a direction of specification and complicating of applied methods and criteria [1–3, 11, 20–23]. Thus, accidents and disasters (the TMI in the USA, the CNPP in the USSR, and
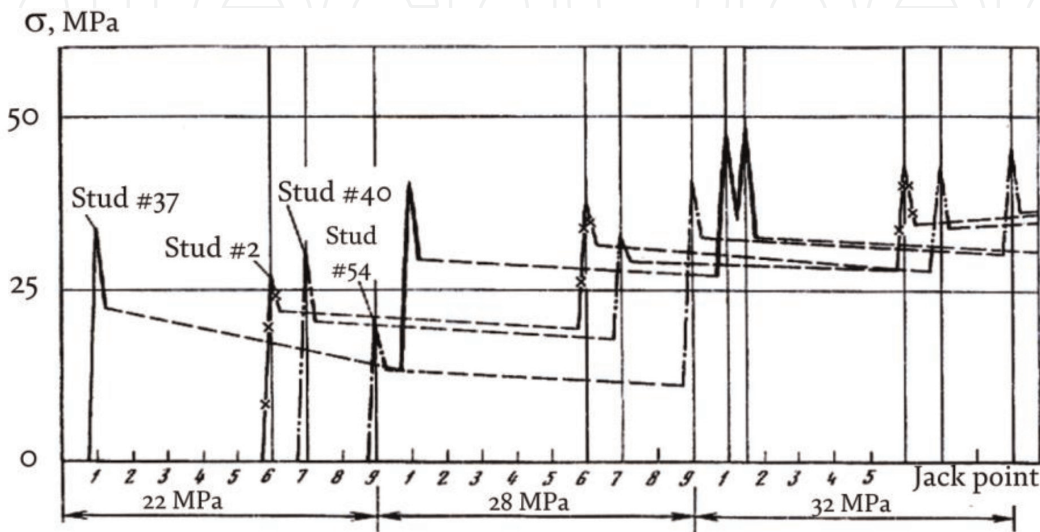

Figure 11.
The diagram of stresses change in studs at sealing of the main joint of the VVER-1000 reactor.

the Fukushima-1 in Japan) added additional information baseline for such development.

To the traditional solution of a problem of service safety [2, 6–10, 20–25], three groups of approaches had a direct ratio:

- From the position of strengths (in its multicriteria expression)

- From the position of life time (in time and cyclic statement)

- From the position of inadmissibility of large plastic strains

Traditional methods of strength justification were founded on a complex of determined characteristics of mechanical properties of materials and fracture criteria (yield point—σ_y, ultimate strength—σ_u, fatigue limit—σ_{-1}, and long-term strength—σ_{lt}). On the basis of these parameters of strength and fracture (present in standard and technical specifications for reactor structural materials), the status of safety and life time margins (n_σ, n_N, n_τ) has been generated. These margins are included in the reference, educational, and standard literature [1, 2, 12, 20–26]. Today, a common system of criteria and strength margins guaranteeing a fracture of nonadmission for equipment components at observance of the given service conditions is developed.

Mathematical modeling at the determined normative requirements to strength and life time came down to two approaches:

- To modeling parts of rods, plates, and thin shell types on the basis of analytical solutions of the theory of a strength of materials and theory of elasticity

- To modeling real objects on the basis of numerical solutions by finite-element method, finite difference method, and integral equations method

Research of seismic impacts was the most complicated at computational and experimental modeling:

- By finite-element method (FEM) for all parts of the first circuit (**Figure 12**)

- By methods of physical modeling of a reactor with reactor internals (**Figure 13**)

It has thus appeared that most high stresses and damages from seismic loads occur at the zone of attaching of pipelines to a reactor vessel.

On the basis of such modeling, nominal σ_n and maximum local σ_{max} stresses in concentration zones were defined. However, in these traditional approaches, normative materials often did not contain the direct data quantitatively instituting strength and life time of considered objects taking into account a statistical property of parameters σ_y, σ_u, σ_{-1}, and σ_{lt}. Occurring actually dissipation of parameters for strength calculation and life time of a NPP environment is caused by instability of manufacturing procedures at production of structural materials and NPP bearing parts (reactor vessels, pipelines, pumps, and heat exchangers). In the last decades, this deficiency has been eliminated, and the sphere of the traditional analysis of serviceability of the NPP equipment includes the theory and criteria of life time and reliability [2, 20–27].

In addition to normative calculations of reactors on [1] at the complicated regimes (**Figure 14**) of an assembly, test and service loading (assembly, a
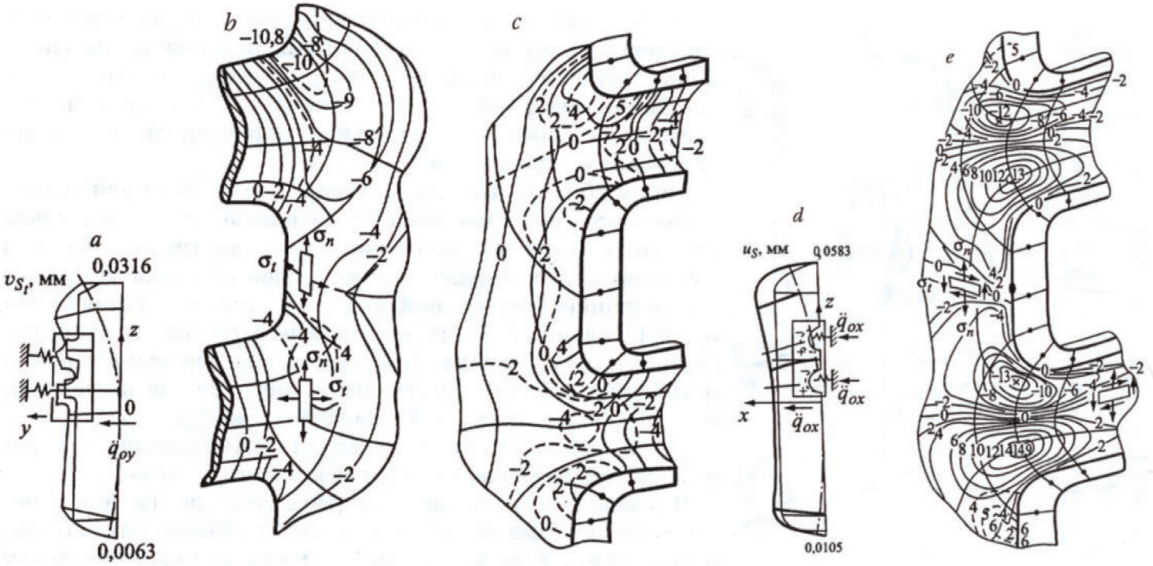
Figure 12.
Loads and stresses in a connecting pipes zone of a reactor vessel at seismic impacts for YOZ plain—the computational scheme (a), response stresses (MPa) on outside (b) and interior (c) surfaces; for XOZ plain—the calculation scheme (d) and stresses (e) on an interior surface.
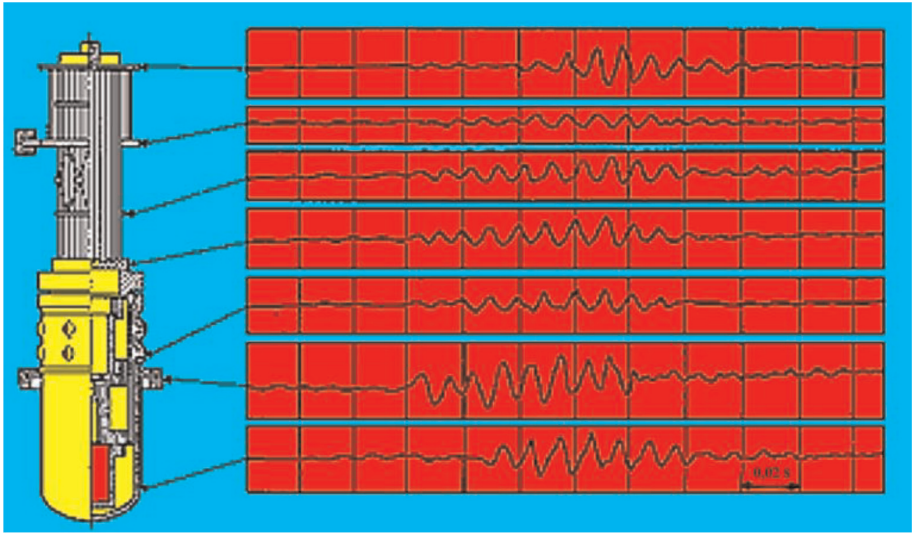

Figure 13.
A research of a dynamic state of a reactor simulator at seismic excitation.

tightening of studs, a hydroshaping testing, launch, capacity change, emergency operations, and shut-down) for events of occurrence of high levels of stresses improved strength, and life time calculations were carried out on the equations type

$$e_a = \frac{1}{(4N)^{m_e} + \frac{1+r_e}{1-r_e}} \cdot \ln \frac{1}{1-\psi_c} + 0,43 \frac{\sigma_b(1+\psi_c)}{E \cdot N^{m_\sigma}\left(1+\frac{1+r_\sigma}{1-r_\sigma}\right)}, \qquad (1)$$

where e_a is the amplitude of strain at a design regime; N is the life time at a crack initiation stage, in cycles; σ_b is the ultimate strength of a material ($400 \le \sigma_b \le 950$ MPa); ψ_c is the reduction of area in a neck of a specimen at single-pass rupture ($0.3 \le \psi_c \le 0.7$); r_e, r_σ are the cycle asymmetry parameters on strains and stresses, accordingly; and m_e, m_σ are the characteristics of a real material ($0.5 \le m_e \le 0.6$), ($0.08 \le m_\sigma \le 0.12$). Values of parameters in Eq. (1) $e_a, \psi_c, r_e, r_\sigma, m_e,$ and m_σ are relative and dimensionless.
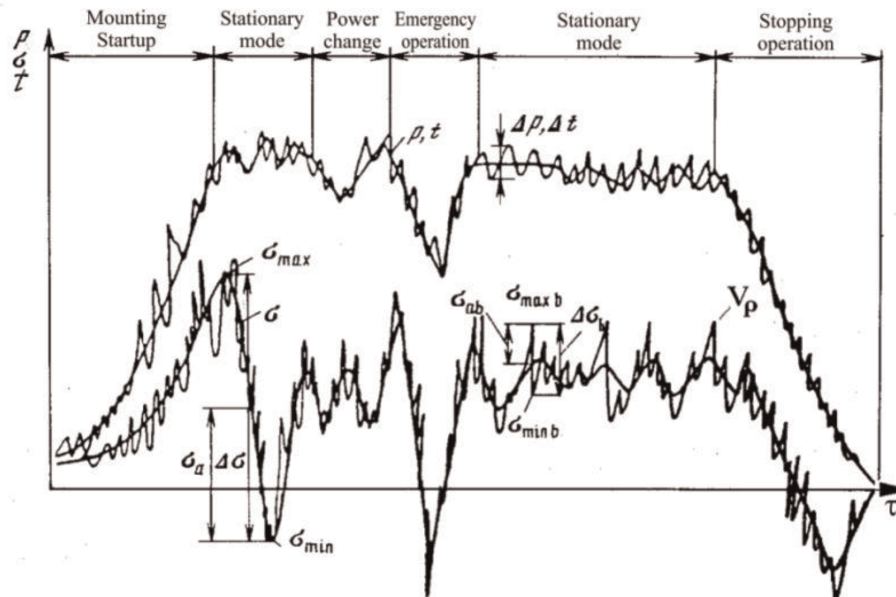
Figure 14.
The diagram of change of service loading parameters.

Calculation on Eq. (1) with the use of deformation criteria can be brought together to calculate by force criteria (on stresses) to accept $\sigma_a^* = e_a \cdot E$ (E—a modulus of elasticity).

Equation (1) is true for a wide band of life times ($10^0 \leq N \leq 10^{12}$). Permissible regimes of a stress loading are established in Eq. (1) with introduction of two margins n_{σ_a} and n_N. Then, the computational curve of permissible values $[e_a]$ (or $[\sigma_a^*]$) and $[N]$ is accepted as lower enveloping curves on each of these margins.

For the complicated regimes of a two-frequency loading (low-frequency with frequency $f_l = f_0$ (hertz) and amplitude of stress $\sigma_{al}^* = \sigma_{a0}^*$ (MPa), and high-frequency with f_h (hertz) and σ_{ah}^* (MPa), accordingly) on the basis of generalization of experimental data, life time decrease from the number of cycles of basic loading N_0 (cycle) to two-frequency life time N_2 (cycle) is considered [1, 28] in equation

$$N_2 = N_0/\chi; \chi = \left(\frac{f_h}{f_0}\right)^{\eta\frac{\sigma_{a0}^*}{\sigma_{ah}^*}},$$ (2)

where χ and η are dimensionless characteristics of a material and parameters of a two-frequency regime.

The same approach is used to calculate life time taking into account the presence of contact (wear resistance) and seismic impacts.

The presence of initial or service defects of cracks type with depth l is reflected in calculations of survivability on the basis of the equations of linear and a nonlinear fracture mechanics by change of stresses K_I (MPa·m$^{1/2}$) and strains K_{Ie} intensity factors [2, 20, 29]. For one-time brittle or a ductile fracture,

$$K_I = \sigma\sqrt{\pi l} \cdot f_\kappa \leq \frac{K_{Ic}}{n_{K_\sigma}}; K_{Ie} \leq \frac{K_{Iec}}{n_{K_e}},$$ (3)

where K_{Ic} and K_{Iec} are the critical (fracture) stresses and strains intensity factors, accordingly; n_{K_σ} and n_{K_e} are the dimensionless margins on stresses and strains intensity factors, accordingly ($n_{K_\sigma} \leq n_{K_e}$).

Reliability of equipment $P_{QR}(\tau)$ along with the account of the probabilistic approach to estimations of mechanical properties of a structural material is defined

also (**Figure 15**) on probabilistic characteristics of service stress loading $Q(\tau)$ and life time $R_{N\tau}(\tau)$ on the basis of distribution functions f of service impacts $Q^s(\tau)$ and of ultimate loads $Q_c(\tau)$ for the given life times N_c, τ_c. Thus, usually "trees of events" and "trees of failures" on experience of previous service of analogous technosphere objects are used. In such statement, the risk can be defined as

$$R(\tau) = 1 - P_{QR}(\tau). \tag{4}$$

More oriented on the quantitative solution of a safety problem for complicated NPP installations, capable to cause severe accidents and disasters, are new methods and criteria of the following groups [2, 6–8, 11, 18–21, 24–26, 29–33]:

- Survivability (ability and steadiness of operation at occurrence of damages at different stages of accidents and disaster evolution)

- Safety (taking into account the risk criteria and characteristics of accidents and disasters)

- Risk (in probability-economic statement)

From the above-stated, the up-to-date justification of strength, life time, reliability, survivability, safety, and risks (**Figure 16**) should be based on results of corresponding calculations and tests with observance of the special and new requirements established by corresponding normative-legal documents.

For long-term operated high-risk installations of a nuclear energetic to which the NPPs with reactors of the VVER concern, the BN and the RBMK types' rate, initial parameters of strength, life time, risk, and safety were defined in an explicit and implicit kinds on stages of their design and commissioning on acting then norms and rules which place at the different displayed in **Figure 16** footsteps (on time and analysis level).

Thereupon, during estimations of their state, two scientific and application approaches are possible:

- To realize stage by stage an estimation of the initial, exhausted, and remaining life time

- To estimate current life time, as initial for the given level of the service damage that has been accumulated in the previous operating period
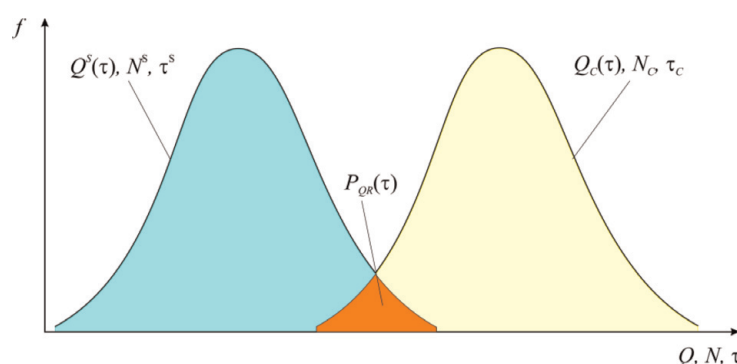

Figure 15.
The scheme of determination of reliability, failures, accidents, and disaster probability $P_{QR}(\tau)$.
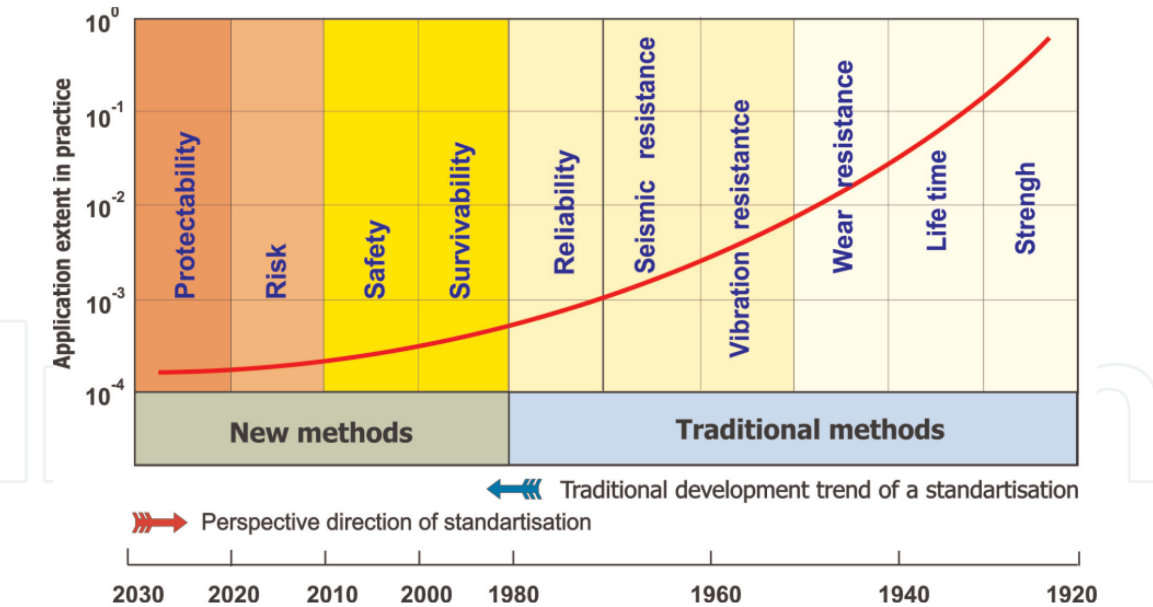
Figure 16.
A structure and evolution of standardization methods on the determined justification of strength and life time with the use of physical modeling of materials behavior at a static, cyclic, and long-term loading.

At the present time, the first approach was found to be the largest application. However, subsequently, the second approach appears to be deciding owing to its higher precision at estimations of the remaining strength, life time, and safety.

3. An estimation of risks and service safety

On the basis of the normative documents developed and accepted to present safety of power engineering as a whole, and NPPs in particular, the level of individual risks and risks of a possibility of accidents and disaster initiation should be estimated. In the process of perfecting NPPs and their nuclear reactors, these risks were reduced and will be reduced from 10^{-4} to 10^{-8} 1/year and less. For example, the reactor of natural safety with plumbeous heat-transfer agent will have a probability of fracture considerably below 10^{-8} 1/year [8, 11]. Individual risks of nonnuclear power engineering lay within the limits 10^{-4}–10^{-7} 1/year (**Table 1**).

The great importance for the analysis, support, and improvement of safety of the considered equipment within the limits of dominating and active concepts,

Types of the electric power manufacture	Risk estimation on the person in a year
Hydropower plant	10^{-5}-10^{-6}
Solar power plants	10^{-4}-10^{-5}
Wind power plants	10^{-4}-10^{-5}
Thermal power plants	10^{-4}-10^{-5}
Nuclear power plant	10^{-4}-10^{-8}
Reactors of the first generation	10^{-4}-10^{-5}
Reactors of the second generation	10^{-6}
Reactors of the third generation	10^{-7}
Reactors of the fourth generation	10^{-8}
Perspective reactors of the fifth generation	$<10^{-8}$

Table 1.
Comparative data about a radiation-ecological risk for different directions of the electric power manufacture.

N	Analyzed situations	Protectability	Risks type
1.	Service standard conditions	Heightened	Controlled
2.	Shifts from standard conditions	Sufficient	Adjustabled
3.	Design accidents	Partial	Analysed
4.	Out of design accidents	Insufficient	Heightened
5.	Hypothetical accidents	Low	High

Table 2.
Types extreme (emergency and catastrophic) situations and level of protectability from them of high-risk installations.

strategies, norms, orders, and margins has the level of a scientific-practical justification of the predictable and acceptable risks characterizing generally regular and limiting states of these installations.

For all spectrum of technosphere installation types of emergency and catastrophic situations, the level of their protectability and types of accompanying risks at transition from standard conditions operation in regular states to emergency and catastrophic at service can be described (**Table 2**) as:

- Regular situations—occurring at installations operation in the breaking points established by norms and rules; risks for them controlled; and protectability from them increased

- Regime emergency situations—occurring at a shift from service standard conditions at regular operation of potentially dangerous installations; aftereffects from them predicted, risks for them controlled; and protectability from them sufficient

- Design emergency situations—arise at a runout of installation out of breaking points of regular regimes with predicted and acceptable aftereffects; risks for them analyzed; and protectability from them partial

- Out-of-design emergency situations—arise at nonreversible damages of important parts of installation with high losses and human sacrifices and with necessity of carrying out a recovery work; risks for them heightened; and the level of protectability from them insufficient

- Hypothetical emergency situations—can arise at the not forecast in advance scenarios of evolution with the greatest possible losses and sacrifices; are characterized by high risks; protectability from them low; and restoration of installations is impossible

The complex calculation-experimental analysis of the initial and remaining service life of an NPP is founded first of all on an estimation of service damages accumulation conditions at different service regimes taking into account corresponding state equations, and also on the study of conditions of transition in limiting states taking into account service kinetics of mechanical properties of materials, criteria of strength, crack resistance, and survivability.

Generally termed procedures are implemented with the use of a complex criteria equations, computational equations, and design parameters applied to the analysis and definition of regular and limiting states of engineering objects. The complex criteria include the following equations:

For an estimation of static and long-term strength,

$$F_Q\{\,\sigma, e, t, \tau\} = F_Q\left\{ f_1\left(\frac{\sigma_y^\tau}{n_y}, \frac{\sigma_u^\tau}{n_u}, \frac{\sigma_{lt}^\tau}{n_\sigma}, \frac{e_c^\tau}{n_e}, \frac{\tau_c}{n_\tau}\right), f_2(m)\right\}, \tag{5}$$

where F_Q is the functional characterizing dependence of stresses from actual force impacts Q; σ, e are the operating in time τ at temperature t stresses and strains; f_1 is the functional dependence, which includes σ_y^τ, σ_u^τ, and σ_{lt}^τ that are the yield, strength, and long-term stress points of a material for deformation time τ, accordingly, e_c^τ, e_c^τ is the critical values (at fracture) of strains at this time; n_y, n_u, and n_σ, n_e, n_τ are the margins accordingly on yield and strength stress points, on stresses, strains, and time; $f_2(m)$ is the functional dependence (in most cases, power) for a hardening parameter m in elastoplastic field of a deformation [2, 20, 21].

Для оценки ресурса по параметрам числа N циклов и времени τ

$$F_L\{\,\sigma, e, N, \tau\} = \left\{ f_1\left(\frac{\sigma_a}{n_\sigma}, \frac{e_a}{n_e}, \frac{N_f}{n_N}\right) f_2(\sigma_y, \psi_c, m_p, m_e)\right\}, \tag{6}$$

where F_L is the functional characterizing dependence of life time from amplitudes of stresses σ_a, strains e_a, number of fracture cycles N_f, and margins corresponding to them, and from plasticity of material ψ_c (the relative cross throat at fracture) and exponents for an equation of a fatigue curve for plastic m_p and elastic m_e components of cyclic strains e_a [2, 7, 20, 21].

For a crack resistance estimation,

$$F_K\{\,\sigma, e, K_I, K_{Ie}, \tau, t\} = F_K\left\{ \frac{\sigma}{n_\sigma}, \frac{e}{n_e}, \frac{K_{Ic}}{n_K}, \frac{K_{Iec}}{n_{Ke}}, \frac{\tau_c}{n_\tau}, \frac{t_c}{n_t}\right\}, \tag{7}$$

where F_K is the functional characterizing dependence of stresses K_I and strains K_{Ie} intensity factors, from their critical values K_{Ic} and K_{Iec}, from stresses σ and strains e levels, from critical time to fracture τ_c and critical temperature t_c with corresponding margins [2, 7, 20, 21].

For a survivability estimation,

$$F_{L_{ld}}\{\sigma, e, l, N, \tau, K_I, K_{Ie}\} = F_{L_{ld}}\left\{ (\Delta K_I, \Delta K_{Ie}), \left(\frac{dl}{dN}, \frac{dl}{d\tau}\right)\right\}, \tag{8}$$

where $F_{L_{ld}}$ is the functional characterizing dependence of survivability parameter from values of service stresses and strains, causing material damage d, from sizes of faults (cracks) l, from crack growth rates on number of cycles dl/dN, and time $dl/d\tau$ parameters, and also from values of ranges of stresses K_I and strains K_{Ie} intensity factors [2, 7, 20, 21].

For a risk and safety estimation,

$$F_R\{P(\tau), U(\tau)\} = R(\tau); \tag{9}$$

$$F_S\{R(\tau), n_R\} = S(\tau) \le \frac{1}{n_R} R_c(\tau) = [R(\tau)] = F_M\{R_c(\tau), n_R, M(\tau), m_M\}, \tag{10}$$

where F_R is the functional, characterizing risk $R(\tau)$ as analytical dependence of probability $P(\tau)$ of occurrence on installation of an emergency situation of this or that type and probable loss $U(\tau)$ in case of its implementation; F_S is the functional characterizing parameter of safety $S(\tau)$, which bundles parameters of really occurring risk with its critical $R_c(\tau)$ (limiting) and admitted $[R(\tau)]$ (acceptable) values through margin factor on risk n_R defined in advance.

Thus, the level of installation safety functionally (F_M) depends on values of critical risk, from margin on a risk n_R, and also from costs $M(\tau)$ of carrying out steps to decrease danger (risk) of installation and from effectiveness factor of these costs m_M [8, 18, 24].

The mentioned complex functional criteria in Eqs. (1)–(10) allow to implement the full sequence of installation calculation for the purpose of providing for its service safety, beginning from strength parameters and completing at protectability parameters with acceptable values of risk both on a design stage, and at concrete stages of service, including a decision made about life time extension.

At an estimation of the remaining life time on resistance to cyclic fracture, levels of cyclical stresses, cycle asymmetry parameters, a stress concentration, cyclical properties of a material, service temperatures, special conditions of loading, and residual stresses and strains are subject to analysis. Under these data calculation processes and parameters of impacts, fracture stresses and life time are defined. On the basis of such definition are the functionals that resulted above in Eqs. (4)–(10), which include calculation dependences (state equations, curve of deformations and fractures, and strain and force criteria). In improved calculation zones of welded joints, a plastic deformation in the most loaded zones, variety of operating conditions and impacts, and dispersion of characteristics of mechanical properties [2, 10, 20–29, 31, 34–36] are considered.

As appears from Eqs. (1)–(10) the computational-experimental justification of static, long-term, and cyclic strength, life time, and risks included in comprehensive analysis of conditions of safety service of the NPP equipment at regular and unnominal situations, sampling of types of limiting states, calculation schemes and calculation cases, methods of the analysis of stress-strain states, methods of preliminary diagnostics of technical state, assignment of margins on strength and on life times, study of probabilities of limiting states reaching, an estimation of risks of accidents and disasters [2, 9–11, 20–36].

The built-up calculation of curve (permissible amplitudes of stresses and life time at a cyclic loading, and also of the maximum stresses and time before fracture in the long term) is carried out for an estimation of initial and remaining life time on the basis of a schematization of history of loading, sampling of computational schemes, and computational cases. The calculation of initial and remaining life time is carried out in two alternatives: an approximate calculation and improved calculation.

The concept of an estimation, a diagnosis, and a prediction of service life of the NPP is correlated with the sampling of state variables of the equipment on the level of wearing and life time exhaustion. To define the factors and parameters influencing on life time, it is necessary to attribute maximum deviations of wall width and errors in measurement, a staging of prediction of life time, results of resource and strength researches, levels of diagnosing of installations, and influence of engineering preliminary diagnostics efficiency on the level of a fracture risk.

On the basis of summarizing of results of a life time design justification of reactors, it is possible to establish a dependence of life time on commissioning terms, for example, an NPP with VVER type reactor of all generations (**Figure 17**). To a twenty-first century kickoff in our country and abroad, the design life time (expected life) has increased to 40–60 years; by 2025, the design life time can increase to 100 years [1, 3, 7, 11, 24].

Thus, the key problems of design, manufacture, service, upgrading, and a leading-out from service of nuclear units of the following (the fourth and the fifth) generations with heightened characteristics of life time and safety are:

- Transition to new principles of reactor core build-up, sharply reducing severe accident possibility with its melting

- Use of joint guard from severe accidents by new organization of working master schedules both in regular and in the emergency situations promoting to decrease of negative and dangerous aftereffects of accident propagation

- Introduction in practice of making and service of reactors with an in-depth analysis of risks of occurrence and propagation of the emergency and catastrophic situations, considering both probabilities of these situations and their aftereffects

- Inclusion in the analysis of heightened life time, risks and safety of reactors of such base criteria as strength, life time, reliability, survivability, physical protectability, and economic justification

- Orientation to escalating requirements to safety of the NPP formed by national and international laws, norms, and rules

- Elimination of unreasonable conservatism in already accepted normative and technical documents and introduction in the safety analysis of new threats and risks (including risks of terrorism)

- Statement as the corner-stone fundamental and applied researches of safety of nuclear reactors of problems of forming of unified methodical baseline on integrated study of external and interior impacts of a wide spectrum, responses to these impacts of critical important bearing elements of the NPP in linear and nonlinear fields of a deformation, damages, and fractures

- Setting, justification, control, and monitoring of the major parameters of life time and safety of the NPP operation at regular and emergency situations for confinement of margins on strength, life time, and risks in safety breaking points

Problems of safety maintenance on the basis of the concept of risks generally should to be decided with the use of the determined, statistical, probability, and combined methods of fracture mechanics and mechanics of disasters. Probabilities P_S of realization in an NPP of system threats can be presented with the use of functional F_{PS} [2, 6, 8, 18, 24–26, 29, 32, 33]

$$P_S = F_{PS}\{P_N, P_T, P_O\}, \qquad (11)$$

where P_N is the probability of occurrence of the unfavorable event, stipulated by the human factor; P_T is the probability of such event stipulated by a state of an NPP components; and P_O is the probability of its occurrence stipulated by an environmental exposure.

The type of functional Eq. (11) remains the same and for probabilities of risks realization included in the analysis at design, making, and service of the NPP. The great importance thus has that facts that the role of the human factor in appraisal P_S at change P_N is defined not only human controllers and the personnel, their professional qualities and a physiological state, but the experts, making solutions on all level of the hierarchy by safety of the NPP.

Probabilities P_T essentially depend on the level of protectability of the NPP from accidents and disasters. This protectability is defined by quality of their initial and current state, extent of degradation of installations at the given stage of service, and diagnosing and monitoring level. Such position indicates direct interacting of
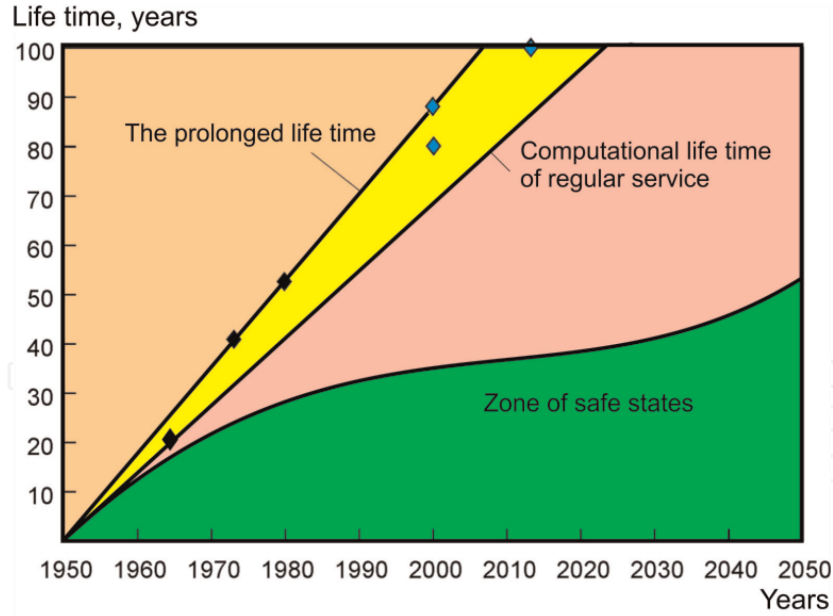
Figure 17.
Characteristics of initial design (full line) and the prolonged expected life (lives times) of the NPPs with type reactors VVER of the first–the fifth generations.

parameters P_T and P_N taking into account base parameters of reliability and quality of technosphere installations.

Probabilities P_S, as it is known, depend on occurrence of dangerous natural processes (earthquakes, floods, hurricanes, tsunami, landslides, etc.) and also from a state of the NPP installations and, hence, from P_T. Adoption unreasonable (from the point of view of risks) $R(\tau)$ solutions on arrangement of technosphere installations and zones of population residing does parameter P_S dependent and from P_N.

Losses U_S from realization of system threats generally can be recorded through the functional F_{US}

$$U_S = F_{US}\{U_N, U_T, U_O\}, \tag{12}$$

where U_N is the losses caused to the population at interacting of primary and secondary knocking factors at realization of strategic system threats; U_T is the losses caused to technosphere installations; and U_O is the losses caused to an environment.

Values U_N, U_T, and U_O can be measured both in natural units (for example, a death-roll of people, number of the blasted installations, and the square of injured territories) and in equivalents (for example, in economic, monetary parameters).

As a whole, in Russia, taking into account social and economic transformations, global processes to power supply and experience and prospects of nuclear energetics development based characteristics of risks R of accidents and disasters of the natural-technogenic character, defined by their losses U (or severity) and probability P (or quantity), have rather complicated character of a time history τ with a common trend to increment (**Figure 18**).

Accepting that the relative risks $\overline{R}(\tau)$ increase eventually owing to natural aging processes, degradation, accumulation of damages, and level of safety $\overline{S}(\tau)$ depends on the relative protectability $\overline{Z}(\tau)$.

$$\overline{R}(\tau) = F_R\{\overline{U}(\tau), \overline{P}(\tau)\}; \overline{S}(\tau) = F_S\{\overline{R}(\tau), \overline{Z}(\tau)\}, \tag{13}$$

where the fact of accident and disaster occurrence will correspond to the condition
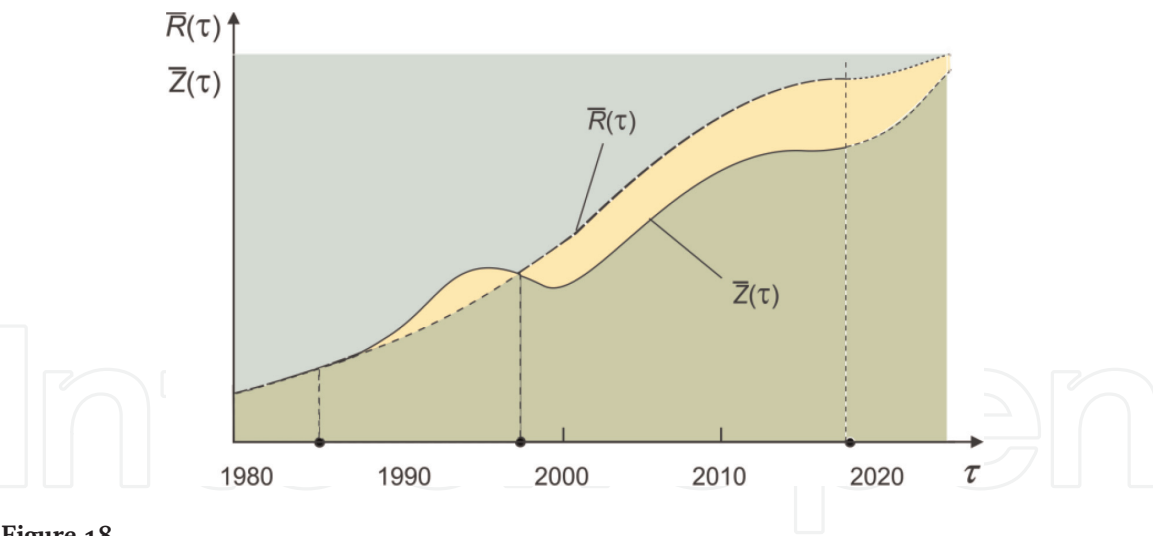
Figure 18.
The time history of the relative risk levels and protectability.

$$\overline{S}(\tau) = \overline{R}(\tau)/\overline{Z}(\tau) \leq 1. \tag{14}$$

Such conditions occurred at the moment of Chernobyl disaster (1986), last years the twentieth centuries at damages of collecting channels of steam generators PGV-1000 type, on boundary line of centuries at damages of welds to a weld zone of the principal circuital pipeline to the steam generator [4, 11].

In **Figure 18** the major role of improving of all service parameters of the NPP, and first of all life time and safety which promote decrease of probabilities of accidents and disasters occurrence $P(\tau)$ and accompanying them losses $U(\tau)$ is visible.

When for the equipment of the concrete NPP, the relative system risks $\overline{R}_S$ (for population $\overline{R}_N$, for technosphere installations $\overline{R}_T$, and for environment $\overline{R}_O$) are defined, the surface of limiting states on values of these system risks $\overline{R}_S$ varying on some random paths $V(R)$ can be plotted (**Figure 19**).

$$\overline{R}_S = \sqrt{\overline{R}_N^2 + \overline{R}_T^2 + \overline{R}_O^2}. \tag{15}$$
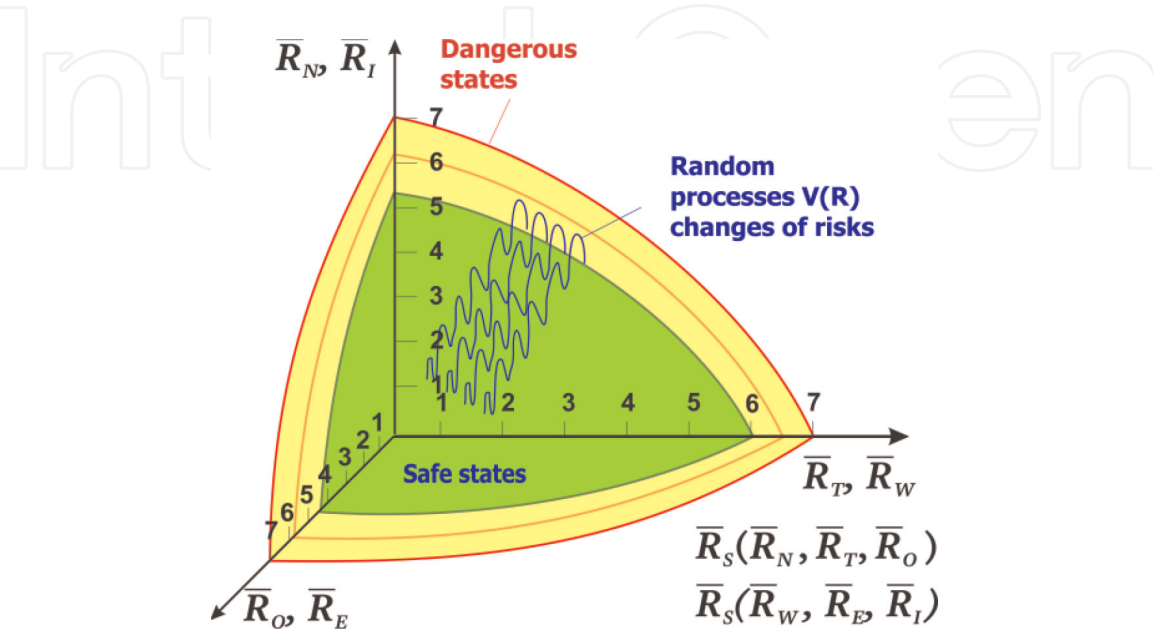

Figure 19.
The surfaces dangerous and safe states on values of risks.

19

To reach the acceptable protectability of the NPP equipment, implementation of complex steps on the decrease of system risks $\overline{R}_S$ is necessary.

If on axes $\overline{R}_T$, $\overline{R}_N$, and $\overline{R}_O$ to put aside classes from 1 to 7 for accidents and disasters on extent of increment of their severity (1—local, 2—object, 3—district, 4—regional, 5—national, 6—global, and 7—planetary), then the quantitative assessment of extent of the NPP safety and any of its components by criteria of risks is represented possible. Such estimation is given by the radius vector in three-dimensional space "$\overline{R}_T$-$\overline{R}_N$-$\overline{R}_O$". The strength and life time improvement on all stages of installation design, making, and service should promote decrease in danger of these installations.

For an NPP transfer in safe states with the use of risk criteria $\overline{R}_N$, $\overline{R}_T$, and $\overline{R}_O$ (**Figure 19**), it is necessary to reduce the possibility (risk $\overline{R}_S$) of uncontrollable emission of potentially dangerous substances W and energies E and also a loss of control (disruption of data flows I),

$$\overline{R}_S = \sqrt{\overline{R}_W^2 + \overline{R}_E^2 + \overline{R}_I^2}, \tag{16}$$

or to reduce the relative risks of accidents and disasters $\overline{R}_N$, $\overline{R}_T$, and $\overline{R}_O$ as in Eq. (15) and $\overline{R}_W$, $\overline{R}_E$, and $\overline{R}_I$ as in Eq. (16).

This result can be attained by the creation of monitoring systems for diagnostics and monitoring of risk parameters $\overline{R}_N$, $\overline{R}_T$, $\overline{R}_O$, $\overline{R}_W$, $\overline{R}_E$, and $\overline{R}_I$ and guard $Z(\tau)$, and also by the introduction in the analysis of safety $S(\tau)$ scenarios of occurrence and propagation of emergency and catastrophic situations.

The state, regional and object control, regulating and providing of safety $\overline{S}(\tau)$ by system risks criteria $\overline{R}_S(\tau)$ comes to the qualitative both quantitative statistical and determined analysis on the given interval of time $\Delta\tau$ of all service parameters and to implementation of complex activities on decrease of system risks from actual unacceptable R_S to acceptable (admissible) levels $[R_S]$:

$$\overline{R}_S = F_R\{\overline{P}_S, \overline{U}_S\} \le [\overline{R}_S] = (1/n_S) \cdot \overline{R}_{Sc} = F_R\{[\overline{P}_S], [\overline{U}_S]\} = F_M\{m_M{}^{-1}[\overline{M}]\}, \tag{17}$$

where n_S is the safety factor on system risks; $\overline{R}_{Sc}$ is the unacceptable (critical) risk; $[\overline{P}_S]$ and $[\overline{U}_S]$ are the acceptable (permissible) probabilities and losses; $[\overline{M}]$ is the necessary acceptable expenditures for decrease of risks; and m_M is the cost-effectiveness ratio ($1 \le m_M \le 10$).

Safety of the NPP by criteria of risks can be considered ensured if the inequality $n_S \ge 1$ is attained.

The interval of time $\Delta\tau$ for which risks $\overline{R}_S$ are defined usually is accepted to equal 1 year ($\Delta\tau$ = 1 year).

According to Eqs. (15) and (16), control and planning with the use of the criteria baseline grounded on risks come to following primal tasks:

To the development of scientifically well-founded methods of the analysis of risks $\overline{R}_S$ and their basic quantities $\overline{P}_S$ and $\overline{U}_S$

To decision making about the level of allowable values $[\overline{R}_S]$, $[\overline{P}_S]$, and $[\overline{U}_S]$ with an estimation of margin values n_s

To scientifically well-founded level of definition of necessary expenditures $[\overline{M}]$ on decreasing risks with sampling and improving of efficiency of these expenditures m_M

Thus, predicting, monitoring, and forestalling of accidents and disasters for an NPP (including by improving of all parameters of strength, life time and

survivability) appear to be essentially more effective, than liquidating of aftereffects of catastrophic situations (type of the TMI, the CNPP, and the Fukushima-1). Values $\overline{M}$ at a suitable justification of activities on the decrease of risks can be considerable (in m_M time) less losses $\overline{U}_S$ caused to economy by vulnerability of the equipment for all types of NPPs.

As it was already mentioned, safety of nuclear energy installations $S(\tau)$, as well as all other complicated engineering systems, on the given interval of time τ is defined in Eq. (13) by two basic quantities: probability $P(\tau)$ of unfavorable event occurrence (an unfavorable situation) and probable loss $U(\tau)$ from this event. Values $P(\tau)$ and $U(\tau)$ are generally statistically uncertain, demanding for their quantitative assessment of great volumes of the information on the nature, behaviors, sources, and scenarios of unfavorable events both for each of considered installations and for the given set of installations (group, batch, and series) at occurrence and propagation of unfavorable events and also the information on aftereffects for installations, persons, and an environment at occurrence, propagation, and liquidation of unfavorable events.

4. The analysis of limiting states

In nuclear energetics with reactors of all types and all generations (from the first to the fourth) prior to the beginning of the twenty-first century, at failure analysis, the basic attention was given to parameter $P(\tau)$ that defined reliability of safety operation of the NPP. Special meaning was added thus to the forestalling and prevention of the heaviest on the aftereffects of catastrophic situations with the peak damages—melting of the core and a radioactivity runout for breaking points of all guard barriers—casings of the fuel element, cartridge, reactor vessel, reactor hall, and containment. In this case, reactor vessel fracture is extremely dangerous. This event concerns the seventh group of limiting states.

Significant aftereffects arise also at fracture of the basic elements of the first circuit of a reactor vessel and collecting channels of steam generators, pumps, volume compensators, bubbler tanks, and also housings and runners of turbines in the second circuit. These fractures amount the sixth group of the limiting states creating threats to the population, the NPP, and the environment.

If while in service of the NPP because of occurrence of damages of parts of the first circuit has arisen a radioactivity outside breaking points of the NPP and there were thus threats of bombarding radiation for the population, then it is necessary to attribute these events to the fifth group of dangerous limiting states.

The leakages caused by partial damages (faults of crack type or depressurizations of connectors) and creating threats for human controllers and the personnel in the NPP concern the fourth group of limiting states.

The third group of limiting states should be bundled to the considerable damages of the above-termed parts of the first and the second circuit without a radioactivity runout for breaking points of an NPP, which are not demanding their mandatory substitution.

The second group of limiting states concern occurrence in bearing structures of the NPP of partial damages without a radioactivity runout for breaking points of the first circuit, not demanding their substitution, but demanding carrying out of repair-and-renewal operations.

The first group of limiting states is amounted by those of them which are bundled to damages and the faults that have fallen outside the limits admissible under inspection norms and calculation, but not demanding mandatory carrying

Group	Types of limiting state	Danger extent	Objects for threats
LS-7	Core damage. Fracture of a reactor vessel with a radioactivity runout in an environment	Extremely high	The population, NPP, environment
LS -6	Fracture of the basic parts of the first and the second circuits with a radioactivity runout in an environment	Excessively high	The population, NPP, environment
LS -5	Large leakages in the first circuit with a radioactivity runout for NPP breaking points	Very high	The population, NPP, environment
LS -4	Damages and leakages in the first and the second circuits with probable runout of a radioactivity for NPP breaking points	High	The population, NPP, environment
LS -3	Damage and leakages in the first and the second circuits with a runout of a radioactivity inside of the NPP	Heightened	Human controllers, NPP parts
LS -2	Partial damages without a radioactivity runout for the breaking points of the first circuit demanding reconditioning	Not high	The damaged equipment
LS -1	Partial damages without a runout of radioactivity which are not demanding reconditioning	Low	The damaged equipment

Table 3.
Groups of limiting states for the analysis of the NPP safety.

out of repair-and-renewal operations and that can be admitted to prolongation of service before the next examination.

These facts allow to execute summary classification by groups of limiting states for the NPP equipment (**Table 3**) from the most dangerous admissible (the seventh group of limiting states LS-7) to the least dangerous admissible (the first group of limiting states LS-1).

For the groups of limiting states indicated in **Table 3** taking into account summarizing of great volume of normative and technical materials and results of the executed researches, it is possible to describe demanded (admissible) probabilities $[P(\tau)]$ occurrence of unfavorable events. To such probabilities there correspond their actual levels obtained from statistics of their occurrence while in service of NPPs of all generations. Each severe accident or disaster on an NPP, happening at the moment τ_c, was accompanied by comprehensive analysis of their reasons and sources, and also realization of considerable on volumes and expenditures of activities for safety improving. Eventually, at $\tau_s > \tau_c$, after such accidents or disasters, decrease of probabilities from $P(\tau_c)$ to $P(\tau_s)$ was observed.

For values of probabilities $P(\tau_c)$ and $P(\tau_s)$ for all reactors operated in the world at $\tau \leq \tau_c$ and $\tau = \tau_s$, it is possible to estimate on ratios

$$P(\tau_c) = \frac{N_d}{N_{tc} \cdot \tau_c}; P(\tau_s) = \frac{N_d}{N_{ts} \cdot \tau_s}, \qquad (18)$$

where N_d is the quantity of the reactors that have obtained damages at the given i-th type of limiting state under **Table 3**; N_{tc} is the total of reactors to the time τ_c of occurrence of the given i-th type of damage; N_{ts} is the total of reactors to the time τ_s; τ_c is the mean time (years) of service of one reactor to the time of reaching of the given i-th type of limiting state; and τ_s is the mean time of the service of one reactor.

As it was already mentioned, unfavorable events on an NPP (disasters, accidents, failures, and disruptions) are accompanied by corresponding losses $U(\tau)$ both at the moment of occurrence of these events τ_c and after them ($\tau \geq \tau_c$). These losses are caused to the person (to human controllers, the personnel, and the population), to technosphere installations (to an NPP and other installations of its infrastructure), and also to the environment. Now while miss direct legal and normative documents by the quantitative definition of these losses. Some suggestions on this problem are stated below.

For a tentative estimation of loss $U(\tau)$, it is possible to use the simplified statistical and expert information on such losses. Generally, values of losses are defined by two basic parameters:

- Losses of human lives or health at occurrence and progressing of unfavorable situations

- Economical losses (for example, in Rubles or USD) from a loss of life, from maiming to people, and from fractures and damages of technosphere installations and the environment

Direct loss $U(\tau)$ for the LS-7 limiting state interlinked immediately to fracture of the NPP or full termination of its service. Then, the datum of loss $U(\tau)$ can be accepted to the equal cost of the NPP. In this, the loss can and should include charges $U(\tau_1)$ within 1–2 years on a primary elimination of the consequences of disaster or accident (realization of protective measures, evacuation of the population, and termination of infrastructure installation operation). These charges at $(\tau_1 \geq \tau)$ several times (2–4) can exceed the initial loss $U(\tau_c)$. Decrease of secondary consequences of heavy disasters on an NPP (making of shelters, recultivation, medical examination and the help, and compensating payments) demands complementary essential annual expenditures $U(\tau_2)$ for a long time $\tau_1 < < \tau_c \leq \tau_2$. In **Figure 20** is displayed schematization of the relative losses $\overline{U}(\tau) = U(\tau_i)/U(\tau_c)$ depending on time $\Delta\tau$ after the occurrence of heavy disaster ($\Delta\tau = \tau_s = \tau$) at reaching the most dangerous limiting state of the LS-7 type, summarized in **Table 3**.

With the reduction of the hazard level of accidents and disasters (at transition of limiting states from the LS-7 to the LS-1), value $U(\tau_c)$ and $\overline{U}(\tau)$ decrease because of decrease of losses $\overline{U}(\tau_1)$ and $\overline{U}(\tau_2)$.

From assemblage of tens methods for definition of risks parameters as the most simple is the statistical or determined-statistical method according to which it is possible to write

$$R(\tau_i) = P(\tau_i) \cdot U(\tau_i), \tag{19}$$

where τ_i is the time for which one the risk assessment is conducted and $P(\tau_i)$ and $U(\tau_i)$ are the probabilities and losses for time τ_i.

If under τ_i is fathomed the time of unfavorable event occurrence of τ_c, then according to Eq. (19), it is possible to obtain
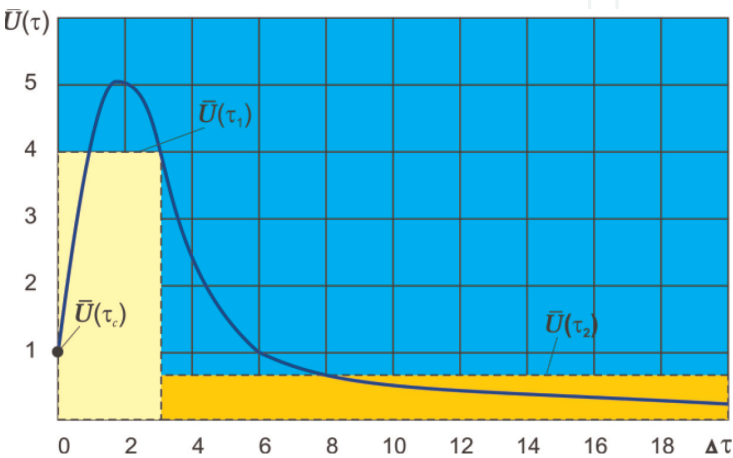

Figure 20.
The time-history and schematization of losses U(τ).

$$R(\tau_c) = P(\tau_c) \cdot U(\tau_c) = P^*(\tau_c) \cdot U(\tau_c). \tag{20}$$

Risk $R(\tau_c)$ is possible to consider as risks of the implemented unfavorable events at $\tau_i = \tau_c$ and to use them for prediction of events for times $\tau_i \geq \tau_c$. One such prospective risk appears as the risk for the current phase of service $\tau_i = \tau_s$. In this case, on the basis of Eq. (19), it is possible to write

$$R(\tau_s) = P^*(\tau_s) \cdot U(\tau_s), \tag{21}$$

where τ_s is the time after unfavorable event $(\tau_s \geq \tau_c)$.

This time can be situated in the interval $\tau_c \leq \tau_1 \leq \tau_2$. Then, for one operated unit of the NPP, the common risk at reaching the given i-group of limiting state from the LS-7 to the LS-1 will constitute

$$R(\tau_c)_\Sigma = \sum_{i=1}^{7} R(\tau_c)_i \tag{22}$$

If at loss estimations to consider not only direct losses at occurrence of unfavorable event $U(\tau_\kappa)$ together with complementary losses $U(\tau_1)$ and $U(\tau_2)$, then it is possible to define common (integral) losses as

$$U(\tau)_\Sigma = U(\tau_c) + U(\tau_1) + U(\tau_2) \tag{23}$$

These integral losses respond to the appropriate risks

$$R(\tau_\Sigma)_\Sigma = \sum_{i=1}^{7} U(\tau_\Sigma)_i P^*(\tau_s)_i. \tag{24}$$

On the basis of results of an estimation considered above risk components, it is possible to build dependences between basic parameters of risk for the NPP— probabilities $P(\tau)$ occurrence of unfavorable situations and losses $U(\tau)$ from them (**Figure 21**).

The line had above and design points in the **Figure 21** belong to probabilities $P(\tau_c)$ and to losses $U(\tau_c)$ for the moment of accident or disaster occurrence on the NPPs. The lower line made like overhead characterizes a negligible zone of risk parameters $[P(\tau_c)]_{\min} - [U(\tau_c)]_{\min}$ and the midline characterizes a zone of acceptable risks $[P(\tau_c)] - [U(\tau_c)]$. If to allow common (near-term and long-time)
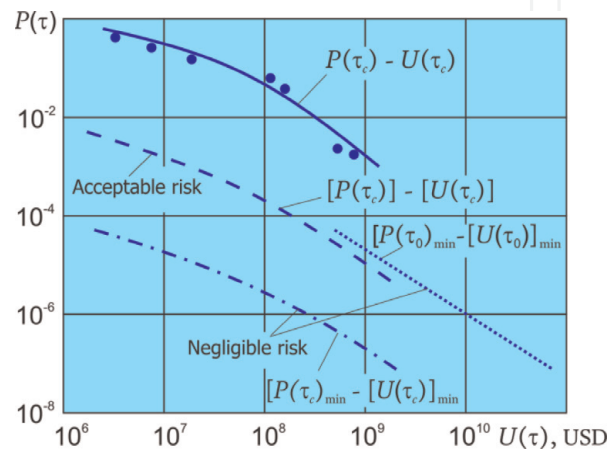

Figure 21.
Parameters of risks for the NPP with reactors of VVER types.

for negative consequences of accidents and disasters, it is possible to build a line of negligible risk parameters $[P(\tau_\Sigma)]_{\min} - [U(\tau_\Sigma)]_{\min}$.

5. Conclusion

From stated above follows that the major problems which have been not decided while to the full for a NPP there are problems of provision of their protectability and safety on the basis of new scientific fundamental and application researches on mechanics, hydrodynamics, economics, mathematical and physical modeling of dangerous processes resulting to heavy disasters, and also development of detailed methods of the analysis of risks for heavy disasters.

Results of the fulfilled scientific researches and developments in this direction, integrated [3–8, 15–17] in the serial of monographic publications on strength, life time, and safety of power nuclear reactors, are initial scientific baseline for the applicable normative, designer, technological solutions on provision of protectability of the NPP equipment from heavy disasters on the basis of criteria of acceptable risks.

The above-mentioned results of analytical and experimental researches can be considered in the capacity of a theoretical basis for the subsequent development of practical models of the computational analysis of risks for strategically relevant installations of a nuclear energetic on the basis of the complex Eqs. (1)–(24). Development of such models, and the most important—their filling up statistically reliable probability distribution of fractures on groups of limiting states (see **Table 3**) on the one hand, and economical computations of losses, with another, it is necessary to consider as the major task for a solution of a problem of safe development of power supply of human community.

At up-to-date and subsequent stages of evolution of power engineering in Russia in the capacity of a basic recommended position, it is necessary to use the position about provision of an acceptable risk level of occurrence of accidents and disasters. In this connection, it is not obviously possible to ensure from social-economic and technological stands the declared principle of absolute safety with null risks ($R(\tau) = 0$). Owing to it, the solution of the delivered problem is brought together to determination of scientifically well-founded admissibility of occurrence of the emergency situations with possible minimization of loss caused by them, with an estimation of the greatest possible, acceptable, and controlled risk both at probable occurrence of global and national accidents and disasters, and their realization at regional and local levels.

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