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USSR Report

ENGINEERING AND EQUIPMENT

(FOUO 6/81)

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NUCLEAR ENERGY

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MODELING TRANSIENT AND EMERGENCY MODES ON SIMULATION TRAINER

Moscow ELEKTRICHESKIYE STANTSII in Russian No 8, Aug 81 pp 9-11

[Article by candidate of the engineering sciences S.G. Muradyan and engineers A.A. Ayrapetyan and O.S. Babadzhanyan, Yerevan Affiliate of the All-Union Scientific Research Institute for Nuclear Power Stations]

[Text] Unforeseen deviations from the specified operating mode are possible at the AES's in service because of disruptions in the operation of the production process systems, equipment or power unit monitoring and control systems [1].

In such situations, which are called nonstandard in the following, the operational personnel of the power unit should quickly determine the source of the deviation and take the requisite steps to restore normal operating conditions for the power unit. Incorrect actions by the operating personnel in nonstandard situations frequently lead to serious emergencies, and for this reason, reaction speed and error free action on the part of the operators when controlling the operational modes of AES power units take on special significance, and special teaching and training centers (UTTs) are being created for them, where these centers are equipped with trainers and other technical instruction tools.

Faults occur randomly in the systems of AES power units and constant readiness on the part of the operators during their shift work is required for the timely detection and elimination of the faults, something which also explains the stressful nature of their activity.

Operators go through training in the simulators in both teaching and testing modes. In the teaching mode, operator training in the simulator is carried out in accordance with previously composed scenarios, in which specific nonstandard situations are studied. The entire instructional course program is broken down into topics, upon the completion of which there is a test of the mastery of the material which has been covered. Upon completing the training course, operators in the teaching and training centers go through final certification and receive appropriate skill certifications [2].

In order to reproduce as completely as possible the work conditions of the operators on shift duty in the training simulator (the stressful psychological and physiological state and the constant readiness for intervention in the power unit control system), it is expedient during testing to simulate the random nature of the occurrence of defects in the power unit systems.

A method of simulating nonstandard situations in trainers for situations which occur in a power unit, taking into account the random nature of their occurrence, is treated in the following.

Nonstandard situations in a power unit can be caused both by defects within a power unit (defects in equipment, in control and monitor systems) and by external effects (for example, a sudden drop in the electrical load, operator errors in controlling the power unit, etc.) [3]. As a rule, operators using the power unit monitoring system do not detect the basic cause of the occurrence of nonstandard situations, but rather a certain condition of the production process system which is a consequence of the disruptions or deviations which have occurred.

Logic and dynamic mathematical models of a power unit are reproduced in the simulation trainer, where these models are developed on the basis of the calculated design configurations of the production process systems, including a specified complement of modeled equipment. The locations and nature of the defects being simulated in the trainer are ascertained by means of preliminary analysis in the calculated design configurations. As a result of doing this work, the set of those parameters is determined (the coefficients and variables) for the mathematical models, a change in which within a specified range (either continuously or in discrete steps) leads to the reproduction of the nonstandard situations in the trainer. It is not precluded that it will be necessary to incorporate supplemental logic or differential equations to introduce a certain class of defects into the mathematical models of the power unit.

Defects in equipment, operational failures and deviations from specified operating conditions of the production process systems of a power unit, which are termed perturbations in the following, can be broken down qualitatively into three classes.

The first class includes perturbations which as a rule do not lead to emergency situations. For example, the failure of individual controls, transducers, etc. Such perturbations are most frequently encountered during the process of operating AES power units.

The second class includes perturbations which, if there is no operator intervention, can lead to emergency situations. Examples of such perturbations are the loss of reactor controls, failures in opening or closing gate valves, in turning pumps on or off, etc.).

The third class includes perturbations which necessarily produce emergency situations. For example, a break in the pipes of the primary loop of power units with VVER's [water-moderated, water-cooled power reactors], the sudden disconnection of the main circulation pumps, the loss of power by the station, etc. Perturbations of this kind are encountered comparatively rarely. Since the simulation trainers are designed for the purpose of instruction, it is apparent that the probability of occurrence of perturbations must be changed in them as compared to actual power units.

The occurrence probability for the first class of perturbations can be reduced in the simulation trainer, since operators can acquire the requisite operational experience for such situations on an actual power unit. And vice-versa, the probability of the occurrence of the third class of perturbations must be increased in the trainers, since their occurrence entails serious consequences, and in this case, the probability of teaching operators the actions to take in such situations is small with an actual unit.

The random nature of the occurrence of nonstandard situations can be simulated by means of a random number generator [4].

A random number generator periodically (for example, with a period equal to the integration step) generates numbers in an interval (a, b), which is broken down into two parts (a, c) and (c, b), in which case, the size of the interval (c, b) is much greater than the size of the interval (a, c). If the random number falls in the interval (c, b), it is assumed that there is no perturbation. In turn, the interval (a, c) is broken down into small intervals, the number of which corresponds to the number of possible perturbations, while the width of the intervals corresponds to the probability of their occurrence. If the random number generator has produced a number in the interval $(a_i, c_i) \in (a, c)$, this means that the perturbation with the corresponding number i has been selected.

As was noted earlier, both discrete and continuous input parameters are included in the set of perturbations. If a discrete input parameter corresponds to the interval (a_i, c_i), then when the random number x falls at any point in the range (a_i, c_i), the given discrete perturbation is selected. However, if a continuous variable corresponds to the interval (a_i, c_i), the probability of the appearance of values of which likewise changes continuously over this range, then the choice of the values of the variable can be made in two ways. The input of the values of the input parameter can be digitized, breaking the interval (a_i, c_i) down into subintervals, the width of which is proportional to the probability of the appearance of discrete values of the input parameter. The choice of the values of the input parameter can likewise be made by means of a certain function of the random number x, similar to the distribution function for the probabilities of the appearance of the values of this parameter.

As an example of the second method, we shall consider the case where with an increase in the value T $[T \in (T_0, T_1)]$, the probability of its appearance falls off continuously:

$$T = T_0 \exp[\alpha(x - a_i)^2],$$

where $$\alpha = \ln(T_1/T_0)/(c_i - a_i)^2.$$

The probability of the occurrence of an arbitrary perturbation is:

$$q = \frac{a - c}{a - b},$$

while the probability of the occurrence of the perturbation corresponding to the interval (a_i, c_i) is:

$$q_i = \frac{a_i - c_i}{a - b}.$$

It is apparent that $q = \sum_{i=1}^{l} q_i$, where l is the number of perturbations.

The monitoring process in the simulation trainer using the method considered here for perturbation input is accomplished as follows.

The specified input mode is set in the trainer and the random number generator is triggered. The input of an arbitrary perturbation (from the selected set) into the production process being simulated should be accomplished during a time t_o, determined from procedural and psychological considerations. If we assume that there are n drawings over the time t, then the probability that one arbitrary perturbation will be selected during this time is

$$Q = 1 - (1-Q)^n$$

It is apparent that with a sufficiently large n, the probability of selecting any perturbation tends to unity.

The coordinate of the point c on the segment (a, b) for specified values of t_o and the drawing rate k is determined from the following expression:

$$c = a - (a - b)[1 - (1-Q)^{1/n_o}],$$

where n_o is the integer part of the expression kt_o.

The production process parameters begin to change the moment the perturbations are fed in; the operator should be capable of eliminating the defects and restoring the normal operating conditions of the power unit through his own actions.

The entire course of the check exercise is recorded and the corresponding printout is fed out at the end of the exercise.

In AES operational practice, it is possible for not just one emergency (nonstandard) situation to arise, but a chain of emergency situations, one after the other. Two methods can be used to simulate such superimposed complex nonstandard situations.

In the first method, the complex nonstandard situation is included as one perturbation in the interval (a, c). In this case, the choice of the complex nonstandard situation has a probability

$$\sum_{i=1}^{m} q_i$$

of being made, where m is the number of complex perturbations. With the choice of the i-th complex perturbation, the program is started which specifies the sequence for the input of the simple perturbations.

In the second case, a complex nonstandard situation can be created by repeatedly feeding in perturbations, turning on the random number generator at a different frequency for each subsequent step of the drawing (the time segment from one selected nonstandard situation to the next). In this case, it becomes possible to create complex perturbations as the need arises. It is apparent that the choice of the same perturbation in different steps of the drawing cannot be tolerated, and for this reason, following the choice of a perturbation, the interval (a_i, c_i) corresponding to it is added to (c, b).

An advantage of this method is the fact that one can create a rather large number of complex nonstandard situations with the superimposition of two, three or more perturbations from the specified set of perturbations. For example, with the superimposition of three perturbations, the overall number of complex nonstandard situations will be equal to $l(l - 1)(l - 2)$.

The time allocated in the trainer for each student in the teaching mode is limited, and for this reason, the trainees during the training process can in practice study only a portion of the nonstandard and emergency situations modeled in the simulation trainer. All of the situations provided for simulation can be reproduced in the testing modes using the proposed method, i.e., the traineee will sometimes be confronted by situations with which they are familiar only from theoretical exercises. In conclusion, it must be noted that the method considered here provides for simulation of operator's work conditions in simulator for on-line power units, something which allows us to hope for a more objective estimate of their level of training. Moreover, during the process of training in the simulator, the operators will also acquire definite psychological and physiological skills when working in nonstandard situations.

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PERFORMANCE OF LENINGRAD NUCLEAR ELECTRIC POWER STATION REVIEWED

Moscow ELEKTRICHESKIYE STANTSII in Russian No 8, Aug 81 pp 6-9

[Article by engineer A.P. Yeperin, Leningradskaya AES]

[Text] The 1970's were marked by the rapid expansion of a new area in the field of power engineering. This new area was born in the Soviet Union in 1954 with the start of the first nuclear electric power station in the world in Obninsk.

A widescale program of nuclear power station construction in the European region of the USSR was planned through the directives of the 24th and 25th CPSU Congresses. An even more ambitious program of AES construction was approved at the 26th CPSU Congress.

The progress in the development of our nation's nuclear power engineering is linked to no small extent with the creation of the Leningradskaya Nuclear Electric Power Station imeni V.I. Lenin.

The construction of the station was started in 1967, while the main unit with an electrical capacity of 1,000 MW having an RBMK-1000 reactor and two 500 MW turbine generators was brought on line in December of 1973. In July of 1975, the second unit of the station with the same reactor went on line at the industrial load.

In 1975, the construction of the second stage of the station was started: the third and fourth power sets with capacities of 1,000 MW each. The third power unit was placed in service in December of 1979, while on February 9th, 1981, the creators of the Leningradskaya AES (LAES) marked a new victory: the fourth power unit was brought on line at the industrial load. The socialist obligations assumed by the collectives of builders, installers, alignment workers as well as operational workers in honor of the 26th Congress of the Communist Party of the Soviet Union were successfully completed with this event. The Leningradskaya AES became one of the largest nuclear power stations in the world when the fourth unit was brought on line.

The operation of the first and subsequent units of the station has made it possible to accumulate invaluable experience which was used in the design of other units and nuclear electric power stations with RBMK [channel type, graphite moderated high power] reactors. In this case, deficiencies were found in the project plan and design solutions, and ways of eliminating them were determined. Custom-made designs were mastered in a short period of time, which were related to the repair of the reactor and turbine equipment; the operational documentation was worked up. The first loading and unloading machine (RZM) underwent its testing successfully. An analysis of the operation of the first unit of the Leningradskaya AES has made it possible to begin the design of the next series of AES's with RBMK reactors, but now with a capacity 50 percent higher (the RBMK-1500). In this case, the core dimensions of the reactor will remain practically the same as for the 1,000 MW reactor (the RBMK-1000). This has become possible because of the development of a new type of fuel cassette with heat exchange intensifiers, which have been tested successfully at the Leningradskaya AES.

Special attention is being devoted to questions of safety in servicing the reactor installation and the production process systems related to it, as well as to issues of environmental protection in the design of nuclear power stations.

All of these questions have been successfully resolved in the design of the power generation units of the Leningradskaya AES imeni V.I. Lenin.

Safety in reactor control is assured by a highly reliable control and protection system (SUZ), which incorporates three systems of automatic controllers for the overall reactor power and a system of local automated controllers and protection to maintain the power within the specified range in individual regions of the reactor core. A SFKRE system (physical monitoring of the liberated energy distribution) provides for monitoring the energy liberated in each production process channel. This system, as a part of the complex with the "Skala" centralized monitoring system, which is designed around specialized computers, continuously performs calculations to determine the thermal engineering reliability of each production process channel of the reactor. The operator who controls and monitors reactor operation receives continuous data from this system in the form of light signaling and printout charts. This makes it possible to perform a timely analysis of changes in the operational mode of the reactor installation and take the appropriate steps.

A separate system provides for continuous monitoring of the hermetic seal integrity of the jackets (KGO) of the fuel rods in which there is nuclear fuel in the form of uranium dioxide and highly radioactive solid and gaseous uranium fission products. This system makes it possible to determine the number of a production process channel in which there is a cassette with a loss of seal of the fuel rod. When such a cassette is detected, the loading and unloading machine comes to assist, which installs a new cassette in place of the one with the seal failure, while the spent cassette in unloaded in a holding tank in a separate sealed cylindrical case. Thus, the cladding seal status monitoring system, as part of a complex with the loading and unloading machine, makes it possible to maintain the purity in the circulation loop through the reactor and provide for a normal radiation status for the servicing personnel.

Because of the fact that the reliability and safety of reactor operation depend on the condition (integrity) of the production process channel piping, to monitor their condition, the appropriate continuous monitoring KTsTK system has been developed (monitoring the integrity of the production process channels), which reacts to the slightest leaks of coolant from the channel into the graphite brickwork of the reactor, and makes it possible to determine the number of this channel. The existing equipment permits the replacement of a channel where necessary in a shutdown reactor.

In case of an emergency shutdown of a unit, the designers and project planners have provided systems for scramming the reactor and subsequently cooling the core. In this case, special attention has been devoted to the reliability of the electrical power supply for the station shutdown mechanisms which participate in these operations. Besides the traditional ways of boosting the electrical power supply reliability for the station, a diesel electric power generator is additionally installed which is started automatically by the actuation signals for the emergency protection and provides electrical power for the reliable power supply section.

Despite the fact that the water flowing through the reactor core has a relatively high radioactivity, this nonetheless is no threat to the safety of the servicing personnel. Such a situation is assured by the fact that all of the production process systems with the radioactive coolant are made as tight systems and housed in special protective boxes. Possible leakage of the radioactive water is localized by means of a system for trap water collection and processing, as well as intentional leaks. The water which is processed and purified by this system is returned to the production process loop.

The liquid radioactive wastes which appear during the process of cleaning and reprocessing the production process loop water are directed via a special route into special tanks for further reprocessing and subsequent burial of the highly radioactive residue. The atmospheric purity of the ambient air basin and the air in the rooms is maintained by means of production process and overall ventilation exchange systems, the operational mode of which provides for clean air delivery to attended rooms and exhaust with subsequent purification in special filters from unattended rooms. Following the filters, the air is ejected through a ventilation pipe 150 meters high.

Despite the reliability of the steps taken to protect the servicing personnel and the environment against radiation, there is an automated dosimetric monitoring system at the station for the gamma radiation level, as well as the concentration of radioactive gases and aerosols in the most important production rooms and in the air ejected through the ventilation stack. The cleanliness of the wall surfaces of the rooms and the production process equipment is checked periodically as well as the presence of radioactivity in the atmospheric air, the water basin, soil, vegetation and foodstuffs. The monitoring system confirms the reliability of the adopted protective measures.

Operational experience with the first two units of the station, and the increase in the stringency of the requirements placed on assuring the safety of AES's and protecting the environment, as well as the achievements of science and engineering, have necessitated new design solutions for individual equipment and production process questions in the design of the third and fourth units of the station.

All of this was responsible for a number of special features of the third and fourth units as compared to the RBMK-1000 reactors which had gone on line previously:

--The modular layout of the main and auxiliary installations and production process systems, which makes it possible to resolve questions of repairing the equipment and fittings in a nonoperating unit relatively simply and preclude errors by personnel during changeovers;

--The improvement of the system for containing the products from a loss of seal accident by containers and tanks in emergencies, related to pipe breakages in the multiple forced circulation (MPTs) loop;

--The introduction of an improved system for emergency cooling of the reactor;

--The use of new types of main circulation pumps with a new end seal for the shaft;

--The cooling loop of the protection and control system is made in a gravitationally driven configuration using an independent operational loop.

The structural design and production process features of the second stage units have brought about a marked change in the layout of the main building, the systems housed in it and have predetermined specific features of the start-up and alignment work as well as the physical and power starts of a unit.

A major features of starting a reactor of the third and fourth units is the fact that two percent enriched fuel is used in the initial charge of all fuel rods in them for the first time in reactors of this type.

In the process of running the physical starts of the RBMK reactors of the first stage of the Leningradskaya AES, extensive experimental data was obtained on the physics of the reactor core. The analysis of the work done in this case and the results obtained showed that differences are possible between identical reactors which are due to the scatter in some of the technological parameters such as the graphite density, the fuel content in a fuel assembly (TVS), etc. Moreover, there is necessity of an experimental check of new equipment or operational solutions with each start. Taking this into account, as well as the fact that the third unit reactor was the first of the RBMK series reactors in which only two-percent enriched fuel was used for the initial charge, the physical start of the reactor was accomplished in accordance with a program which provides for an experimental check of the design data and an actual estimate of the main neutron physics characteristics, the compensating capability of the protection and control system rods and the supplemental moderators (DP's), as well as setting up a full scale load while meeting the requirements of nuclear safety.

The excess reactivity which is due to the increase in enrichment is compensated by increasing the number of control and protection system rods from 179 to 211 and the length of the moderating portion of these rods from 5 to 6 meters, using "heavy" supplemental moderators, containing inserts made only of boron steel. A provision was made for the use of moderator rods which can be installed in the central cavity of a fuel cassette as an additional means of compensating for excess reactivity in the physical start program. In contrast to all of the previous RBMK-1000 reactors, the charging of the third unit reactor took place with water present in the production process channel (TK's) of the multiple forced circulation loop.

Critical experiments were performed during the charging which made it possible to correct the computational procedures for predicting the initial charge of the reactor; measurements were made of the bulk neutron fields for the full scale charge for the purpose of determining the nonuniformity coefficients of the cold unpoisoned reactors and determine the precision of the neutron physics calculations based on two-dimensional programs; experiments were run to determine reactivity effects in order to obtian data for running three-dimensional computational programs.

A set of start-up and alignment operations and experiments was performed during the power start period, which made it possible to check and more precisely specify the characteristics and operational modes of the equipment, the production process systems at various power levels, including the neutron physics characteristics of the core, the temperature conditions of the metal structures and the grphite brickwork, as well as effects caused by activation and reactor radiation, i.e., check the quality of the biological shieldings; the radio-chemical composition of the coolant for all loops; the radiation status in the rooms and on site, as well as the dynamic characteristics of the unit. The control point settings for the protection, interlocking and signaling were also made more precise during this same period; the safety valves and the automatic control systems were adjusted; the equipment and instruments of the physical monitoring system for the energy distribution, monitoring the hermetic seal of the jackets of the fuel rods as well as monitoring the integrity of the production process channel were calibrated and aligned.

The programs which were carried out in executing the physical and power start of the third unit reactor makes it possible to accomplish the start-up and aligtnment operations as well as charge the core and bring the reactor up to power in the fourth unit a shorter period of time.

The experience of builders, installation and set-up workers as well as operational workers, supported by the requisite organizational and technical measures (the implementation of joint construction and installation work, a significant consolidation of the main metal structures and assemblies outside the installation area, as well as the pre-assembly examination of the pumping equipment, fittings, electrical equipment, etc.) and their self-sacrificing labor made it possible to bring the fourth unit of the station on line within a curtailed timeframe. The connection of the first turbine generator of the third unit took place three months after the physical start and only 1.5 months after the physical start of the fourth unit.

The mastery of the design capacity of the fourth unit of the Leningradskaya AES is a complex and important task, on which the station staff is now working. The experience accumulated and the successes achieved in the start and mastery of the first three units inspire confidence in the successful resolution of this problem.

The station staff finished 1980, the concluding year of the 10th Five-Year Plan, with good results:

--The five-year plan for electrical power output was fulfilled on November 4, 1980, and the plan for power generation was fulfilled on November 17, 1980;

--The annual plan for electrical power output was fulfilled on December 18 and that for power generation on December 20;

--The installed capacity utilization factor was 73 percent, despite the fact that the installed capacity of the third power unit was mastered in the first half-year following start-up;

--The production cost per KWH of electrical power was reduced by 4.1 percent and amounted to 0.797 kopeck/KWH.

Based on the results of the All-Union Socialists Competition of 1980, the station staff was awarded the challenge Red Banner of the CPSU Central Committee, the USSR Council of Ministers, the All-Union Central Trade Union Council and the Komsomol Central Committee, as well as the memorial medal "For High Work Quality and Efficiency in the 10th Five-Year Plan", which was entered on the All-Union Board of Honor of the USSR Exhibition of National Economic Achievements.

The Leningradskaya AES imeni V.I. Lenin has become the major supplier of electric power to the Leningrad Power Administration system. The plan for electric power generation in 1981 is 22.6 billion KWH. By the end of the year, the total amount of electrical power generated by the station from the time it was started will reach 100 billion KWH.

The successes of the staff of the Leningradskaya AES imeni V.I. Lenin once again underscore the correctness of the party and government policy related to the further development of power engineering in our nation and inspire confidence that the tasks ahead in this field will be carried out successfully.

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ANALYSIS OF EQUIPMENT FAILURE AT ACTIVE SOVIET NUCLEAR POWER STATIONS WITH VVER-440 REACTORS

Moscow ATOMNAYA ENERGIYA in Russian Vol 50, No 4, Apr 81 (manuscript received 14 Nov 80) pp 248-250

[Article by F. Ya. Ovchinnikov, L. M. Voronin, B. B. Baturov, A. A. Abagyan and S. A. Lesnoy]

[Text] One of the major trends in development of nuclear power in the Soviet Union is construction of nuclear electric facilities with water-cooled water-moderated power reactors [VVER]. The first power facility with a VVER-210 was put into industrial operation in 1964 at the Novocherkassk Nuclear Electric Plant. A large series of power facilities with VVER-440 reactors has been put into operation since 1971 in the USSR and several other nations (East Germany, Bulgaria, Finland, Czechoslovakia) with technical assistance from the Soviet Union.

Operation has confirmed the correctness of engineering decisions in design developments, as well as conformance of the actual working characteristics of power facilities to the projected levels. Operational experience has also enabled determination of ways to further improve equipment and technological systems in accordance with increasing requirements for safety and reliability of nuclear electric plant operation.

Safety problems for normal working conditions can be considered completely solved. However, for emergency conditions these problems need further research. International experience in the development of nuclear power shows that the very concept of "safety" and the methods of achieving it are undergoing continuous changes in connection with massive construction of nuclear power facilities and the search for rational, technically feasible means of ensuring safety. VVER reactors of the first generation had shielding and localizing systems corresponding to the limited scale of a maximum credible accident that was accepted at the time. Considerable emphasis was placed on the factor of keeping the nuclear electric plant far from populated areas. The safety systems in power facilities with unified VVER-440 reactors that are now being introduced are designed for counteracting more extensive damage up to and including a break in the pipelines of the main circulation loop with maximum diameter for which the consequences of an accident are potentially more serious. Obviously it is very difficult to solve this problem by technical means alone, i. e. by using hardware to meet all safety requirements in building a nuclear electric plant that does not subject the surrounding environment to at least a slight risk of contamination. All efforts in the area of safety should be directed toward reducing the degree of risk. Such efforts will be fruitful only upon condition of inseparable combination of technical safety facilities with a high level of organization in utilizing these facilities.

Experience in the operation of nuclear electric plants throughout the world shows that while we are developing and constantly improving hardware for preventing and localizing large accidents (up to and including instantaneous transverse rupture of maximum-diameter pipelines), we need to be just as serious in working out and perfecting methods and means of preventing and clearing up so-called "minor accidents." In large part, these methods and means are the same as those aimed at ensuring the reliability of nuclear power plants as sources of energy, since any disruptions in the operation of major equipment that are due to failures and defects lead to power limitations for purposes of preventing deviations of the parameters of the facility beyond safe limits. In other words, to ensure safety, a reduction in reliability predetermines the necessity of placing constraints on the working conditions of a facility.

Comprehensive in-depth analysis of equipment operation to find the weakest links in technological systems of nuclear electric plants and improve their reliability has been a matter of course since the startup of the first VVER power facilities. Since 1977, a unified system has been in operation in the USSR for collecting data on failures and defects of nuclear electric plant equipment. The acquisition of reliable information enables isolation of the most typical failures that lead to emergency outages, unplanned down time and reduced economic efficiency of nuclear electric plants. Timely determination of the causes of equipment failures and defects (especially for the equipment of systems having to do with the safety of a nuclear electric plant) means that effective work can be done on improving this equipment from the design stage to final operational use. For the sake of convenience of such analysis and evaluation of the influence of failures and defects on the operational reliability and safety of nuclear electric plants, equipment has been divided into groups in accordance with functional designation. The table shows the spectrum of distribution of failures of equipment by percentages as typical of power facilities with VVER-440 reactors.

In analyzing the information, consideration was taken of all kinds of failures, both complete and partial, that lead or may lead to limitations in the operation of major equipment, as well as those failures that do not affect normal operation of power facilities due to the secondary nature of equipment, or built-in redundancy. The table shows that 11.3% of the failures pertain to equipment of the primary circuit having the greatest significance from the standpoint of ensuring reliability and safety of the nuclear electric plant. Failures of reactor equipment, including the control system, amount to ~4%, failures of steam generators 3.5%, and of pipelines--less than 1%. This shows the fairly high level of reliability of the main circulation loop. The remaining 88.7% of failures and defects pertains mainly to equipment that is not specific to nuclear electric plants and is typical of conventional power facilities.

A large number of failures (~38%) pertain to monitoring and control instrumentation. However, this has almost no effect on the reliability of the power facility since it most frequently involves instruments and communication lines with adequate backup. Analysis of the statistics of equipment failures for 1977-1979 showed that

Distribution of Failures and Defects of Equipment in Nuclear Electric Plants With VVER Power Facilities According to Functional Groups and Individual Kinds of Equipment

Equipment Group	Number of Failures per Unit per Year	Percent
Primary Circuit Equipment:		
reactor equipment	5	4.3
steam generator equipment	4	3.4
main circulation pumps	2	1.7
main shutoff gates	1	0.9
pipelines	1	0.9
For the Group	13	11.3
Turbine-Unit Group:		
turbines	1	0.9
condensers	3	2.6
steam-superheater separators	3	2.6
regenerative heaters	6	5.2
For the Group	13	11.3
Pump Equipment of All Kinds	9	7.8
Fittings (Except for Main Shutoff Gates	10	8.7
Blower Equipment	7	6.1
Compressor Equipment	4	3.5
Electrical Equipment:		
turbogenerators	1	0.9
electric pump drives	3	2.6
electric drives of control assemblies	2	1.8
breakers, disconnects	9	7.8
For the Group	15	13.1
Monitoring and Control Instrumentation:		
primary instruments	9	7.8
secondary instruments	23	20.0
communication lines	12	10.4
For the Group	44	38.2
TOTAL	115	100

the most characteristic defects are welding flaws (up to 32%) and hidden flaws in materials (up to 28%). Failures and damage through fault of servicing personnel amount to less than 7%, which is evidence of a rather high skill level.

To work out requirements for equipment reliability (which are especially necessary on the stages of design and manufacture), reliability indices are calculated on the basis of statistical data on equipment failure. One such very important index for restorable items is the parameter of failure rate $\omega(t)$. Because of the comparatively small volume of the statistical sample, only point values of $\omega(t)$ have been obtained, without estimation of the confidence level. Nevertheless, these values of $\omega(t)$ establish a lower limit of reliability and can be used as primary normalized data in the design, manufacture and utilization of equipment. For example $\omega(t)$ in hr^{-1} for the reactor is $(1.6-2.1)\cdot10^{-5}$, for the steam generators --$8.2\cdot10^{-5}$, for the reactor control system--$6.2\cdot10^{-4}$, for the main circulation pump--$2.6\cdot10^{-5}$, for the turbine--$8\cdot10^{-5}$. Comparison of these data with those available for equipment of non-Soviet nuclear electric plants shows that they are completely comparable if consideration is taken of the fact that ordinarily only total failures are considered in calculations of $\omega(t)$ in non-Soviet practice.

The given information shows that failures and flaws apply mainly to subsidiary equipment, or to auxiliary systems of major equipment. Therefore these is no reduction in the reliability and safety of the nuclear electric plant as a whole. This is evidenced by the stable and high level of the coefficient of utilization of installed power of VVER-440 power facilities: in 1970 this index was 72.6%, in 1978--80.7%, in 1979--73.8%.

We point out that the optimum coefficient of utilization of installed power for power facilities with VVER-440 reactors in the USSR is 80%, which corresponds to 7000 hours of operation of the equipment at rated power per year. This is determined by established periodicity and standards of duration of repairs of major equipment (reactor equipment, turbines and so on).

Analysis of the structure of the coefficient of utilization of installed power shows that underuse of installed capacities associated with unplanned repairs and equipment defects, i. e. due to down time having a direct relation to reliability and safety of the nuclear electric plant, amounts to no more than 3.7%, whereas this index was 8% in the early years of operation of VVER-440 reactors. This is evidence of an appreciable improvement in the reliability of major equipment over the elapsed period.

Continued work in the following major areas will ensure retention of the attained reliability level and further improvement:

perfection of equipment design;

improving the quality of the equipment during manufacture and the quality of installation as a basis for operational safety and reduction of the probability of failures and damage. Programs of quality control have been developed and implemented at the manufacturing plants for major equipment. All equipment arriving at the nuclear electric plant goes through a pre-instllation entry inspection. Improvements are being made in the technology of installation and welding processes, and in methods and equipment for quality control on welding jobs;

checking the condition of equipment during utilization with application of up-to-date methods for early detection of defects;

improving and increasing the technical level of utilization;

improving the efficacy of supervision for observance of directive and normative-technical documents in the process of manufacture, installation and utilization of nuclear electric plant equipment;

improving the skill of personnel working in the nuclear electric plant, and systematic personnel safety training in accordance with specially developed comprehensive programs.

On the whole, experience in utilization of VVER-440 power facilities brings us to the conclusion that they are sufficiently highly reliable and safe.

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DESIGN MEASURES TO ENSURE OPERABILITY OF NUCLEAR ELECTRIC PLANTS WITH RBMK REACTORS UNDER EMERGENCY CONDITIONS

Moscow ATOMNAYA ENERGIYA in Russian Vol 50, No 4, Apr 81 (manuscript received 14 Nov 80) pp 251-254

[Article by I. Ya. Yemel'yanov, S. P. Kuznetsov and Yu. M. Cherkashov]

[Text] A number of advantages of RBMK uranium-graphite boiling-water channel reactors have brought about their extensive use in nuclear power in the USSR [Ref. 1]. RBMK reactors are characterized in particular by high reliability thanks to monitoring and control of parameters of individual process channels, and also replacement of the fuel assembly without shutting down the reactor.

In successful operation at the present time are seven power facilities with RBMK reactors having 1000 MW of electric power; a program has been worked out and is being implemented on constructing and utilizing several more reactors. Design, construction and utilization experience has enabled us to go on to construction of the RBMK-1500--a channel reactor of the same design and dimensions with power of 1500 MWe.

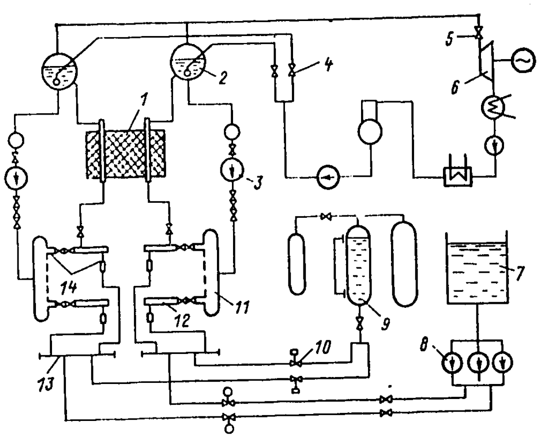

Fig. 1. Schematic diagram of the circulation loop of the RBMK-1000: 1--reactor; 2--separator; 3--main circulating pump; 4--level regulator; 5--pressure regulator; 6--turbogenerator; 7--water tank; 8--pumps of self-contained emergency reactor cooling system (ERCS); 9--pumped storage unit of ERCS; 10--fast-action valve of ERCS; 11--pressurized collector; 12--distributing group collector; 13--ERCS collector; 14--flow limiters

The fundamental heating arrangement of the RBMK-1000 (Fig. 1) is typical of one-loop boiling-water reactors. The coolant is circulated through the reactor by eight main circulating pumps, two of which are backup units. Steam separation is done in four separators of gravity type, and the saturated steam then goes to the turbines and to the intermediate steam superheaters. Two turbogenerators are installed on each power unit with RBMK-1000 reactor.

The RBMK parameters are monitored by a centralized control system based on computer equipment. The system provides operating personnel with visual and recorded information on the operation and condition of components of the structure.

Energy distribution through the volume of the core is monitored by a physical control system that includes radial and heightwise intrareactor sensors. An algorithm for calculating the power of all process channels with respect to discrete reference points, i. e. by channels with sensors, is realized in the PRIZMA program for the plant computer that is part of the centralized control system. The program utilizes characteristics of radial energy distribution obtained by a program of physical calculation on an external computer, and also the position of the control rods at the same instant of time when the physical calculation is done. These data are fed to the plant computer via punched tape.

The power of the process channels is periodically calculated upon order by the operator. The results are used to correct the energy distribution. On-the-spot control of energy distribution is done by operational personnel directly in accordance with the readings of the intrareactor sensors.

The distribution of coolant flowrate through the process channels is monitored by flowmeters installed at the inlet to each channel. The reactor design provides for on-the-spot control of the distribution of water flowrate through the process channels by changing the position of the channel regulating and cutoff valves. The capability of measuring and controlling the water flowrate in each process channel is a distinguishing feature of the RBMK; this capability ensures the required maneuverability of the distribution of reserves before a heat exchange crisis upon a change in the power of the reactor or radial energy distribution.

Leaky fuel elements and loss of integrity of channel pipes are detected in ample time by systems for monitoring gas tightness of fuel element cladding and the integrity of each process channel pipe.

The reactor control system is designed with consideration of requirements of nuclear and radiation safety. For example, local automatic controller and local scram systems are provided to prevent unsteady deformations of energy distribution. For purposes of maintaining power conditions of operation of the facility when complete shutdown of the reactor is not required, provisions are made in the reactor control system for rapid controllable power reduction in some cases, in addition to the conventional facilities for complete damping of the chain reaction. These provisions include disconnection of some main circulating pumps, reduction of the water level in the separators, reduction of feed water flowrate, partly disconnecting or dumping the load by turbogenerators in the unit, increasing the energy intensity of the fuel elements and the like. The various types of emergency protection are actuated by signals of equipment malfunction.

To study emergency conditions, a mathematical model was developed including equations of kinetics, hydrodynamics and heat exchange, and algorithms for operation of equipment and systems for automatic control of nuclear electric plant parameters. Comparison of the results of calculations done with the aid of the model, with data on individual dynamic processes that have occurred in operating nuclear electric plants showed that the model satisfactorily describes the dynamics of a power facility. Some emergency conditions, mainly involving a transition to natural circulation, were studied on special model stands.

This paper discusses the results of investigation of emergency conditions caused only by total de-energizing of the power facility, disruptions in the feed water supply system, and breaks in large pipelines of the circuit.

When the facility is de-energized, emergency equipment operates that scrams the reactor; early in the accident, the core is cooled by main circulating pumps that do not immediately stop, and then by natural circulation of the coolant. For reliable cooling of the reactor during this period, the main circulating pumps must have fairly large flywheel masses, and for this purpose special flywheels are incorporated into their design. A diesel generator facility is automatically switched in to supply emergency feed pumps, the pumps of the emergency reactor cooling system and the like, that are necessary for carrying off the residual energy release in the reactor.

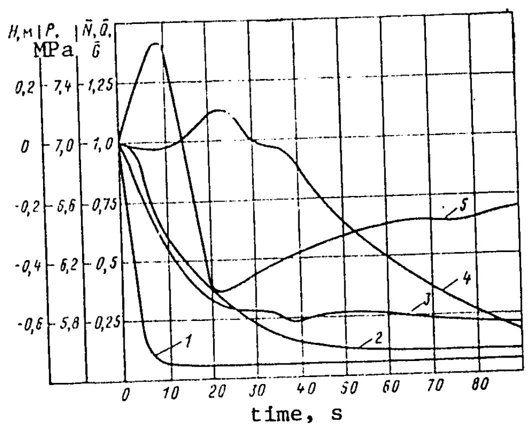

Fig. 2. Emergency state of de-energizing of the facility: 1--relative neutron power $\bar{N}$; 2--relative thermal power $\bar{Q}$; 3--relative coolant flowrate $\bar{G}$; 4--level H in the separators; 5--pressure P in the separators

The major parameters of the power facility in the state of de-energizing are shown in Fig. 2. At the beginning of the transient process, $\bar{Q}$ is somewhat higher than $\bar{G}$, i. e. in this period the average subcritical power reserve through the reactor is less than the rated value. However, calculations show that the reserve to the heat exchange crisis in the most stressed channels does not fall below unity. Thus it can be stated that in the early period of accidental de-energizing the reactor parameters do not go beyond safe limits.

Within 30-35 s after de-energizing of the main circulating pumps of the power facility, the core is being cooled by natural circulation of the coolant, and the stability and intensity of this circulation depend in large measure on several factors such as the design of the loop, the pressure therein, change in temperature and flowrate of the feed water and so on. To determine the reliability of core cooling in the mode of natural circulation, a set of experiments was done both on the technological model stand, and on the reactors of the first and third power units at the Leningrad Nuclear Electric Plant. Steady-state and transient conditions were studied with various parameters influencing the development and intensity of natural circulation. As a result, the parameters were determined that ensure reliable cooling of the reactor with natural circulation.

It should be noted that reliability and safety of reactor cooling in such a mode has been confirmed by work experience with operating nuclear electric plants with RBMK reactors.

Dynamic studies of the characteristics of a power unit during emergencies in the system of feed water supply have enabled determination of the conditions of safe operation of the facility, and elaboration of requirements for the equipment and for action by operational personnel in such emergency situations.

Deactivation of one of four operating feed pumps has demonstrated that in this state there are slight and even changes in technological parameters, and therefore it is not required to introduce any auxiliary protection of the equipment. To bring the reactor power into line with the altered feed water flowrate it is sufficient to reduce the power manually by adjusting the controller setting. When two or more feed pumps are disconnected at the same time, the emergency protection ensures safety of the facility by automatically reducing the reactor power to a safe level.

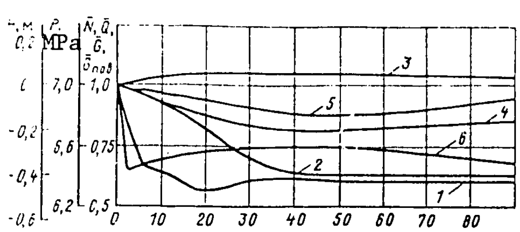

Fig. 3. Emergency state of disconnection of two feed pumps (1-5, see Fig. 2; 6--relative flowrate of feed water $\bar{G}_{пов}$)

The transient process with disconnection of two feed pumps and operation of emergency protection in accordance with a signal of reduction in the flowrate of feed water by 25% of the current value (Fig. 3) is characterized by the following: an automatic regulator reduces reactor power to 60% of the rated level with slight overcontrol; maximum deviation of the level in the separators is observed within 50 s, and amounts to 150 mm, after which the level is recovered; the steam pressure in the separators is maintained at a level close to the nominal value by means of a pressure regulator that unloads both turbogenerators in the facility.

Safety of the facility in the case of total cessation of feed water supply in the power units is ensured by emergency scramming of the reactor upon a signal of flowrate reduction below 50% of the current value. In this state, the water supply to the circulation loop by emergency feed pumps is ~10% of the nominal level; these pumps are energized within 10-20 s after cessation of the supply of feed water. Studies have shown that cessation of the feed water flow leads to a reduction of the level in the separators. This can cause undesirable trapping of steam on the downcomers of the loop, cavitation cutoff of main circulating pumps, and impediments to the development of natural circulation. To prevent the level from falling in the separators, the main circulating pumps are disconnected, which slows down the rate of reduction of steam content in the core and outlet lines of the reactor. As a result, less water is required from the separators to replace the steam in the circulation loop.

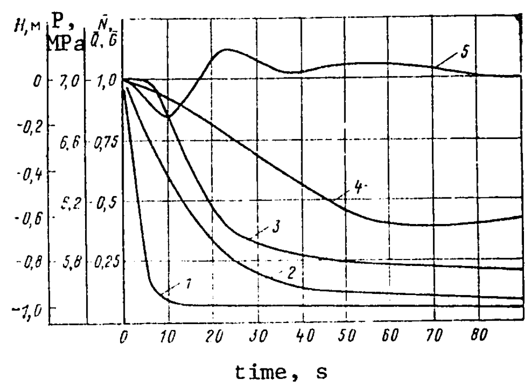

Fig. 4. Emergency state of disconnection of all four feed pumps (1-5, see Fig. 2)

The state of cessation of the supply of feed water with disconnection of the main circulating pumps is analogous to the state of de-energizing of the facility, which has been found to be safe. The main parameters of the power unit in this state are shown in Fig. 4. Analysis shows the following:

the rate of reduction in thermal power is greater than the rate of falloff in water flowrate throughout the entire transient process, which is evidence of reliable cooling of the core;

maximum drop in the level in the separators is observed within 75 s, and then the level begins to rise; the steam pressure in the separators decreases at the beginning of the process, then becomes somewhat higher than nominal, and after 72 s the pressure stabilizes at the nominal level.

Thus these results show reliability of cooling of the core in a state of complete instantaneous cessation of the supply of feed water accompanied by disconnection of the main circulating pumps. Therefore the main circulating pumps are disconnected with a delay of ~9 s after emergency protection operates in power facilities with RBMK reactors upon a signal of reduction in the feed water flowrate below 50% of the current value. The core is cooled down by natural circulation of the coolant.

It is assumed that the most serious emergency situations can arise when large pipelines of the power facility are ruptured. The design provides for technical facilities that prevent discharge of the steam-gas mixture into the service areas, and especially beyond the limits of the nuclear electric plant. Most typical damage to the circulating loop is breaks in small tubing (drains, impulse lines and the like). Rupture of a large pipeline is extremely improbable. Experiments on full-scale specimens have shown that a leak is possible in pipelines with diameter of ~800 mm at a pressure of 8.5-9.5 MPa if fatigue cracks are ~75% of the wall thickness in depth, and ~470 mm long [Ref. 2]. Inspection of the metal guarantees that there will be no sudden rupture of the pipeline, since the critical dimensions of defects are large, and they should be detected in planned shutdowns of the facilities. During inspection, the metal is examined and checked by special techniques (ultrasonic flaw detection, acoustic emission). In spite of this, the design of the nuclear electric plant provides for measures to ensure safety in case of instantaneous transverse rupture of the largest pipeline.

At the initial instant the leakage is about 6 metric tons per second in the case of complete instantaneous rupture of a pipe 300 mm in diameter, and ~40 metric tons per second for such rupture of a pipe 900 mm in diameter. As a result of analysis of emergency situations, two independent signals have been selected for operation of emergency reactor protection: a rise in pressure in the rooms where the pipelines of the loop are accommodated, and a reduction of the level in any

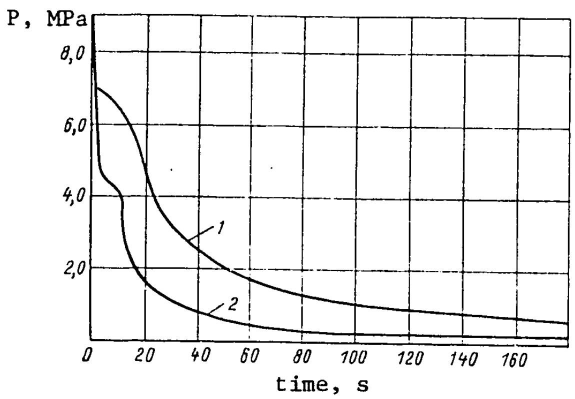

Fig. 5. Change of pressure in RBMK-1000 circulation loop upon rupture of a pressurized collector 900 mm in diameter: 1--pressure in separators; 2--pressure in the pressurized collector

separator to a value that exceeds its deviation from the nominal under transient conditions.

The most dangerous pipeline rupture is one on the pressure side of a main circulating pump, since this instantaneously cuts off the supply of coolant to the channels of the emergency half of the reactor. It is just such a hypothetical accident that has dictated the fast-action emergency reactor cooling system (ERCS), its maximum capacity (about 1.1 metric tons per second) and minimum time of discharge of all the coolant from the emergency loop (10-12 s). Fig. 5 shows the pressure change in the circuit with instantaneous rupture of 900 mm pressurized collector of a main circulating pump. Coolant from the ERCS is fed to the channels of this half to prevent damage to the fuel elements.

Water from the ERCS is sent to each distributing group collector, and to avoid nonproductive discharge of the water through the cross section of the break in the pressurized collector, check valves are provided at the inlet to the distributing group collector. The ERCS consists of two subsystems (see Fig. 1): the main subsystem with pumped-storage unit, and a prolonged cooling subsystem with special pumps and tanked water reserves. The cooling water is supplied from bottles, and after they are emptied the water is supplied by pumps to the ERCS of each half of the reactor, and thence through pipelines to each distributing group collector. Installed in the water feed lines to the collectors are fast-action valves that open when pressure rises in the rooms. When this happens, the water goes to the reactor loop in which the level has dropped in the separators, or the pressure differential has decreased between the pressurized collectors and the separators. Such an algorithm of engagement of the main subsystem of the ERCS ensures cooling of the core with complete or partial rupture of a large-diameter pipeline, and precludes a false alarm in case of accidents that do not involve loss of integrity of the circuit.

Studies have shown that acceptable temperature conditions of the fuel elements are ensured by the speed and productivity of the ERCS in case of any pipeline rupture up to and including a maximum break.

All equipment and pipelines of the circulation loop of the reactor are accommodated in securely tight enclosures that prevent emissions of the steam-gas mixture from the rooms of the nuclear electric plant into the atmosphere in case of breaks. The steam-gas mixture goes through special tunnels to a localizing unit where the steam is condensed. The enclosures are designed for an excess pressure of ~0.4 MPa, which is not surpassed even in the case of complete instantaneous rupture of the largest pipeline, thanks to special condensation devices of the bubbler type, a system of bypass valves, sprinklers and heat exchangers.

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INVESTIGATION OF EFFICIENCY OF DECONTAMINATING RBMK-1000 COOLANT OF TRANSURANIUM ELEMENTS

Moscow ATOMNAYA ENERGIYA in Russian Vol 50, No 4, Apr 81 (manuscript received 11 Feb 80) pp 274-275

[Article by A. M. Vorob'yev and N. P. Starodonova]

[Text] For purposes of radiation safety at nuclear electric plants of all types, both Soviet and non-Soviet, provisions are made for purifying the water coolant of radionuclides. The bypass purification facilities at nuclear electric plants are intended for extracting products of corrosion, soluble salts and radionuclides from the purging water. In nuclear electric plants with RBMK reactors, 200 cu. m of water per hour is purified with total volume of the multiple forced circulation loop of ~1200 m^3. The water is taken off from the pressure side of the main circulating pumps, and after appropriate treatment and cooling to 40-50°C it is sent to a purification facility, where it goes in sequence through mechanical (hydraulic perlite), mixed cationic-anionic exchange (KU-2, AV-17) and trapping filters. The process includes monitoring for pH, hardness, Cl^- content, products of corrosion (iron, copper), specific electrical conductivity and content of some radionuclides before and after purification. The purified coolant is heated to 270°C and returned to the circuit.

In addition to radionuclides of corrosion origin and fission products, the water of the multiple forced circulation loop may be contaminated with α-emitters that are formed as a consequence of slight surface contamination of the fuel elements with uranium or migration of radionuclides from defective fuel elements. Accumulation of transuranium elements in nuclear fuel was studied in Ref. 1, 2. It was shown that sequential capture of neutrons in the fuel forms a large quantity of isotopes of plutonium, americium and curium that are quite dangerous from the radiation standpoint. For example, 19 kg of ^{237}Np, ~700 kg of isotopes of plutonium, >100 kg of isotopes of americium and >2 kg of curium are formed after a three-year run in 80 metric tons of fuel with 3.3% ^{235}U enrichment at a reactor power of 1000 MW [Ref. 2].

Unfortunately, we have been unable to locate any published data on the coefficients of diffusion of transuranium elements from intact and leaky fuel elements. However, this quantity can be roughly estimated by considering the similarity of properties of lanthanides and actinides. For example, if the relative diffusion of iodine and cesium from leaky fuel elements is taken as 1, then diffusion is equal to 10^{-4} for cerium [Ref. 3]. In case of severe loss of integrity of fuel elements and contact between fuel and coolant, there may be a considerable increase in this parameter. In emergency situations with fuel meltdown, α-emitters may play a decisive part in raising the radioactivity of the coolant and formation of the radiation environment at the nuclear electric plant and its surroundings. In this connection, coolant purification takes on special significance.

To study the effectiveness of bypass decontamination with respect to transuranium elements, the content of ^{239}Np, plutonium isotopes, ^{241}Am and ^{242}Cm was determined in the coolant before and after purification. The studies were done in different periods of operation of the nuclear electric plant. The ^{239}Np content was determined by a γ-spectrometric method using a semiconductor Ge(Li) detector. Identification was done simultaneously with respect to five peaks (210, 228, 278, 316 and 334 keV). Americium and curium were determined by radiochemical methods based on precipitation on bismuth phosphate after the plutonium had been oxidized to the hexavalent state. Reduced plutonium (III) or (IV) was then also coprecipitated with bismuth phosphate. The α-activity count (100% efficiency) was then done in a layer of solid scintillator (ZnS) by the technique of Ref. 1. With background of 0.2 pulse/minute, the sensitivity of the method was 10^{-13} Ci/l (1 Ci = 3.700 10^{10} disintegrations per second). The error of determination was ±20% [Ref. 2].

Content of α-emitters in the water of the multiple forced circulation loop, Ci/l — Table 1

Utilization period	ΣPu, Am, Cm	ΣPu	$^{241}Am + ^{242}Cm$	ΣU
Initial (up to one year)	$(1\div5)\cdot10^{-13}$	$(1\div2)\cdot10^{-13}$	$(1\div2)\cdot10^{-13}$	$1\cdot10^{-13}$
Normal	$(1\div10)\cdot10^{-11}$	$(1\div5)\cdot10^{-11}$	$(1\div5)\cdot10^{-11}$	$(1\div3)\cdot10^{-11}$
Loss of integrity of fuel elements	$1\cdot10^{-9}$	$4,6\cdot10^{-10}$	$5,4\cdot10^{-10}$	$5\cdot10^{-10}$

Table 1 summarizes averaged data of investigation of the content of α-emitters in water for different periods of operation of a nuclear electric plant. Calculations show that the content of transuranium elements in the initial period of utilization [$(1-5)\cdot10^{-13}$ Ci/l] is due mainly to surface contamination of fuel elements with uranium during fabrication. As the length of the run increases, microcracks appear in the cladding, and the content of transuranium elements in the coolant increases by two orders of magnitude. A further increase in the content of these elements (to 10^{-9} Ci/l) is observed with occurrence of defective fuel elements subject to replacement. Some time after their replacement, the concentration of α-emitters in the coolant again stabilizes at a level of $(1-10)\cdot10^{-11}$ Ci/l.

Table 2 gives some results of evaluation of the efficiency of bypass purification of transuranium elements on nuclear electric plants with RBMK-1000 reactors.

This table shows that the water purification factors with respect to the indicated transuranium elements are fairly close, and are equal to 3000 in the case of fresh

Table 2

Efficiency of bypass removal of transuranium elements from coolant

Radio-nuclide	Coolant activity Ci/l		Purification factor
	before	after	
ΣPu, Am, Cm	$1,2 \cdot 10^{-8}$	$3,1 \cdot 10^{-13}$	3700
	$1,2 \cdot 10^{-9}$	$6 \cdot 10^{-13}$	2000
	$7 \cdot 10^{-10}$	$9,1 \cdot 10^{-13}$	755
	$2 \cdot 10^{-10}$	$4 \cdot 10^{-13}$	500
	$4 \cdot 10^{-10}$	$4 \cdot 10^{-13}$	1000
	$5 \cdot 10^{-11}$	$1 \cdot 10^{-13}$	500
	$4,4 \cdot 10^{-12}$	$< 1 \cdot 10^{-13}$	> 44
	$3 \cdot 10^{-13}$	$< 1 \cdot 10^{-13}$	
			$K_{av} = 1075$
ΣPu (^{239}Pu, ^{238}Pu etc.)	$3,5 \cdot 10^{-10}$	$7 \cdot 10^{-13}$	500
	$4,6 \cdot 10^{-10}$	$3 \cdot 10^{-13}$	1535
	$2 \cdot 10^{-10}$	$2 \cdot 10^{-13}$	1000
	$1,8 \cdot 10^{-10}$	$2 \cdot 10^{-13}$	900
	$1 \cdot 10^{-11}$	$< 1 \cdot 10^{-13}$	> 100
	$2 \cdot 10^{-12}$	$< 1 \cdot 10^{-13}$	> 20
			$K_{av} = 985$
$^{241}Am + ^{242}Cm$	$5,35 \cdot 10^{-10}$	$3,3 \cdot 10^{-12}$	160
	$5,4 \cdot 10^{-10}$	$4 \cdot 10^{-13}$	1350
	$2,2 \cdot 10^{-10}$	$3 \cdot 10^{-13}$	740
	$7,5 \cdot 10^{-11}$	$1 \cdot 10^{-13}$	750
	$2 \cdot 10^{-12}$	$< 1 \cdot 10^{-13}$	> 20
			$K_{av} = 750$
^{239}Np	$4,5 \cdot 10^{-4}$	$3,1 \cdot 10^{-7}$	1450
	$1,1 \cdot 10^{-4}$	$3,1 \cdot 10^{-7}$	360
	$2,5 \cdot 10^{-5}$	$1,3 \cdot 10^{-8}$	1900
	$1,5 \cdot 10^{-4}$	$3,8 \cdot 10^{-7}$	390
			$K_{av} = 1025$

charging of the filters with ion-exchange resins, about 1000 after 5-8 months, and 25-100 after 9 months on this charge. Individual values of K_{pur} for a filter service period of 5-8 months amount to 1025 for neptunium, 985 for plutonium and 750 for curium.

The degree of removal of transuranium elements from the coolant on ion-exchange filters is higher than for radionuclides of corrosion and fission origin. These results are to be expected since it is known that the strength of sorption of cations on KU-2 resin increases with increasing cation charge. The radionuclides of interest to us are arranged in the following sequence with respect to the strength of bonding to the cation exchange resin: Cs < Ba, Sr < Y, La, Ce, Am, Cm < Ru < Zr, Nb < Np, Pu. The bypass purification factors for removal of transuranium elements from the coolant are an order of magnitude higher than for other radionuclides. Removal of transuranium elements is good when steam is separated from water in drum separators: the corresponding purification factors are 10^4-10^5, i. e. the steam going to the turbines is nearly free of α-emitters.

Thus the standard purification system in nuclear electric plants with RBMK-1000 reactors ensures effective removal of neptunium, plutonium, americium and curium from the coolant.

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THERMOPHYSICAL PROPERTIES OF WORKING FLUIDS OF GAS-PHASE NUCLEAR REACTOR

Moscow TEPLOFIZICHESKIYE SVOYSTVA RABOCHIKH SRED GAZOFAZNOGO YADERNOGO REAKTORA in Russian 1980 (signed to press 1 Aug 80) pp 2, 295

[Annotation and table of contents from book "Thermophysical Properties of Working Fluids of a Gas-Phase Nuclear Reactor", by Viktor Konstantinovich Gryaznov, Igor' L'vovich Iosilevskiy, Yuriy Georgiyevich Krasnikov, Nina Ivanovna Kuznetsova, Vladimir Ivanovich Kucherenko, Galina Borisovna Lappo, Boris Nikolayevich Lomakin, Georgiy Alekseyevich Pavlov, Eduard Yevgen'yevich Son and Vladimir Yevgen'yevich Fortov, Atomizdat, 1800 copies, 304 pages]

[Text] An analysis is made of the current state of the theory of calculation of the thermodynamic, transport and optical properties of gases at high pressures and temperatures, and the state of experimental research in the physics of a non-ideal completely or partly ionized plasma. Numerical calculations are done on the thermophysical properties of plasma of working fluids used in a gas-phase nuclear reactor--alkali metals, uranium, hydrogen and their mixtures--over a wide range of temperatures and pressures.

The book is a review and a reference, and may be of use to specialists working directly with gas-phase reactors, and also to engineers and scientists specializing in different areas of low-temperature plasma physics.

Tables 19, figures 106, references 465.

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INDUSTRIAL TECHNOLOGY

UDC 621.007.52: 681.3

MODELING ON COMPUTER SYSTEM FOR AUTOMATED DESIGN OF INDUSTRIAL ROBOTS

Moscow VESTNIK MASHINOSTROYENIYA in Russian No 5, May 81 pp 7-9

[Article by V. V. Varentsov, candidate of technical sciences, K. F. Kravchenko, engineer and M. I. Poteyev, candidate of technical sciences]

[Text] Industrial robots (PR), as shown by an analysis of their technical characteristics, were designed basically according to four kinematic arragements (Fig. 1). About 75 percent of the PR have the kinematic arrangement shown in Fig. 1a; the remaining three arrangements (Figs. 1b, 1c, 1d) are used in PR 10, 5 and 3 percent respectively of the total number of developed PR [1, 2]. In designing PR using the indicated kinematic arrangements, it is necessary to execute fairly similar operations. In particular, they include: the selection of the kinematic arrangement of the PR on the basis of the given technological process and the indicated working zone to be serviced; the preparation of the dynamic equation for the PR and its drives; the formulation of limitations imposed on the system; the calculation of the basic parameters of the PR mechanism and its drives; the design of the control system of the PR; the optimization of the dynamic, kinematic and precision characteristics of the PR adhering to the imposed limitations; the formation of the layout drawing of the PR, its drives, control system, etc.

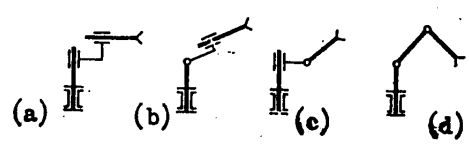

Fig. 1. Kinematic arrangement of industrial robots (PR).

The similarity of these operations in designing the multiplicity of PR for various purposes attests to the urgency of the problem of creating components and systems for an automated design of industrial robots (SAPR PR), that assumes the achievement of all the enumerated operations on a computer with the possibility of a dialogue between the designer and the computer to achieve an optimal result.

Investigations have shown the possibility of creating the SAPR PR. It must represent the totality of subsystems, oriented toward implementing certain design solutions. Each subsystem must contain algorithms for finding the design parameters of the manipulator, the control system and of their own individual units. These include algorithms for determining the parameters of the manipulator's arm, taking into account its elasticity, kinematic characteristics, load etc.; the algorithms for the determination of the parameters of the gripping device, taking into account the properties of the loads being transferred, the time lag, loads etc.; the algorithms for calculating the parameters of the drives, taking into account the combination of their movement etc. A sufficiently developed SAPR PR must also have a bank of stored design solutions to use in similar situations.

On this basis an SAPR PR may be represented in the form of a block diagram (Fig. 2)

Subsystem for selecting a kinematic arrangement (of manipulator, gripping device etc.)

↓

Subsystem for establishing limitations

↓

Subsystem for using the bank of stored data — yes

↓ no

Subsystem for calculating basic parameters of manipulator, arm gripping device, drive etc.

↓

Subsystem for optimizing parameters

↓

Subsystem for imposing limitations on the design parameters, taking into account GOST, materials used, standard parts etc.

↓

Subsystem for forming the drawing of the arrangement ←

Fig. 2. Block diagram of the SAPR PR

Individual subsystems were simulated on the YeS-1030 computer to check the possibility of creating the SAPR PR. The simulation included a calculation of the basic parameters of the mechanism and drives of the manipulator (Fig. 3). Such an algorithm makes it possible to calculate and optimize the basic parameters of the manipulator with a pneumatic drive that provides a minimum cycle time. No provision is made in this algorithm for a dialogue with the computer and for preparing the arrangement drawing. A method of statistical trials (the Monte Carlo method) is used here to find the optimal solution, as well as a systematic approach that implies the consideration of the manipulator mechanism system, taking its drive alone, as well as the determination of the optimal parameters of the manipulator mechanism and its drive under the condition of the combined movements of the links for forward and reverse strokes of the gripping device in accelerating and decelerating modes, taking into account the elasticity of the outgoing link.

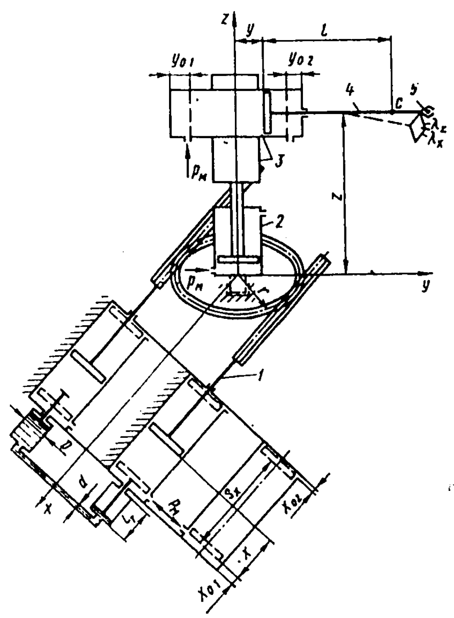

Fig. 3. Kinematic arrangement of a manipulator with a pneumatic drive:

1. rack
2. pneumatic cylinder for lifting the shaft
3. pneumatic cylinder for radial advancement of the outgoing link
4. outgoing link
5. load

Since the motion of the outgoing link of the manipulator along the vertical is not related to other motions in the given arrangement, the drives, therefore, for only turning and advancing are considered in the described part of the simulation algorithm. It is assumed that reciprocal action pneumatic cylinders are used in each of these drives. The rotating part of the manipulator is mounted on a cog wheel; the motion to it from the rod of the turning pneumatic drive is transmitted by a rack or chain.

Taking into account the elasticity of the outgoing link of the manipulator, the dynamic equations of its motion and the characteristics changes in pressures in the chambers of the pneumatic cylinders of the drives may be represented in the nondimensional form as follows:

$$\ddot{\xi} = \frac{\beta_1(\sigma_{\xi 1} - \sigma_{\xi 2}) - \beta_3 \xi - 2\beta_2(\eta + L')\dot{\eta}\dot{\xi}}{1 + \beta_2(\eta + L')^2}; \qquad (1)$$

$$\ddot{\eta} = \beta_5(\sigma_{\eta 1} - \sigma_{\eta 2}) + \beta_6(\eta + L')\dot{\xi}^2 + \beta_7\dot{\eta}; \qquad (2)$$

$$\ddot{\varepsilon} = -\frac{(\eta + L')\ddot{\xi} + \dot{\xi}\dot{\eta} + \beta_{10}\beta_{15}\varepsilon}{\beta_{10}\beta_1\beta_4}; \qquad (3)$$

$$\dot{\sigma}_{\zeta 1} = \frac{k}{\zeta + \zeta_{01}}\left[\beta_4\beta_8\varphi(\sigma_{\zeta 1}) - \sigma_{\zeta 1}\dot{\zeta}\right]; \qquad (4)$$

$$\dot{\sigma}_{\zeta 2} = \frac{k}{\zeta - \zeta_{02}}\left[\beta_4\beta_8\psi(\sigma_{\zeta 2}) - \sigma_{\zeta 2}\dot{\zeta}\right], \qquad (5)$$

where

$$\sigma_{\xi 1} = \frac{p_{\xi 1}}{p_M}, \quad \sigma_{\xi 2} = \frac{p_a}{p_{\xi 2}}, \quad \sigma_{\eta 1} = \frac{p_{\eta 1}}{p_M},$$

$$\sigma_{\eta 2} = \frac{p_a}{p_{\eta 2}}$$

[here $\sigma_{\xi 1}, \sigma_{\xi 2}, \sigma_{\eta 1}, \sigma_{\eta 2}$ -- are pressures in the charge and discharge chambers of the pneumatic cylinders of the drives according to coordinates ξ and η ; $p_{\xi 1}, p_{\xi 2}, p_{\eta 1}, p_{\eta 2}$ -- pressures in the charge and discharge chambers of the pneumatic cylinders of the drives according to coordinates ξ and η (of the shaft turning and arm advancing drives)]

$\xi = \frac{x}{S_x}$ (here x -- coordinate of the piston position of the turning shaft drive; S_x -- piston stroke); $\beta_1 = \frac{F_x p_M}{ACS_x}$ [here F_x -- piston area of shaft turning drive; p_M -- air pressure in the pneumatic main; $A = \frac{u^2}{r^2} m_5$ (u=2 -- coefficient of speed transmission; r -- radius of the dividing circumference of the cog wheel of the shaft; m_5 -- mass of load; $C = \frac{\sum J_z^{2,3} + J_{cz}^4}{m_5}$ ($J_z^{2,3}$ -- moment of inertia of the second and third link with respect to axis z; J_{cz}^4 -- moment of inertia of the fourth link with respect to the vertical axis, passing through the center of mass of the link)]; $\sigma_{\zeta 1} = \frac{p_{\zeta 1}}{p_M}$, $\sigma_{\zeta 2} = \frac{p_{\zeta 2}}{p_M}$ (here $p_{\zeta 1}, p_{\zeta 2}$ -- pressures in charge and discharge chambers of the pneumatic cylinders of the PR drives according to the x and y coordinates; $\beta_2 = \frac{\varkappa S_y^2}{C}$ (here $\varkappa = \frac{m_4 + m_5}{m_5}$; ; m_4 -- mass of arm; Sy -- piston stroke of drive for moving arm); $\beta_3 = \frac{\alpha_1}{AC}$ (here α_1 -- proportionality coefficient); $\eta = \frac{y}{S_y}$ (here y -- coordinate of piston position of drive for moving arm); $L' = \frac{L^*}{S_y}$ (here $L^* = \frac{m_4 l + m_5 L}{m_4 + m_5}$, l -- distance from the axis of the shaft to the center of the mass of the arm; L -- length of the arm) $\beta_5 = \frac{F_y p_M}{S_y (m_4 + m_5)}$ (here F_y -- piston area of drive for moving arm);

$$\beta_6 = \frac{A \varkappa S_x}{m_4 + m_5}, \quad \beta_7 = \frac{\alpha_3}{m_4 + m_5}$$

(here α_3 -- proportionality coefficient); $\varepsilon = \frac{\lambda}{\Delta}$ [here $\lambda = \sqrt{\lambda_x^2 + \lambda_y^2}$ ($\lambda, \lambda_x, \lambda_y$ -- deflection of the center of mass of the arm during its deflection oscillations and its components according to coordinates x and y)] $\Delta = \sqrt{\Delta x^2 + \Delta y^2}$ ($\Delta, \Delta x, \Delta y$ -- given accuracy of positioning and its components along the x and y coordinates)];

$$\beta_{10} = \frac{\Delta x}{S_x}, \quad \beta_{15} = \frac{\tilde{C}}{\varkappa ARS_y} \left[\tilde{C} = \frac{3EJ}{L^3}\right.$$

(here E -- Young's modulus; J -- moment of inertia of the cross section of the arm with respect to its longitudinal axis); R -- universal gas constant];

$$\beta_4 = 755\frac{f_{x1}^{э}}{S_x F_x}, \quad \beta_8 = 755\frac{f_{y1}^{э}}{F_y S_y}$$

(here $f_{x1}^{э}, f_{y1}^{э}$ efficiencies of the inlet hole areas of charge chambers of the pneumatic cylinders of drives for turning the shaft and moving the arm); k = 8.28 coefficient; $\zeta_{01}=\frac{y_{01}}{S_y}$.

$$\zeta_{02} = 1 + \frac{y_{02}}{S_y}$$

(here y_{01}, y_{02} -- harmful volumes of charge and discharge chambers);

$$\varphi(\sigma_{\zeta 1}), \psi(\sigma_{\zeta 2})$$

functions of air consumption;
ξ, η -- nondimensional coordinates x and y; $\zeta=(\xi, \eta)$ -- index

In solving the system of equations (1) - (5) the following values are given: force of gravity of the carried load G, maximal motions S_x and S_y in accordance with coordinates x and y and precision of positioning Δx along axis x.

To be determined are: masses m_3 and m_4 of the pneumatic cylinder and outgoing link, moment of inertia $J_z^{(2,3)}$ of links 2 and 3 of the PR with respect to axis z, moment of inertia $J_{cz}^{(4)}$ of the outgoing link 4 with respect to the axis that passes through the center of the mass, and the design parameters of the drives; piston areas of pneumatic cylinders F_x and F_y; volumes of the charge and discharge chambers $x_{0,1}$; $x_{0,2}$; $y_{0,1}$; $y_{0,2}$; the effective areas of the inlet and outlet holes of the charge and discharge chambers $f_{x,1}^{э}, f_{x,2}^{э}, f_{y,1}^{э}, f_{y,2}^{э}$; the parameters of the hydraulic type braking device: the diameters of the damper piston and of the by-pass tubing D and d; the dynamic viscosity coefficient μ of the braking fluid; the length of the braking section l_T.

Functional limitations are imposed on the solution of the system of equations (1) - (5) as follows:

$$\sigma_{\zeta,1} < 1; \ \sigma_{\zeta,2} > \sigma_a; \tag{6}$$

$$\ddot{\xi}_T \leqslant \frac{1}{y+l}\left(\frac{\Delta x r 3EJ}{m_{4,5}\, l^3} + 2\dot{x}\dot{y}\right); \tag{7}$$

$$\dot{\xi}_{yn} \leqslant \frac{\Delta x r}{u\,(y+l)} \sqrt{\frac{3EJ}{m_{4,5}\, l^3}}; \tag{8}$$

where $\sigma_a = \frac{p_a}{p_M}$; $m_{4,5} = m_4 + m_5$; p_a -- atmospheric pressure.

Condition (6) requires that the pressure in the working chamber of the pneumatic cylinder should not exceed the pressure in the main, while the pressure in the discharge chamber should not exceed the atmospheric pressure. Condition (7) requires that the amplitude of the deflection oscillations of link 1 in the braking mode and the system against the stop should not exceed the value of positioning accuracy Δx of the manipulator (Fig. 3). Condition (8) requires that the amplitude of deflection oscillations of link 1, when it is against the stop, should not exceed the value of the precision accuracy of the manipulator.

In selecting optimal system parameters, given the initial values of β_i, the system of equations (1) - (5) is solved in the accelerating mode for forward and reverse motions of the gripping device. A combination of system parameters is found by the method of statistical trials that would provide a minimal time T_p of the manipulator motion; the realization of this time is achieved by selecting the instants of connecting the drives. Considering the motion of the system in the braking mode, and solving the system of equation (1) - (5), taking into account limitations (6) - (8), parameters of the braking devices are found that provide minimal time T_T of braking. The parameters obtained of the manipulator mechanism, its drives and braking devices provide for the minimum time of passage between end positions of the system $T_{min} = T_p + T_T$.

As a result of calculations for the initial data G= 6.3H; S_x = 0.079 m; S_y=0.3 m; $\Delta x = \pm 0.025$ mm, we obtained parameters summarized in the Table. They provide passage time T_p = 1.6 seconds at the following instants of drive connections: for the forward movement of the system, the drive for advancing the arm is 0.3 seconds later than the connection of the drive for turning the shaft; for the reverse movement of the system, the connection of the drive for turning the shaft is 0.2 seconds later than the connection of the drive for advancing the arm.

Table

Parameters for the turning drive (in the numerator) and the drive for radial motion (in the denominator)

$\frac{F_{x,1}}{F_{y,1}}$, cm²	$\frac{F_{x,2}}{F_{y,2}}$, cm²	$\frac{D:d}{—}$	$\frac{f^{э}_{x,1}}{f^{э}_{y,1}}$, cm²	$\frac{f^{э}_{x,2}}{f^{э}_{y,2}}$, cm²	$\frac{x_{0,1}}{y_{0,1}}$, cm³	$\frac{x_{0,2}}{y_{0,2}}$, cm³	$\frac{\mu, \text{Pa sec.}}{—}$
$\frac{13}{6,5}$	$\frac{13}{5,36}$	$\frac{14,5}{—}$	$\frac{0,11}{0,0357}$	$\frac{0,11}{0,0314}$	$\frac{9,1}{0,195}$	$\frac{9,1}{0,161}$	$\frac{0,02}{—}$

Note. Index 1 corresponds to the charge chamber, index 2 -- to the discharge chamber.

In order to check that the parameters found are optimal, a manipulator that has the characteristics shown in the Table was designed and manufactured. Its tests indicated that the passage time between extreme positions was 1.65 seconds. The instants of connecting the drives for turning the shaft and advancing the arm were changed many times in the tests. Other parameters of the manipulator, in particular

$$f^{э}_{x,1},\ f^{э}_{x,2},\ f^{э}_{y,1},\ f^{э}_{y,2},\ \frac{D}{d},$$

were also changed and led to a reduction in passage time T_p by more than 5 percent. This confirms the optimality of the parameters found and the authenticity of the algorithm for their determination.

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2291
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CLASSIFICATION AND TECHNOLOGICAL CRITERIA FOR CONTROLLING INDUSTRIAL ROBOTS

Moscow VESTNIK MASHINOSTROYENIYA in Russian No 5, May 81 pp 3-6

[Article by A. A. Panov, candidate of technical sciences]

[Text] Industrial robots [PR] have rapid readjustability. This PR property is characterized by the following: method and degree of accessibility of programing the task on moving the parts, and the number and type of instructions; the cost of programing the task on moving the parts, changing the programs and readjustment, as well as requirements for the automatic control hardware of the programing and the skill of the service personnel; reprocessable program structures, including processing signals from the processing programs and sensors; operation modes, the possibility of servicing and connecting into the control system; the number and volume of assigned programs on moving parts and the sequence of its finishing off; performance monitoring and diagnostics fundtions.

Striving to adapt the principle of PR action to human ability in processing data, recognizing parts and executing motions leads to the necessity of classifying them according to the degree of "intellectuality" into three generations. The most important technological criteria that characterize these generations -- technological criteria of the technique of control and processing data are shown in Table 1.

At present, PR of the first generation are the most widely used; in the future, the utilization of program control systems will make it possible to find the solution for the prevailing number of problems on moving processed parts, as well as to change the functions of the control, storing and processing of data.

The increasing costs of using robots in production are concerned primarily with the automatic control hardware of the PR.

The following requirements are demanded of the PR control system: control of the movements of the gripping device (tool) in the working zone in accordance with a given program for moving the parts; programing for various working modes; providing for interaction with the technological equipment when it is used in a complex automated system; executing functions of interoperational monitoring.

The origination, in practice, of various control systems is due to the following: the multiplicity of PR design versions that have original features; technical progress in the development of automatic control hardware using microelectronics;

expansion of the PR utilization field in machining and assembly; dependence of the technical characteristic on the type and degree of free movements of the PR, the type of system control and programing method; trends in the development of automation of machining and assembly.

Program manipulators may be classified in the following manner according to types of control systems and conditions of application.

1. PR program'... PR are considered programable, whose readjustment is implemented by means of composing a corresponding movement control and introducing this program into the control system. A rigidly programable manipulator has a constant program in the control system; changing it is done by an adjuster.

2. Control systems with rigid programing may be positional and loop. In control systems with rigid programing the cycles of motion and signal exchanges are constant. Such control is used basically in simple PR of the cycle loaders type. Positional control systems make it possible to produce point-by-point positioning in the working space where the grip(tool) moves; a functional movement relationship, predetermined by the control system, exists between these points.

The loop control systems achieve a functional relationship in the movement of axes; the grips (tools) trace out strictly defined loop lines in the working zone presented by the program. The use of this or another loop control system is determined by the mobility of the structurally preset degrees of freedom of movement of the axis.

Two versions of loop control are used in practice, differing in the programing of the data processing within the system with the direction of the motion loop line called multipoint control; the programing is implemented by "motion along the loop." In cases where the motion loops are produced with complex motion shapes, for example, when applying varnish-paint coatings with a continuous direction of the loop line, the programing is implemented, as a rule, by "starting-up and storing" and, in reproducing the program, the motion loop between the points is obtained by interpolation.

3. Nondigital and digital programed control systsms. Nondigital programed control systems are most frequently used in combination with PR in which the preset positions are fixed by means of regulated stops, cams etc. The flexibility of such a system depends to a great degree on the type and capacity of the memory. Numerical programed control systems (ChPU) are reading devices using numerical values as signs of the path data and symbols for instructions, as well as coded data. The positions of a preset path in the form of numerical data are a component of the motion program; positioning of the path may be programed with respect to the given values of the positions, the number of various positions along each axis, the frequency of origination of certain given positions in the motion cycle and, if necessary, may be easily corrected. Due to data coding and the use of a corresponding memory, a great number of program steps, instructions and data may be achieved and reprocessed.

4. Special and universal control systems. If a control system is designed for a certain type of PR or problems of the motion of a type of part, it represents a special solution.

Table 1

PR generation	Functions: control	Functions: data processing
First -- PR with program control	Program cycle, correlation of instructions, control of movements by means of program instructions, by comparing the given and actual values, special functions depending upon the type of control convenience of servicing and field of application	Control instructions are received strictly according to dimensions given in the movement program. The achievement of the control functions is not monitored
Second -- "sensing," "seeing" PR	Expansion of functions for executing tasks; position correction, determination of position and shape; recognizing the moved material, the environment etc.; self-monitoring; processing condition. Control movement using comparison of criteria	Additional to functions of the first generation: obtain signals from sensors, enter the program cycle according to a previously programed algorithm
Third -- "thinking" PR	Simulating object (part) working movement etc. by a "training stage." Movement program plotted by the control system on the basis of a given model of the part and certain criteria of optimization, as well as the model of the environment. Scan control functions by comparing with model.	Start-up on the basis of solutions which have been adopted based on the process and environment and the ability of self-teaching. Adaptive principle of action.

Table 1 Continued on next page

Table 1 (continued)

PR generation	Forming assignment for shifting	Typical criteria for obtaining data
First -- PR with programed control	Program for shifting	Data of path from sensor of positions or path. Analysis of position of clamp and location is determined indirectly.
Second -- "sensing," "seeing" PR	Same	Use additional tactile, optical and visual sensors (especially for the analysis of condition and recognition of the object)
Third -- "thinking" PR	Working task ("target program)	Use of a great number of sensors. Primarily visual sensors (remote control equipment in combination with sensors with various principles of operation) for direct recognition of parts, positions, conditions, etc.

Control systems intended to solve multipurpose control problems should be called universal systems; they are implemented primarily by developing and storing special control functions, i.e., by developing means of programing. Under concrete conditions, the development of special software may also be required.

In creating special control systems, the designer determines, based on the required control functions, the control volume, convenience in servicing, as well as production possibilities etc., the necessary groups of structural components and units. Thus, for simple cases, pneumatic control systems are specified. New developments of special ChPU systems should include minicomputers.

Universal control systems should have built-in mini- or microcomputers or be connected to a central computer.

Programed control systems in principle are not meant for processing paths and motions on a digital basis as compared to freely programed ChPU computer systems.

Programing problems on moving parts span all actions for preparing motion programs and introducing them into the control system. The motion program contains all the control functions intended to implement the concrete problem of PR control.

The control system must provide economical programing according to the requirements for their application and readjustability when complying with special features of motion. The basic requirements must include convenience for the consumer and, if possible, the least laborious preparation of the program and the easiest correctability and optimization. Methods for data introduction and programing should be classified according to the place of programing and the type of control system.

Programing methods are selected according to conditions for the use of PR and, at the same time, must satisfy the requirements mentioned above. Programing at the site of use is very convenient because the programer usually does not need special knowledge, and special auxiliary means for programing are not required. However, in a number of cases of PR use, especially for comprehensive automation of assembly, programing outside the boundaries of the work position (external programing) is required. In this case, the program is made up when preparing for work and is recorded on the program carrier assigned to the control system. Such programing must be used only in those cases where programing at the site is impossible or inexpedient (for example, the motion task due to its complexity, or switching with the process at the working position does not lend itself to description; implementing control functions which may be programed only when preparing for work; or the PR receives control data from central control).

When the PR is programed at the worker position, after the motion cycle is determined, all the necessary data for designing the program is fed directly into the control system.

Programing by motion cycles of parts is limited by the program cycle, i.e., the motion cycle is designed as a sequence of instructions for direct control of motors of axes and grips, inputs and outputs etc., and is implemented manually. The motion paths are adjusted by means of stops or position sensors directly on the PR; therefore, they are not a component part of the motion program.

Table 2

Programing place	Method for data input	Methods of Programing	Type of control system: Special — Nondigital position control	Special — Digital position control	Special — Loop program control	Universal — Programed control systems	Universal — ChPU computer systems
At site of use	Manual	Programing by cycles	A	-	-	-	-
	Self-teaching	Start-up and store	A (B)	A, B (C, D, E)	A, B, C (D, E)	-	A,B,C,D,E
		Moving along loop	-	-	A, B, C, (D,E)	-	A,B,C,D,E
Outside of site of use (external programing)	Manual	Mechanical program carrier	A	-	-	-	-
	Corresponding to a control system for data coding on program carriers	A, (B)	A,B (C,D,E)	-	-	-	-
	Machine	Specialized program language	-	-	-	A,(B,C), D,E	A,B,C,D,E
		Universal program language	-	-	-	A,(B,C) D,E	A,B,C,D,E

Note. Realized components: A -- program cycle; B -- path conditions; C -- motion conditions; D -- logic solutions; E -- monitoring, diagnostics. Some components in parenthesis indicate their limited realization.

The input of data into the program memory may be done manually by adjusting, connecting, pushing an operator's pushbutton etc. with the motion reading exhibited on the control panel, and automatically by functional switches due to which data is transferred to the addressed places in the memory, i.e., by the self-teaching method. A PR programed at the site whose program also includes path conditions in the component parts of the program, along with the motion cycle, is called freely programable.

Specified motions are adjusted manually in the required sequence. In this case, the motion cycle and the path conditions are stored step-by-step. The PR position is determined by the path sensors which feed data on the path.

To produce motion along the loop, the grip(tool) is directed manually along the desired motion loop during the adjustment mode and the trajectory is automatically stored as a sequence of points. The number of points stored per unit of time must be such that a sufficiently precise reproduction of the programed motion line is made automatically.

As a rule, programing requires special documentation, equipment and devices for preparing, converting, checking, correcting programs and transferring data to the memory. All tasks on motion data must be compiled fully at the stage of work preparation.

The improvement of external PR programing methods is achieved by a similar method of programing machine tools with ChPU.

The use of mechanical program carriers cam discs, rigidly secured by cams, is a special case of external programing only for simple PR. When readjusting, the program carrier is replaced or readjusted anew.

An approximate control system using coded data may be implemented by special coding on program carriers, punched tape, diode matrices or semiconductor memories.

Specialized program languages which are subdivided into simple languages oriented toward a specific problem and languages of a higher order are used; they differ in complexity of control functions and in the volume of reprocessed data.

The motion program recorded in the corresponding language, up to the time that it may be placed in the program memory, must be translated into the elementary control language. There is also the possibility of converting instructions in specialized languages directly into control instructions (by means of interpreters). Problem oriented languages are used primarily in connection with universal control systems for the solution of complex problems for multimachine tool servicing and assembly operations.

In external programing, problems on the motion of an industrial robot on the site of its use, mixed programing is also utilized.

Conclusions

An analysis of control systems and programing methods for PR makes it possible to establish a relationship between the control system type, methods of programing, the control application field and the content of motion programs. Based on these general criteria, it is possible to classify control systems. In practice, some control system criteria may be found in combination when only a part of the degrees of motion belong to positioning. The type of control system is determined according to the optimal variation of a positioning robot with a given accuracy and quick action which may be achieved by mathematical simulation of its motions.

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TECHNOLOGICAL MHD INSTALLATIONS AND PROCESSES

Kiev TEKHNOLOGICHESKIYE MGD USTANOVKI I PROTSESSY in Russian 1980 (signed to press 21 Oct 80) pp 22, 189-190

[Annotation and table of contents from book "Technological MHD Installations and Processes", by Anatoliy Fedorovich Kolesnichenko, Institute of Electrodynamics, UkSSR Academy of Sciences, Izdatel'stvo "Naukova dumka", 1000 copies, 192 pages]

[Text] This monograph examines qualitative and spatial transformations in bounded regions filled with an electrically conductive liquid or gas and placed in physical fields: electric, magnetic and gravitational. Included among such transformations are changes in the form of energy and energy transfer--conversion of electromagnetic energy to heat and mechanical work, transfer of heat and mass of the conductive fluid under the influence of various combinations of electromagnetic, capillary and thermoconductive factors. Resulta of theoretical and experimental research are given on a new class of MHD phenomena--capillary effects that arise during arc and induction treatment of metals. New techniques for processing alloys are described that are based on the use of MHD effects.

The book is intended for scientists, engineers and technicians interested in the development and use of MHD installations.

Figures 82, table 1, references 119.

Contents

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ANTIFRICTION COATINGS ON TITANIUM ALLOYS PRODUCTS

Moscow VESTNIK MASHINOSTROYENIYA in Russian No 5, May 81 p 47

[Article by V. I. Latatuyev, cand. of tech. sciences and G. N. Ganay, engineer]

[Text] It is well known [1-5] that it is extremely difficult to deposit good bonding coatings on titanium and its alloys due to the formation of oxide film on their surfaces.

When the oxide film is removed by etching by using a mixture of nitric and hydrofluoric acids there is observed a strong hydrogen absorption on the titanium surface due to the formation of a hydride (0.06% and more of hydrogen in the outer layers).

The presence of the hydride film reduces sharply the adhesion of the coating to the titanium and deteriorates the strength characteristics, i.e., increases the brittleness of the metal. It is proposed [2] that annealing (400-500°C) removes the hydrogen, and the adhesive properties of the electroplating are improved.

A new and very promising method is one for depositing nickel-phosphorus coatings by chemical nickel plating (KhN) with acid and alkali (ammonium) solutions [6-9] on the titanium and its alloys.

The phosphorus content (2-15%) in the coating imparts specific properties: great hardness and higher antifriction properties which makes it possible to utilize KhN to solve the problem of screwing parts made of titanium alloys together when they are subjected to high loads. Tests were made on samples of the VTZ-1 and the VT-20 alloys. Preliminary tests on the KhN in solutions of well-known compounds (GOST 9.047-75) indicated that titanium alloys either cannot be nickel plated or form a coating with poor adhesion.

On the basis of recommendations [7-8], investigations were made on the preparation of the surface of the titanium alloys before chemical nickel plating (before each chemical operation the parts were washed thoroughly in running water for 2 to 3 minutes) by first degreasing in organic solvents (gasoline, dichloroethane etc.) for 5 to 10 minutes, then in an alkaline solution for 5 to 10 minutes at a temperature of 60 to 70°C.

The oxide film was removed in solutions of sulfuric or hydrochloric acids at various temperatures and times. The best results were obtained by pickling in concentrated hydrochloric acid at $50 \pm 5^{\circ}C$ for 8 to 10 minutes (2 to 3 micrometers are etched off). Too lengthy etching destroys the surface due to autocatalysis. For example, 15 to 20 micrometers are etched off.

The most critical operation is the activation of the titanium alloy surface -- the deposition of a thin film (approximately 0.1 micrometer) of nickel that catalyzes the KhN process.

A solution* consisting of 150 to 250 grams/liter of nickel sulfate and 100 to 120 milliliters/liter of hydrochloric acid at 20 to $25^{\circ}C$ was taken as a basis. The activation time was 5 to 10 seconds.

After thorough washing in running water, the samples were immediately immersed in the KhN solution. In this case, solution "D" was used which consisted of 20 to 30 grams/liter of nickel sulfate, 20 to 25 grams/liter of sodium hypophosphite, 10 to 15 grams/liter of sodium acetate, 0.003 grams/liter of thiourea and 2 milliliters/liter of acetic acid, at a temperature of $90 \pm 2^{\circ}C$ and a speed of 20 to 25 micrometers/hour.

For removing hydrogen and increasing adhesion, the samples were heat-treated in air at $400^{\circ}C$ for 2 hours. The adhesion of the plating met the GOST requirements.

The developed technology solves the problem of depositing an antiscuff nickel-phosphorus plating on threaded parts made of titanium alloys.

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CARRYING CAPACITY OF WORKING BLADES OF GAS-TURBINE ENGINES UNDER VIBRATION LOADS

Kiev NESUSHCHAYA SPOSOBNOST' RABOCHIKH LOPATOK GAZOTURBINNYKH DVIGATELEY PRI VIBRATSIONNYKH NAGRUZHENIYAKH in Russian 1981 (signed to press 29 Dec 80) pp 4, 313-314

[Annotation and table of contents from book "Carrying Capacity of Working Blades of Gas-Turbine Engines Under Vibration Loads", by Valeriy Trofimovich Troshchenko, academician, UkSSR Academy of Sciences, Valentin Vladimirovich Matveyev, Boris Alekseyevich Gryaznov, Sergey Semenovich Gorodetskiy and Anatoliy Beynaminovich Roytman, Institute of Strength Problems, UkSSR Academy of Sciences, Izdatel'stvo "Naukova dumka", 1000 copies, 316 pages]

[Text] This monograph gives the procedure and results of investigation of fatigue of high-temperature alloys subjected to complex thermomechanical loading, and also the fatigue of working blades of turbines and compressors in gas-turbine engines with consideration of their changes during utilization. A method is described for modeling experimental loads for turbine blades (experimental facilities). Data are given on the damping properties of turbine blade materials, lock joints of compressor and turbine blades, and also on the damping capability of working blades that is due to aerodynamic drag of the ambient flow and structural hysteresis with consideration of operating conditions. An examination is made of methods of reducing the dynamic stress on blades by using structural damping and dissipation of the energy of vibrations in the material. Recommendations are made for increasing the carrying capacity of the turbine blades in gas-turbine engines.

For scientists, engineers and technicians dealing with problems of developing and using gas-turbine engines.

Figures 245, tables 38, references 209.

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