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CONTENTS

PLENARY SESSION

Ensuring Safety and Enhancement of Efficiency of Russian Nuclear Power

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The paper presents the experience related to Russian NPPs safe operation and the main results in 2013. As of 01.01.2014, there were 33 power units in operation with total capacity of 25.2 GWe. Total electricity generated by nuclear power plants in 2013 was equal to 172.2 bln kW·h (about 16.5 % of total electricity generation in Russia). Load Factor (LF) was 77.9% in average.

The results obtained have been supported by activities focused on enhancement of operational safety and reliability of power units, maintenance improvement, life extension of the operating power units, and enhancement of the management structure.

The paper reviews activities that have contributed to NPPs performance enhancement, provides trends of operational events and radiation safety performance at Russian NPPs, and describes Rosenergoatom nuclear generation goals and technical & economic objectives set for 2014.

The main targets of the Rosenergoatom production programme for 2014 are:

- generation target – 172.6 bln kW·h;
- planned LF value – 75.9 %.

Special attention is given in the paper to the Rosenergoatom activities aimed at operating life extension (OLE) of Russian nuclear power units in operation. As of 01.01.2014, the OLE activities have been accomplished for 19 power units. These activities are going on at 10 power units.

The paper provides information on the schedule for transition of VVER-1000 power units to operation at the uprated power level of 104% Nnom.

The main areas of activities aimed at implementation of the target-oriented modernization of nuclear power units with the goal to increase their safety and reliability are briefly described, and the results are presented.

There is provided information in the paper as regard to works related to the recovery of RBMK operating life characteristics.

The milestones and basic results of activities towards implementation of the nuclear power unit decommissioning concept are given.

In addition, the paper comprises the status information on NPPs currently under construction and information about planned new nuclear builds in the time period up to 2030.

The paper states the conclusions drawn based on lessons learned from the accident at Fukushima NPP and briefly describes some measures taken at Russian NPPs in order to exclude any progression of such accidents.

Information related to cooperation with WANO is given, in particular, as regard to establishment of the WANO-MC Regional Crisis Center, WANO peer reviews of Rosenergoatom and its nuclear plants.

The paper informs about OSART missions conducted at NPPs. In particular, the results of OSART Follow-up visit to Smolensk NPP are described.

The main conclusions made in the paper are as follows:

- the management and staff of Rosenergoatom are ensuring safe operation of Russian NPPs;
- Rosenergoatom has proven its capabilities to respond in adequate, prompt and efficient manner to new challenges in the NPP operational safety area;
- the system for ensuring Russian NPPs safety, which is based on the Defense-In-Depth concept, does not require any revision, and does constitute the technical policy basis for Rosenergoatom as the operating organization.
- Rosenergoatom fully executes its role and responsibility of an operating organization as stipulated by laws of the Russian Federation in the field of the use of atomic energy.

50 years of Novovoronezh NPP safe operation

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The report covers information about Novovoronezh NPP safe operation experience from Unit 1 (210 MW capacity) start up at the end of September 1964 which reached rated capacity in December 1964. Novovoronezh NPP Unit 1 is water cooled water moderated reactor. Unit 1 operational experience enabled to elaborate design and engineering solutions led to improvement of power flux distribution along the reactor core using boron control.

In 1969 Novovoronezh NPP Unit 2 of rated capacity 365 MW was put into operation using boron shim based on new investigations and engineering development. In December 1971 Unit 3 of 440 MW capacity started its operation and Novovoronezh NPP became the most powerful nuclear power plant not only in the Soviet Union but in Europe: its capacity amounted to 1015 MW. Unit 4 of 440 MW capacity started generating in 1972. At the end of the first quarter 1973 full rated capacity was reached. In 1980 Unit 5, the pilot unit with VVER-1000 with two 500 MW turbines, was put into operation.

The main focus points of the report are upgrade and design solutions for equipment and system improvement implemented in the process of Novovoronezh NPP operation:

- Evolution of fuel element and fuel assembly design (starting from fuel element of Unit 1 to fuel elements and fuel elements with gadolinium absorber of Unit 5);
- Evolution of Reactor Control and Protection System control rods (peculiarities of 1, 2, 3-4 and 5 Units control rods, change in control rods groups sets at Unit 3 and 5 aimed at optimization of power flux distribution);
- Evolution of fuel loading design (from out-in-in to in-in-out);
- Novovoronezh NPP "hot cell" investigations (support in implementation of new fuel types);
- Evolution of nuclear fuel management systems and implementation of system for VVER-440 leaky spent fuel assembly management;
- Unit 3 and 4 reactor core cooling improvement;
- Improvement of diagnostics techniques and in-service inspection methods in detection of different flaw types and modern equipment use.

The reports includes information on Units 3, 4 and 5 modernization and service life time extension beyond 30 years with justification of remaining life of power units' safety related components.

Scientific and Engineering Task Solution for Fast Breeder Reactor Operation: from BN-600 towards BN-800

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Long-term operation of BN-600 power unit and results of scientific research work conducted under field conditions are of primary importance both for safety enhancement and perfection of operating NPP operation and for selection and justification of design and circuitry solutions when designing power units of the new generations. Under the field conditions the compliance of actual and design characteristics and equipment operation modes are tested, and moreover, new effects of full-size facility can also be discovered, which cannot be discovered during test on bench and facilities of smaller size, and possibilities to use equipment reserve as well.

Power unit with BN-600 has been successfully operated since 1980. Within the period of its operation the tasks set when it was developed were solved as follow:

- Long lasting life time tests of heavy equipment operating in sodium environment, sodium technology application;
- Workout and updating of operation modes, monitoring and control systems of the power unit.

Main goals of BN-800 design:

- Restoration of experience on development and manufacturing of heavy equipment (RPV, steam generator);

- Workout of closed fuel cycle technologies, in the first turn, application of mixed uranium-plutonium fuel.

During 34 years of BN-600 operation the main task was fulfilled – operation of a high-capacity power unit with fast sodium reactor and sodium steam generators was mastered. The achieved level of operation reliability is featured by an annual load factor, which was at the level ~ 80 % for the last years of operation. During years of operation the other important tasks were also solved:

- Tests of 421 pilot FAs were conducted for investigation of structural materials and constructions of various type;
- Fuel burnout was increased from 7 % to 12 % t.a.;
- Additional method of reactor heat power distribution among loops was implemented;
- Sector-type system CTM (SSCTM) was implemented intended for detection of FAs with leaking fuel elements to detect the fission products (sources of delayed neutrons) in primary circuit coolant;
- Specific features of heat hydraulics of sodium circuits, sodium technology and updating and optimization of heat hydraulic processes and modes were investigated.

Results of the scientific and research work conducted address both safety enhancement of operating power units and selection and justification of design and circuitry approaches when designing and operation of fast breeder sodium reactors of the new generations.

Technologies of repair and replacement of reactor and steam generator heavy equipment have been mastered (72 modules of SG, 3 LPCs, 6 FPs, 1 AFP).

Long lasting life time tests of heavy equipment operating in sodium have been conducted.

The most important result of operation is justification of construction of new power units with fast breeder reactors (BN-800, BN-K).

Restoration of long-term performance of graphite structure at power unit № 1 of Leningrad NPP

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Since 2011 through 2012 the accelerated development of deformation of graphite structure and distortion of reactor channels were noticed at power units of the first generation with RBMK type reactors resulting from radiation thermal damage of graphite structure. At the same time no fragmentation or loss of carrying capacity of graphite units were observed.

Mechanism of deformation and change of integrity of graphite units and structure is related to occurrence and progressing of radiation defects in the lattice of graphite and change of stress and strain state of graphite units during reactor operation.

The below tasks were identified as the main tasks for recovery of the long-term performance of graphite structure:

- achievement of the values of deflections of fuel channels and operating channels of control and protection system (CPS CR) less than 50 mm;
- reduction of rate of deflections grow when operation to less than 15 mm/year.

In order to solve these tasks the technology of operability restoration (long-term performance) of graphite structure by method of longitudinal cutting of restricted number of graphite columns was developed and implemented. For better effectiveness of deflection reduction after graphite columns cutting, it was supposed to use tensioner- devices that will enable to move the most deformed sections of graphite columns from the periphery towards the center. At the same time the options of cutting the graphite units into 2 and 4 parts were considered. Cutting into 4 parts in diagonal rows will enable to reduce the deflection immediately in two directions.

To implement such technology the following measures were performed:

- technological process «Recovery of long-term performance of RP elements at power unit №1 of Leningrad NPP» TP 12.157.00.00 was developed;
- organization structure for the project management was set up on the level of Corporation Rosatom and Leningrad NPP;
- equipment for performance of work on long-term performance restoration was designed and supplied on Leningrad NPP;
- nuclear, radiation safety and safety engineering were justified for work performance on long-term performance restoration;

The technique of repair-recovery activities incorporates the following main operations (without account of preparatory operations and procedures of RP process flowchart recovery):

1. In each zone of recovery activities the following groups of cells shall be determined in rows:

- cells wherein the current monitoring of curvature is going on;
- cells wherein graphite columns cutting is done;
- cells for force impact.

2. Deflection measurement is performed on the cells.

3. To enable graphite units cutting the dismantle of collars, return bends, bellows and fuel channels and segmental compensators are performed, if needed.

4. Graphite column is inspected with the help of TV system to identify the state of butt joints of graphite units, sizes and azimuthal location of cracks.

5. If required, clearances shall be closed in vertical joints of graphite units.

6. Longitudinal cutting of graphite units is performed and cutting products are removed.

7. After longitudinal cutting the inspection of proper cutting shall be performed, «adequate cut» of graphite units, availability of graphite fragments in the cell (wedges), coupling of cuts. Should graphite fragments are available in the cell the cell must be cleaned and fragments removed. Removal of graphite fragments from the cell by means of longitudinal cutting of fragments by millers of different diameters is allowed.

8. In the cells for force impact the process channels are dried, tensioner-devices are mounted and force impact is applied to the cells.

9. Deflections are measured on the cells for inspection.

10. Fragmented graphite columns shall be calibrated for fuel channel mounting and the cell inspected and cleaned.

11. Fuel channels, bellows, return bends and collars are mounted.

12. After completion of work on the zone cells they are loaded into FA.

Below are the main results of work on long-term performance recovery at power unit № 1 of Leningrad NPP:

Implementation of the complex of measures for recovery of long-term performance of reactor plant enables to continue the operation of power unit №1, Leningrad NPP, at the acceptable safety level.

Target values of deflections of fuel channels and control and protection system working channels set up in the "Regulation (typical program) of in-service inspection of process channels, CPS channels and graphite structure of RBMK -1000 4.064 ПМ" were achieved.

Complex of unique technical means for performance of work on graphite structure treatment and recovery of deflection of fuel channels and CPS channels was established.

Technological concept established at LenNPP was successfully used at power unit №2 of Kursk NPP and can be applied later at all RBMKs-1000.

Operation of power units of Balakovo NPP using 18-month fuel cycle

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Results of experimental-industrial operation of Balakovo NPP when transfer to 18- month fuel cycle, scope of issued justifying documents, results of probabilistic safety assessment as well as the list of accomplished activities, updates and problems when transfer to 18-month fuel cycle are presented in the report.

Economic and technical indices of Balakovo NPP reached in 2013 as well as anticipated indices after transfer to stationary 18-month fuel cycle is completed are demonstrated in the form of tables and schematics.

Comparison of economic and technical indices of 12-month and 18-month fuel cycles prove the economic expediency of transfer to 18- month fuel cycle, at the same time it is shown that the fuel constituent of electric power production cost grows up, but, nevertheless, due to decrease of non-variable expenses constituent the total cost of electric power reduces and annual income increases.

Based on the probabilistic safety assessment (PSA) results presented in the report the conclusion is made that transfer to 18-month fuel cycle is justified and will result in safety level increase. Decrease of the number of damages of the core for 18-month fuel cycles compared to 12-month fuel cycle can prove the above.

The report points out the advisability of Balakovo NPP operation in 18-month fuel cycle, particularly, under the conditions of global updates that require the uniform distribution of large financial expenses and under the condition of manpower lack. Under such circumstances it becomes possible to vary the duration of fuel cycles within the range of 260 to 510 effective days and, thus, exclude PPM conduct for various power units at the same time inside a single NPP.

The report demonstrates reliability indices of Balakovo NPP equipment and information about defects registered when transfer to 18-month fuel cycle. Operation statistical data prove that at the initial period of fuel cycle when equipment run-in after the unit start up a relatively larger number of failures is observed than during the remaining period but, on the whole, the curve of failure intensity has no tendency towards grow up at fuel cycle interval more than 12 months.

Information proving safe, reliable and effective operation of nuclear fuel is incorporated in the report. Values of primary circuit coolant reactivity for benchmark radionuclides, data on FAs geometrical stability at Balakovo power units presented in the report prove that no worsening of nuclear fuel reliability when operation in fuel cycles beyond 12 months is observed.

In the conclusion the safety indices reached at Balakovo NPP power units in 2013 when use of 18-month fuel cycle are discussed; the outcome is drawn that the good operation experience in the extended fuel cycles gained till now can be used for updating and introducing relevant modifications in the Federal Codes.

Development and upgrade of nuclear fuel for the cores of nuclear facilities

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JSC «TVEL» is the Russian nuclear fuel supplier ensuring needs of 17% of the world market and 45% of the world market for uranium enrichment.

Among the nuclear fuel consumers of JSC «TVEL» there are research, transportation and industrial reactors and foreign NPPs with VVER, RWR and BWR. Within the frames of Rosatom fuel company JSC «TVEL» incorporates scientific-technical block (VNIINM, NNKTS, Centrotech-SPb, OKB-Nizhny Novgorod), manufacturing block of GC (KPZ, Tochmash, UZGTS, Uralpribor), conversion and enrichment (AEHK, UEHK, SCP, ECK) and nuclear fuel fabrication block (MSZ, NPCC, CHMZ and MZP).

Below are the main tasks of corporation «TVEL» while products manufacturing:

- meeting the requirements of the consumers to the in-service characteristics and improvement of technical and economic indices of nuclear fuel;
- ensuring the needed level of safety when use and manufacturing of nuclear fuel;
- enhancement of competitiveness and expansion of market for products sale.

The above goals can be reached by solving the below tasks:

1. Development the design of FA that will meet the requirements of safe, reliable and cost effective operation;
2. Development of new and improvement of existing fuel composites and structural materials;
3. Updating of technologies of fabrication of fuel, structural materials and fuel assemblies;
4. Reduction of conservatism of cores justification, development of design codes and techniques.

Within the recent years new types of nuclear fuel for main types of power reactors have been developed, implemented and successfully operated at the NPPs involving the leading companies of the nuclear branch and large volume of SRSs, EDSs and TDs:

- for VVER-1000 – FA-PLUS and FA-2M and their modifications;
- for VVER-440 – Control channels and FAs of the 2-nd generation with higher enrichment of fuel and – control channels-3,

Technical-and-economic characteristics (fuel burnout, duration of operation and fuel cycles etc.) are on the level of the world manufacturers of nuclear fuel for power reactors.

Further activity on development and updating of the design of fuel elements and fuel assemblies, structural and fuel materials is carried out that enables to provide the effective fuel to the consumers.

Nuclear Power Industry Safety Issues

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To ensure prevention of a catastrophe it is necessary to understand its essence. Great earthquakes, tsunamis result in a heavy toll of human lives: hundreds, thousands, dozens of thousands. This number in case of a typhoon is limited by tens of lives. Explosions in mines lead to tens of lives lost per one accident, and tens of thousands – annually. And the losses may be at level of thousand lives in case of accidents at chemical plants like it was in Bhopal.

The heaviest accidents at nuclear plants can be counted on the fingers of one hand, and any of these accidents (Three Mile Island, Chernobyl) did not result in losses of lives of general public members. Recent Fukushima accident has gone without any victims at all; while the Great East Japan Earthquake together with the tsunami had the toll of 20 000 human lives in 2011. However, number of references in print media to that earthquake is much less than to the Fukushima accident. Hence, the level of human losses in case of nuclear accidents is not the point: there were zero losses while the perception is such as it was a disaster by orders of magnitude heavier than the largest tsunami.

It is hard to escape the conclusion: this vision of nuclear accident consequences exceeding by orders of magnitude the real damage is inseparably connected with an inadequate perception of radiation hazards by vast majority of the society.

First of all, it is high time to formulate more precisely the primary goal of ensuring nuclear power safety as follows: prevention of impacts on society, which may not result in a loss of lives or health.

In order to achieve this goal it is necessary to solve two complementary tasks. The first one is to reduce probability of a severe accident to an acceptable level while providing for national and international emergency response systems implementation. Essential elements of such systems should be national centers for scientific and engineering support, which provide support to decision making and inform mass media at the state level. In Russia, the emergency response system has been developed to a high standard. There will be presented an example of its response to Fukushima accident.

In the same time, there is a second task that, to our opinion, is by far not realized enough. Namely, it is education of the population bringing it to a civilized state of mind in this area. From our viewpoint, it is the state government that shall take care of the population education. In fact, such efforts should be made not by a single state, but in coordination with other states. If a country has decided to embark on nuclear power development, and the government has given a green light to this, then the government shall accept the responsibility for population protection in this regard. International

Convention on Nuclear Safety should be amended by inclusion of a statement regarding education in the area of real danger of radiation.

Provided that the population education issue is resolved, a damage of an accident would be limited solely by costs of the plant itself and relevant remediation works. It is easy to calculate that such damage could be fully covered in case when all nuclear power plants in the world become members of an association for mutual insurance of nuclear risks

Perspectives of development and implementation of closed fuel cycle

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Strategic goals of Russian nuclear power engineering and nuclear fuel cycle development are: the improvement of fuel balance of the country, increase of the portion of high-technology and knowledge-intensive products in internal gross income, efficient solution of greenhouse gases effluents. Achievement of such goals supposes active growth of nuclear generation in the sphere of nuclear power engineering and, afterwards, in the sphere of energy-intensive industrial technologies. Nuclear power engineering will be recognized by the public if the guarantees are provided that requirement to safety, physical security and economic competitiveness are met.

When full scale development of current nuclear industry its system problems are revealed. Even at the nearest stage it is a continuously growth of the amount of spent nuclear fuel and in the long-term perspective it will be the restriction of fuel recourse due to low effectiveness of natural uranium use. Systematic solution of these problems will consist in formation of two-component nuclear power system with VVER and fast breeder reactors operating in closed nuclear fuel cycle. Apart from the power augmentation of VVERs, in accordance with the investment program in force the new key elements must be developed and implemented within the nearest decades for formation of such a nuclear power system. There are head power unit and minor series of NPPs with BN-1200, which will be oriented for operation as part of the industrial power complex in cooperation with the enterprises of fuel cycle closing. In this system reactor BN-1200 will ensure the extended reproduction of fuel and will use regenerate from spent nuclear fuel treatment of VVER and BN reactor, centralized industrial complex of nuclear fuel cycle closing including storage and treatment of VVER spent nuclear fuel, treatment of fast breeder reactor spent nuclear fuel, fuel recirculation in fast breeder reactors and VVERs, conditioning and isolation of radwastes. The function of experimental – commercial trial run of fuel cycle closing technology shall be performed by the power unit of BN-800 in combination with the relevant components of closed fuel cycle infrastructure.

Sodium technology for fast breeder reactors and MOX fuel for fast breeders and partial loading of VVER are considered as top-priority technologies since they can ensure achievement of necessary economic and technical characteristics, they are relevant and have operation experience. Close to the middle of the 20-th years the decision of replacement of VVERs for BN-1200 in investment programs will be possible if their construction costs and in-service expenses are not greater than 15% compared to the VVER technology.

Operation of Nuclear Power Units at an Increased Level of Rated Power. Prospects of Further Power Uprate up to 107-110%

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One of the possibilities of electricity generation increase at VVER-1000 power units is the increase of reactor plant (RP) heat power on account of equipment engineering margins with regard for actual characteristics obtained as a result of its manufacturing and recorded in the process of operation.

Based on the Decision dated 27.11.2003 of STC of Roseneroatom and section №4 of STC of Minatom, Russia, the operating organization along with the architect engineer, the general designer and the scientific leader of the NPP design organized the work on RP heat power increasing at the operating power units.

Starting from the year 2004 Rosenergoatom Concern OJSC carries out activities on increasing the operating NPPs power beyond the rated power level (the power uprate).

The State Corporation “Rosatom” approved the “Program on increase of electricity generation at operating power units of Rosenergoatom Concern OJSC for 2011-2015”, wherein the steps for implementation of the tasks for safety enhancement and increase of electricity generation at operating power units were identified.

One of the main steps towards an increased electricity generation is the increase of heat power at VVER-1000 up 104% Nrated.

Nine power units out of ten VVER-1000 have already been transferred to a higher power level of 104% Nrated. Out of them 7 power units are in the pilot operation, while 2 power units are in commercial operation. Tests at 104%Nrated are scheduled to be carry out at the remaining Kalinin NPP №4 for the year 2014.

The report reflects the main results of VVER-1000 power uprate up to 104%Nrated, shows the scope of justifications and upgradings of the basic process equipment and mentions the use of new modernized FAs.

The report considers also the aspect of further power increase up to 107-110%Nrated. The "Analysis of technical capacity and economic expedience of power increase at unit №4 of Balakovo NPP" was performed for that goal. Results of the analysis give evidence to the feasibility of unit operation at 107–110% of rated power in principle. Based on the analysis performed, in accordance with the program of enhancement of effectiveness of Rosenergoatom Concern OJSC activities, the "Plan of organization and technical measures for increase of electricity generation by means of heat power increase at the pilot Balakovo Unit 4 up to 107-110% Nrated and 18-month reactor cycle" was developed. Pursuant to the plan the R&D studies were performed in order to justify and determine the scope of upgrades and tests for V-320 RP model, as well as the design of TVS-2M FA with mixing grids was identified aimed at increase of DNB margin.

The below are the next implementation stages of the plan:

- Conduct of SG separation tests at N=80%Nrated with three operating loops, and determination of the uprated level.
- Justification of TVS-2M FA with mixing grids for operation at N=110%Nrated.
- Justification of TS, TG and power unit auxiliary equipment upgrade.
- Obtaining of Rostechnadzor approval for carrying out the activities related to power uprate and upgrading.
- Implementation of RP, TS, TG and equipment upgrading measures.
- Conduct of tests at power level of 107-110%Nrated.
- Pilot operation of Balakovo Unit 4 with subsequent transition to commercial operation.

Based on the results of pilot operation at power level of 107-110%Nrated a decision will be made regarding further replication of such experience at other VVER-1000 units.

Medium-Size VVER reactor plants

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Design of a power unit with medium-size reactor plant VVER-600 is developed on the basis of AES-2006 and VVER-TOI nuclear plant designs.

Circuitry and design solutions for reactor plants AES-2006 and VVER-TOI that are based on the traditional VVER technology confirmed by long lasting operation (more than 1830 reactor-years) of operating VVERs were used in VVER-600.

The design is developed as a priority measure for substitution of the capacities of the operating NPPs. Evidently, in order to save and expand the markets covered by the Russian designs and enterprises abroad it is necessary to ensure construction of the head power unit VVER-600 in Russia, since foreign customers who have limitations on the grid capacity showed the interest to the design of NPP of middle power.

Reactor plant (RP) is of two-loop type, thermal power 1600 MW.

Transfer from six and four-loop to a two-loop VVER would allow to:

- significantly reduce the dimensions of reactor building due to reduction of containment:

VVER-600	AES-2006	AP-600
36 m	44 m	39,6 m

- reduce the number of RP equipment pieces (SGs, RCPS, support structures, embedded parts) and, as a result, reduce the number of supporting systems;
- reduce total metal consumption of RP;
- reduce the time for construction and installation of equipment;
- reduce the physical volumes due to application of new technologies and pre-assembled modular components;
- reduce the running expenses for inspection, maintenance and repair of equipment;
- significantly reduce the radiation exposure of the attending personnel.

The VVER-600 design is based on:

- maximum borrow of the main equipment from VVER-1200 design;
- readiness of the Russian industry for equipment manufacturing;
- safety assurance at the level of Generation 3+;
- 60-year service lifetime of RP.

In 2013, at the joint meeting of STC №1 of State Corporation "Rosatom" and STC of Concern Rosenergoatom OJSC on the topic of "Comparative analysis of economic and technical characteristics of two-unit nuclear power plants and nuclear cogeneration plants of medium capacity" it was noted that:

"One of the main tasks of contemporary nuclear power engineering development is the development of an efficient power unit of medium capacity.

Those regions where use of traditional organic power sources is impossible or hindered and the power transmission lines are either absent or imposing restrictions on the capacity of generating plants are the areas of application of NPPs with such power units".

VVER Reactor with Spectral Control – the Way to Efficient Utilization of U-238

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In spite of great complexity of organization of nuclear fuel cycle compared to the fuel cycle of hydrocarbonic power engineering there are weighty arguments in favor of nuclear power engineering development. Comparative estimations of dynamic of change of the main constituents of power sources show that the maximum portion, which can be occupied by the nuclear industry by the end of XXI century, can make up about 30% in energy consumption. Although such evolution of nuclear industry will demand considerable investments, efficiency of such investments will be continuously growing up in the long-term perspective. A full potential of fission nuclear power engineering can only be implemented within a system capable to ensure efficient utilization of energy resource of uranium-238 and thorium-232.

In the nearest future it is necessary to solve the issue of the rates of nuclear fuel reproduction (more specifically, the efficient use of energy resources of uranium-238 and thorium-232) in thermal and fast breeder reactors: how many breeders shall be included in the total fleet of nuclear power sources, what shall be their breeding parameters at various stages of evolution. Service life of new nuclear power units makes up 60 years, therefore in this century nuclear energy systems will consist, in certain portions, of thermal and fast breeder reactors.

The significant and, to a certain degree, the crucial factors for adopting a weighty technical policy for several decades in Russia are the time and scope of further existence and prevailing of thermal-neutron water vessel-type reactors with a flexible (open, partially closed, closed) fuel cycle (U, Pu, Th) in the structure of nuclear industry. It is necessary because the nuclear facilities have a long service life (60 years and more). Justification of selected directions of water vessel-type reactors development including provision of fuel resources for them shall be estimated for the entire lifetime of facilities. It is necessary to continue searching for the ways of upgrading the new and currently constructed VVERs for competitive operation within the structure of future nuclear industry under the conditions of gradually exhausted cheap natural uranium resources and transferring to a closed nuclear fuel cycle with the use of pellet mixed MOX- and MIX- fuel, and afterwards thorium-uranium fuel for operation in closed fuel cycle. Utilization of products of SNF reprocessing in closed fuel cycle would result in saving of natural uranium, accessible resources of which are limited.

Further development of thermal reactors is related to creation of a range of power units, improvement of fuel utilization within the open nuclear fuel cycle (with consumption of natural uranium not greater than 130 t per GW(e)*year), increase of fuel breeding factor within the closed nuclear fuel cycle (up to BF~0.8), reduction of time for construction and commissioning of new power units, increase of efficiency of power conversion and investment attractiveness, transfer to supercritical coolant parameters, the use of advanced composite structural materials, development of new fuel elements with extended surface of heat removal (thin fuel elements, ring-type fuel elements, micro fuel elements etc.), and extension of the area of use.

Thus, the following task is set: apart from upgrading of light water reactors operating in the open fuel cycle it is necessary to study how to upgrade them for operation in closed fuel cycle.

One of the steps towards this goal is a VVER with spectral control of change of reactor core reactivity during fuel burn-up (VVER-S). Various methods of spectral control are known, for example, by means of change of the water-fuel ratio, change of light / heavy water ratio in the core, change of steam content in water etc.

A fundamental feature of VVER-S consists in spectral regulation of the core reactivity in the process of fuel burn-up achieved by means of water-uranium ratio alteration using mobile water displacers and/ or heavy water and full refusal of boron liquid control. In VVER-S, excessive neutrons are absorbed by uranium-238, instead of absorption in boric acid. Thus, the excessive fuel necessary for the whole fuel cycle duration is compensated, and at the same time there is generated plutonium, i.e. new fissile fuel. Of course, it is complicated to ensure an extended reproduction of fuel in a thermal reactor in a manner similar to that in fast breeder reactors but it is still possible to considerably reduce consumption of natural uranium in the open fuel cycle, and to increase twice the fuel breeding factor in the closed nuclear fuel cycle. VVER-S core can be fully loaded with MOX-fuel. No similar thermal reactors have existed before.

Therefore, VVER-S can operate some 30 years on uranium fuel and then some 30 years more on its own plutonium fuel. As a result it would save ~50% of natural uranium. The annual profit as of today does not represent a considerable saving but it becomes a significant one from the viewpoint of the entire lifetime. Considering the expensive resources of natural uranium that cost ~ (130-260) USD per 1 kg, the saving over the entire lifetime would be equivalent to the capital cost of the power unit.

The goals of VVER-S design can be stated as follows:

- minimization of natural uranium consumption when operating in the open fuel cycle;

- operation in the closed fuel cycle with full loading of the core with MOX-fuel and maximum possible utilization of uranium-238 raw material potential, as well as solution of the problem of spent nuclear fuel accumulation;
- attractiveness for customers and competitiveness of the nuclear plant at the world market.

Nowadays, NRC "Kurchatov Institute" in cooperation with EDO Gidropress and Atomenergoproject OJSC have completed development of technical requirements to VVER-S.

Development of nuclear energy in China

Chen Zhaobo
State nuclear power technology corporation LTD, China

The presentation overviews the development of nuclear power in China, including the development policies, present situation and the development plan of nuclear power. It focuses on the construction situation of 4 Ap1000 units, and the progress of self-reliant development of CAP1400. The presentation also introduces the achievements of Sino-Russian nuclear energy cooperation, the excellent performance of the two advanced VVER units in Tianwan Phase I and the progress for the construction of the same type 2 reactors in Tianwan Phase II.

WANO's post Fukushima actions

Jacques Regaldo
Chairman of WANO

WANO's mission is to maximize the safety and reliability of nuclear power plants worldwide by working together to assess, to benchmark and improve performance through mutual support, exchange of information and emulation of best practices.

Fukushima accident strongly impacted the nuclear community and it also brought WANO to question its positioning and scope of activities. Five strategic orientations have hence been decided to strengthen WANO's role, aiming to bring a more consistent, transparent and integrated response to the nuclear operators. Those strategic orientations (expand the scope of WANO activities, develop a worldwide integrated event response strategy, enhance WANO's credibility, improve visibility outside and transparency within the community of nuclear operators, and improve internal consistency) have led to the implementation of twelve operational projects, which results shall be presented during the next WANO Biennale General Meeting in Toronto in 2015.

Thus, WANO peer review process, which constitutes its core-business, has been intensified including corporate and pre start up peer reviews and, for Japanese plants, restart reviews. WANO also chose to expand its scope of activity to include elements of design, based on the principle that the role of a nuclear operator is not only to operate safely, but also to be sure that the plant he is operating is safe.

WANO has also unanimously decided to progressively implement a Nuclear Safety Performance Assessment, based on a common methodology for all its regions, which is probably one the most structuring programme for the Association and its members for the coming years.

WANO's oversight committee judged recently that the schedule for all the projects is on track for 2015, four of them remaining with challenging issues.

At the same time, WANO aims to cooperate strongly at both regional and international levels with all international safety organizations, being convinced that trust can be recovered with a strong safety commitment and credibility of both regulators and operators.

We consider that we rely on each other to improve safety!

Solution of scientific and technical problems of the final stage of RBMK operation

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RDIPE OJSC, Moscow

1 Brief history of RBML operation.

1.1 Shrinkage of openings in graphite units and radial swelling.

Replacement of channels (single-type and stage-by-stage).

1.2 Axial and radial shrink of the fuel cask second set, exhaustion of fuel cask bellows travel, growth of fuel cask inner diameter.

1.3 Vertical shrink of graphite columns (problem of route telescope joint).

1.4 Secondary swelling of graphite units, their deformation and cracking, distortion of setting and channels.

2 Repair-recovery activities for decrease of distortion and rate of setting distortion growth (technology, effectiveness, ways of optimization, main outcomes and tasks).

3 Justification of safety and criteria of operability of reactor internal units.

3.1 Strength analysis and dynamic processes.

3.2 Thermal dependability of FAs.

3.3 Operability (mechanical, overloading) of FAs and CPS rods.

3.4 Analysis of neutronics characteristics and thermal mode of setting.

3.5 Analysis of design based and beyond design based accidents.

3.6 Additional R&Ds and verification of developed design codes.

4 Activities on upgrading of FA and CPS rods (CCrods, EP).

5 Tasks resulting from the analysis of beyond design based accidents at RBMK power units.

The IAEA Fukushima report and implementation of the IAEA Nuclear Safety Action Plan

Caruso G.
Special Coordinator for the IAEA Nuclear Safety Action Plan, IAEA

Following the accident at TEPCO's Fukushima Daiichi nuclear power plant (NPP) in March 2011, the Director General of the International Atomic Energy Agency (IAEA) convened the IAEA Ministerial Conference on Nuclear Safety in June 2011. The Conference adopted a Ministerial Declaration which, inter alia, requested the Director General to prepare a draft "action plan on nuclear safety". The Action Plan covers the relevant aspects relating to nuclear safety, emergency preparedness and response, and radiation protection of people and the environment, as well as the relevant international legal framework. The Action Plan was adopted by the IAEA's Board of Governors and unanimously endorsed by the IAEA General Conference in 2011 with the ultimate aim of strengthening nuclear safety worldwide. This presentation will provide an overview on the progress in implementation of the Action Plan with a focus on the key areas: assess the safety vulnerabilities of nuclear power plants in the light of lessons learned from the Fukushima accident; strengthen IAEA peer reviews in order to maximize the benefits to Member States; strengthen emergency preparedness and response; strengthen the effectiveness of national regulatory bodies; strengthen the effectiveness of operating organizations with respect to nuclear safety; review and strengthen IAEA Safety Standards and improve their implementation; improve the effectiveness of the international legal framework; facilitate the development of the infrastructure necessary for Member States embarking on a nuclear power programme; strengthen and maintain capacity building - including education, training and exercises at the national, regional and international levels; ensure the on-going protection of people and the environment from ionizing radiation following a nuclear emergency; enhance transparency and effectiveness of communication and improve dissemination of information; and effectively utilize research and development.

As part of the Action Plan, the IAEA Secretariat is currently preparing the IAEA Fukushima Report that will be finalized by the end of 2014 and presented to the international community in 2015. The Report will be authoritative, factual and balanced with sufficient technical depth. It will take into account accumulated knowledge from relevant sources up to the time of its completion. The Report will build on the existing documentation available on the topic. However, it will not constitute a review or a summary of the existing publications but rather add value filling out the information gaps and providing lessons learned. The inputs to the Report are being prepared by approximately 180 international experts from over 40 Member States and international organizations who selected by the IAEA on the basis of the knowledge and expertise and taking into account the geographical distribution. The experts are divided into five Working Groups and each WG corresponds to a coherent technical/scientific area covered by the Report:

1. Description and context of the accident
2. Safety assessment
3. Emergency preparedness and response
4. Radiological consequences
5. Post-accident recovery

Rosenergoatom NPP cost control in the environment of tariff rise restrictions

Arkhangelskaya A.I.

Rosenergoatom Concern OJSC

1. The need in an increased internal efficiency of Rosenergoatom performance is stipulated by a requirement imposed by the State as a co-investor of implementation of the Programme of the State Atomic Energy Corporation "Rosatom" activities for long-term period (2009-2015), approved by the RF Government Ordinance № 705 dated 20.09.2008, as well as other decisions of the Government of the Russian Federation.

2. Besides the external challenges, the State Corporation "Rosatom", through the conditions put into scenarios for development strategy forming, middle-term planning and budgeting, sets even more ambitious plans to the industry enterprises, which become an internal challenge being also focused on optimization of operational activities and feeding the growth of Rosenergoatom investment resource.

3. In order to ensure fulfillment of these ambitious plans, State Corporation "Rosatom" has developed and introduced a stimulation system that allows achieving the specified goals by means of key performance targets setting at a challenging level in KPI maps of managerial staff.

The primary area for optimization lies at NPP level, hence, in order to achieve the goals it is not enough to promote an initiative in "top-down" direction, but there is a need in an initiative going in "upward" direction, i.e. from nuclear plants.

4. Apart from pressure on Rosenergoatom operational costs, an important leverage for investment resource increase is the option of loan funds employment, which is limited by the ratio (debt + commitments)/EBITDA $\leqslant 2.7$.

5. Generally, all the mentioned measures work for the growth of internal efficiency of Rosenergoatom and for the increase of the investment resource

level, but nevertheless it is necessary to implement in parallel an investment efficiency enhancement programme that would allow ensuring construction of the power units optimized from both viewpoints of the cost and future operational efficiency, with unconditional observation of safety requirements.

Issues concerning export of russian nuclear power plants to european markets

Jukka Laaksonen
Rosatom Overseas CJSC

Rosatom has a number of strengths to get a significant share of new nuclear built markets in the entire Europe, except France that has its own nuclear industry. The most important thing is that the new VVER plants offered by Rosatom are safe and economic. They are based on long evolution in small increments, and the operating experience has been good as concerns reliability of production and safety. No serious abnormal events have taken place in the history of VVER-1000 plants that extends to nearly 1500 operating years. The new VVER plants have several technical innovations not found in other PWR designs and they are unique in that sense that safety in the case of abnormal events could be assured by a full set of both active and passive safety systems. An important factor behind the safety of new VVER plants is that the Russian nuclear safety regulations known as OPB 88/12 are in compliance with the latest IAEA Safety Standard for NPP design, SSR 2/1 and also with the safety objectives jointly agreed by the European nuclear safety regulators in WENRA. Besides, offering safe and reliable NPPs, Rosatom can tailor the financing arrangements of the exported NPP is several ways, depending on the wishes of each customer. The openly communicated policy of subcontracting the manufacturing of components and construction works to local companies in each country is attractive. Another unique feature of Rosatom is that it has a large home market for new NPP program. The operator of all Russian NPPs is Rosenergoatom (REA) that collects and analyzes systematically the operating experiences of its fleet, and as a common policy makes improvements in safety and reliability of its plants. In this respect it has a similar attitude towards continuous improvement as the utilities in the Western Europe. As concerns introduction of new design features to the VVER plants, REA is an intelligent and critical customer to the Russian designers, and has stringent requirements for verification of positive impact of changes on design. New features are also studied carefully by the Russian research organizations that have opportunities to study new features by large scale testing. In addition to then modern new NPPs, Rosatom can offer nuclear fuel that has proven its high quality in operation and resulted in satisfaction among the customers.

Besides several positive aspects mentioned above, there are also challenges that need to be addressed to make Rosatom export projects smoother and economically more attractive both to Rosatom and its customers. Rosatom has subsidiaries and partners who know well the European markets but those could be used more effectively for marketing and for building credibility in customer countries. Also smooth co-operation between Rosatom entities during project preparation and implementation needs emphasize. Part of the internal co-operation could be enhanced co-operation in licensing and regulation that costs nearly 100 MEuros in regulatory fees and probably even more in the work costs by the suppliers during the entire construction period if it is not optimized from cost point of view. Optimization should involve standardization of main safety document packages so that the same documents can be used with minor changes in all countries. Regulatory practices during construction vary from country to country and good awareness of these practices is essential to avoid spending resources unnecessarily. In addition to awareness of regulatory approach, Rosatom should study environmental conditions and specific hazards on NPP sites. Attention should also be paid to development of Integrated Management System in the main Rosatom organizations because that is a requirement and will be strictly audited by the customers and the national regulators. The policy and practice concerning technical standards to be applied for design, manufacturing, qualifying and testing of SSCs has to be clear and well documented so that it can be argued in front of the customers and regulators. The general policy should be to use Russian standards for mechanical equipment materials and manufacturing, international IEC standards for electrical and I&C equipment and national standards of the customer country or Eurocodes for civil structures. Protection of technical information has traditionally been strict in Russia but there is no good reason to continue such policy. All safety documentation needed for export projects should be evaluated and general decision should be made on declassification of information that is not commercially sensitive or security relevant.

Last but not least I would emphasize the importance of building confidence in all customer countries on the safety and quality of the Russian technology and on Rosatom capability to construct NPPs in budget and schedule. This requires persons who know the communication culture of the customer country and can provide trustworthy information in good English language.

Expected forecast of deflections of RBMK-1000 channels at the stage of graphite cracking

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The issues of software development and application are considered to perform design estimations of kinetic of deflections of RBMK-1000 columns structure aimed at forecasting the guaranteed operation time of power units till the deflections of the columns reach the critical value and, consequently, their safe operation. The main mechanism of deflections accumulation is provided by the fact that the graphite, when increase of the accumulated neutron fluence, transfers from the shrinkage stage to the swelling, which, due to irregular distribution of temperature and fluence on the structure area on the whole and local irregularity of the graphite units volume will cause longitudinal cracks therein. When further irradiation the cracked graphite units are «opened», which will result in significant growth of their overall dimensions, this is the reason of additional deflection of neighboring graphite columns and such deflection is accumulated while moving from the center of the core to the periphery areas.

The description of the detailed three-dimensional model of RBMK-1000 graphite structure containing 3 levels as well as two versions of engineering models– deterministic and stochastic – are given. The first level of the detailed model of structure is a volumetric finite-element model of graphite unit, which enables to perform the calculations of stress and strain state (SSS) taking into account actual geometry and sizes of graphite units, impact of fields of neutron fluence and temperature irregularly distributed on the graphite units, anisotropy of mechanical-and-physical properties of graphite, change of mechanical-and-physical properties of graphite induced by the operation factors: temperature, irradiation, radiation creep and cracking.

Second level – model of graphite column consisting of graphite units of various height, which enables to simulate a contact interaction of end surfaces of graphite units and graphite units and pipes of the process channels occurring when significant deflections of columns as well as conditions of process channels secure.

Third level – model of RBMK core structure containing of graphite columns, which enables to simulate a contact interaction between graphite columns when significant deflections of columns that exceed initial design clearances between them.

The issues of enhancement of effectiveness of finite-element and super-element algorithms implemented in software complex UZOR 1.0 for calculations using the developed models the number of which degree of freedom exceeds 10^8 are considered.

The results of estimations of maximum deflections of graphite columns of power unit 1 of Leningrad NPP and power unit 2 of Kursk NPP for the time of shutdown for repair and after 1, 2 and 3 years after repair are given.

Safety and effectiveness of spent fuel handling at FSUE MChC

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FSUE "Mountain and chemical complex"

Concept of nuclear fuel-power complex development in Russia addresses the closing of nuclear fuel cycle and is implemented in five main directions:

1. Construction of power units with VVER of new generation.
2. Centralized storage of spent nuclear fuel till its processing.
3. Establishment of the fast breeder reactors stock which will enable development of power industry on fast breeders.
4. Establishment of a large plant for spent nuclear fuel processing and fabrication of MOX fuel.
5. Establishment of the place for final isolation of solidified HR wastes.

Two of the above directions are implemented nowadays at FSUE «MChC» and address spent nuclear fuel handling: complex of centralized storage of spent nuclear fuel of reactor facilities VVER-1000 and RBMK-1000 and newly established experimental demonstration center for SNF processing which capacity will amount to 250T/year for SNF.

Activities related to setup of production for MOX fuel fabrication intended for fuel supply to power unit №4 of Beloyarsk NPP with fast breeder reactor 800 are also carried out at FSUE «MChC».

Complex for long-term centralized storage of SNF consists of three storages:

- «wet» storage of SNF for VVER-1000;
- «dry» storage of chamber type for SNF of VVER-1000;
- «dry» storage of chamber type for SNF of RBMK-1000.

On «wet» storage of FSUE «MChC» the reconstruction activities were carried out. The main goal of storage reconstruction was qualitative enhancement of safety level: seismic resistance was increased; cranes were replaced; capacity and reliability of cooling system were improved.

One of the top-priority tasks is augmentation of the capacity of SNF storage for VVERs-1000 to ensure the regular removal of spent nuclear fuel from the NPPs. To solve this task the concept of compact storage of spent FAs is currently implemented. Within the frames of SNF storage-1 reconstruction the additional compartments for SNF storage were also put into service, which enabled to considerably increase the capacity of storage for SNF of VVER-1000 compared to design capacity.

Deterministic and probabilistic safety analyses of «wet» and «dry» SNF storages were performed. The results obtained prove that safety level of «wet» spent nuclear fuel storage (SFS-1) and «dry» spent nuclear fuel storage (SFS-2) at FSUE "MChC" are in compliance with the normative criteria of "General provisions of safety assurance of nuclear fuel cycle facilities"- probability of beyond design basis accidents at the nuclear fuel cycle facility shall not exceed $1 \cdot 10^{-6}$ per year.

In 2012 the start up complex of «dry» SNF storage was put into operation. Design of «dry» storage underwent the international expertise in the company SGN (France). In April 2012 the first train with SNF from Leningrad NPP was accepted. Nowadays the «dry» storage works in a routine order.

Unique equipment of Russian engineering design is used in the technological process that have no analogues in the world. All process operations related to SNF movement are performed in automatic mode, which enables to considerably reduce radiation load on the personnel. Heat removal is implemented by a passive method – by convection. Spent FAs are stored in capsules placed in the bottles and storage socket, thus, ensuring four safety barriers taking into account the fuel element cladding.

Maximum seismic resistance analysis was made for civil structures and equipment of «wet» and «dry» storage of SNF. Maximum seismic impact for the site of storage places is 7 of scale MSK-64.

Based on the analysis it was determined that:

- civil structures of «wet» storage retain the integrity at 8,0score in scale MSK-64.
- civil structures of «dry» storage retain the integrity at 9,6 score in scale MSK-64.

Experimental and demonstration center for spent nuclear fuel processing set up at FSUE «MCHC» is intended for test of innovation technologies for SNF processing of VVER-1000 and methods of resulting RWs handling, which meet the requirements of cost effectiveness and obtaining of initial data for design, construction and putting into operation of large scale plant for SNF processing.

Updated PUREX-process is the fundamental for the technology of SNF processing. Use of innovation technologies that have no analogues in the world enables to introduce the pilot demonstration centers as a radiochemical plant of generation 3+ (UP-2, 3 – generation 2+).

Innovation processes developed and tested in the lab by the leading branch institutes of the country lie in the bottom of the process flowchart of pilot demonstration centers.

Start up complex of pilot demonstration centers is scheduled for putting into operation for 2015 (up to 10 ton/year). Putting into operation of pilot demonstration centers in the scope of full development – 2018 (250 ton/year).

Four basic postulates can be determined for comprehensive safety enhancement when SNF handling:

1. SNF removal from NPP site and its disposal on the facilities of centralized storage.
2. Use of passive systems of heat removal («dry» storages).
3. Use of multi-barrier systems of SNF storage isolation in airtight bottles and units of storage.
4. Set up of the systems for beyond design basis accidents management and localization of their consequences.

For further enhancement of safety when SNF handling it is advisable to process SNF and closing of nuclear fuel cycle.

Scientific- methodological basics of experimental estimations of residual resource using technical diagnostics means

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A new facing the problems related to experimental estimations of residual resource of safe operation occurs due to exhaustion of the designed lifetime of NPPs with VVER, RBMK and fast breeders. At this, the below is included into the main tasks to be solved:

- categorization of NPP and equipment according to risk and safety criteria;
- assessment of design-technological and in-service factors that impact the current state and risks of the elements of such equipment and materials;
- selection of types of the limit states and criteria for assessment of duration, vitality and safety;
- justification of diagnostic methods in order to identify the technical condition of equipment and obtaining the appropriate initial design characteristics;
- selection of calculation and experimental methods for identification and justification of safe operation residual recourse;
- selection of design equations and their parameters for quantitative determination of resources and risks of the equipment being analyzed and NPP on the whole.

Technical requirements for equipment as well as relative regulatory and technical documentation related to design, manufacturing and operation of respective equipment are taken into account while assessment of residual resource of safe operation. Assessment of residual resource is made for critical elements of equipment, which, being operated, bear mechanical, hydrodynamic and thermal loads in wide ranges of cycles, levels of stress and deformations, sizes of defects and impact of environment.

Assessment of equipment technical condition is performed by methods and means of destructive and non-destructive tests according to norms in force (diagnostics, flaw detection) stating the following main parameters used for assessment of residual resource: characteristics of loading (stress, deformation, temperature); characteristics of defects, cracks in the first turn (their sizes, location areas and direction); characteristics of properties of structural materials (base metal, metal of welded joints and build-ups).

Below refer to such parameters: stress σ (deformation e), temperature t, sizes, forms and location places of defects (cracks) l, modifying in time τ and depending on the conditions of in-service loading (pressure p, mechanical, thermal and electromagnetic efforts, rate, acceleration), geometrical forms and dimensions of structural elements and mechanical properties of materials. Maximum (σ_{max}, e_{max}, t_{max}) and amplitude values (σ_a, e_a, t_a) of basic in-service parameters are subject to compulsory determination as well.

Accumulated cyclic, temporal, corrosion and other damages as well as main design, process and in-service factors that change characteristics of limit states are also taken into account when experimental estimations of residual resource of safe operation.

Based on the data obtained and introducing the relevant resources (related to stress, deformation, durability, critical temperatures and sizes of cracks) the service life is determined till the designed residual resource is exhausted or till the next examination and assessment of the condition of NPP equipment being assessed.

Section 1

SAFE AND EFFECTIVE OPERATION OF NPPS

Subsection 1.1
OPERATION, MAINTENANCE AND REPAIR OF NPPS WITH VVER, RBMK, BN AND EGP-6 REACTORS

Topical area
OPERATION OF NPPS WITH VVER REACTORS

Main results of operation and perspectives of new generation fuel use

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Kola NPP

Pilot operation of fuel of new generation continues at Kola NPP unit 4, which has a number of design specific features ensuring excellent operation performance. Economic effectiveness increase is ensured due to decrease of parasite capture of thermal neutrons in structural materials (zirconium weight is reduced), updating of water-uranium ratio (increase of fuel element stroke) and increase of uranium load (pellets of larger diameter without an orifice).

New results of operation are given and brief analysis of data obtained is presented in the report. Goals of pilot operation and specific features of implementation of fuel enriched to 4.87% are described. Experimental characteristics of VVER-440 reactor core with fuel of generations 2 and 3 with enrichment up to 4.87% were obtained. Some issues of PK-2 and PK-3 operation were discussed: evaluation of impact of fuel assembly jacket absence on records of thermal couples when combined operation of assemblies of generations 2 and 3, specific features and methods of fuel load formation and perspectives of new types of fuel use.

Analysis of emergencies in VVERs related to reactivity disturbance

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The report will present a brief description of technique for analysis of emergencies related to reactivity disturbance at VVERs in the form as this technique is currently seen and used in SRC KI.

Examples of analysis of emergencies will be given to justify the new fuel and justify algorithms of operator's actions.

Design approaches regarding reactor core for operation safety assurance of PRW-1000

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JSC OKB «GIDROPRESS»

New scenarios of B-320 power units operation implemented for nuclear plant capacity enhancement, as well as scenarios planned for operating units and units under construction require that additional decisions are made for safe and reliable operation.

To ensure operation of power units in 18-month cycle at power 104% Nrated it required to modify and re-justify the limit curve by the linear power of the fuel elements. Implementation of profile fuel elements became an additional measure for attenuation of in-service restrictions. The decision of wide implementation of such fuel elements can be made in 2017 based on pilot operation results.

To ensure tests of power unit 4 of BalNPP at 107-110% power the design of FA-2M with mixed grids was developed. To the time of tests conduct the reactor core will be fully filled with fuel assemblies of such type. The first stage – justification of such assemblies at 104% power was completed. Tests conducted on grids show that thermal reliability of reactor core at 107-110% power will be ensured. It is expected to implement a big program of experiments for justification of RP safety in a wide range of parameters and modes according to verified design programs for this range.

Trial operation of alloys Э-110M, Э-635M, Э-125 continues at power unit 2 of BalNPP. Results of this operation will enable a direct final evaluation of expediency of wide implementation of such alloys and assess their relative qualities as well.

Occurrence of separate failed FAs makes the issue of ADF expansion more urgent. Available results of trial operation of FA-2M with ADF enable do this in a wide range.

Obtained results of NMC operation also represent the base proving their wide implementation both on the operating units and on power units under construction. Use of NMCs enables obtaining more trustworthy idea of the state of thermophysical parameters in the core.

Specific tasks were also identified for absorbing rods updating. The first step of this work must consist in unification of their design based on the planned calculations. It would be advisable to increase the lifetime of absorbing element to reduce the number of solid radwastes to be buried.

The report points out the results of FA-2M operation at 6 power units of BalNPP and RostNPP. It is demonstrated that in the new more long lasting cycles the operation results are also successful, including the results of handling operations performed during reactor core reloading.

In spite of successful results of operation and based on operation experience of fuel assemblies abroad the task of FA repair ensuring at the NPP remains urgent. To solve this task the significant reserves have been prepared, which can be used under the request of the Customer.

Level of FA-2M reliability achieved at 6 operating PWR units enables use of this design at power units VVER-1000 in Russia and abroad. They are already used at TNPP in China and planned for use at Bushehr NPP.

Water chemistry of secondary circuit at power units without copper containing alloys

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Nowadays power units with PWR of new generation are being constructed under the designs AES-2006 and VVER-TOI. The main feature of the secondary circuit of these reactors is absence of equipment made of copper containing alloys in the condensate and feed line of the equipment, as well as reduced value of ultimate permissible suction of cooling water to $1*10^{-5}$ from the steam flowrate to condenser. As compared to the designs, the regulated level of salt admixtures in steam generator blowdown water is 3–7 times lower compared to the norms for operating VVER-1000 and can be comparable to the norms of foreign PWRs.

To reduce the corrosion damage of pipelines and heat exchanging equipment and growth of sediments on the heat exchanging surfaces of SG the designs of AES-2006 and VVER-TOI provide for secondary circuit water chemistry with correctional treatment by ethanolamine, ammonia and hydrazine -hydrate.

Based on operation experience of ethanolamine WCh maintaining at operating VVER-1000 the iron concentration in SG feedwater makes up about 2-2,5 mkg/dm^3. Compared to hydrazine- ammonia WCh the rate of drift of SG heat exchanging surface reduced by 4-5 times and, consequently, inter-washing interval of SG extended to once every 8-10 years.

However, as operation experience of PWRs abroad shows, use of WCh with mixed amines enables reach the concentration of iron in the feedwater less than 1 mkg/dm^3, at the same time the purification facilities for turbine condensate and SG blowdown water are not overloaded.

This report describes the advantages of maintaining the secondary circuit water chemistry using mixture of amines (ethanolamine and dimethylamine) compared to ethanolamine dosing WCh adopted, and also the possibility of water chemistry perfection both at VVERs of new generation and at operating VVERs-1000.

Use of mixed amines for treatment of secondary circuit operation media will enable to:

- reduce delivery of corrosion products with feedwater to SG to less than 1,0 mkg/dm^3;
- extend lifetime of ionites;
- extend the service lifetime of VVER equipment;
- extend inter-washing interval of SG heat exchanging surfaces to once every 8–20 years.

Possible schemes of turbine condensate purification at various quality of cooling water of turbine condensates are presented in the report for new generation of designs (AES-2006 and VVER-TOI), optimal installation diagram of steam generator blowdown water purification facility (AWT-5) are given in the report and their advantages are demonstrated.

Complex approach to support systems of secondary circuit water chemistry will enable justify and set forth the necessary set of process systems in the design to ensure support of the selected water chemistry.

Development and implementation of severe accident management guides in the light of accident at Fukushima Daiichi

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Currently within the frames of updated measures for mitigation the consequences of beyond-design basis accidents the severe accident management guides (SAMG) are developed at the NPPs with VVER type reactor.

The progress of SAMG development for VVER has been considered. Brief description of the principles for SAMG development, composition and structure is given.

The outcomes on the accident at Fukushima Daiichi are presented related to development of emergency documentation; these are reflected in the documents of IAEA workshop on severe accident management held on 17 through 20 March, 2014. Taking into account the outcomes made at the workshop the below aspects of SAMG updating for VVERs are discussed hereby:

- expansion of areas of SAMG covering due to spent fuel pit;
- expansion of areas of SAMG covering to the state of the shutdown reactor;
- accounting of additional technical means available at VVER in the instructions of SAMG;
- probabilities of SAMG expansion in case of severe accidents occurred at several power units simultaneously and accidents resulting in considerable destructions on NPP site;

- NPP personnel training in SAMG observance and composition of personnel drills.

Nowadays the activities related to the first three aspects of SAMG updating specified above either started or were scheduled for performance.

As far as severe accidents occurred at several power units simultaneously and NPP personnel training in SAMG observance, the proposals have been introduced regarding further work taking into account lessons learned from Fukushima Daiichi.

Development of Full Scope Severe Accident Management Guidance for Tianwan VVER Plant

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The first ex-vessel core catcher has been applied at the operating VVER-1000 AES-91/99 reactors in China Tianwan nuclear power plant for severe accident management. After the Fukushima accident, the China National Nuclear Safety Administration (NNSA) department required the full scope severe accident management guidance (SAMG) and other Post-Fukushima improvement for NPPs in China. The full scope SAMG must include the spent fuel pool accident management and cover all the modes, including shutdown mode. The Tianwan VVER plant SAMG was developed by Nuclear Power Institute of China (NPIC). Besides the characteristic of full scope, the Tianwan SAMG involve following technologies: the simulation of core catcher with MELCOR code, interface analysis between the Emergency Operation Procedures and the SAMG involving the core catcher operation, Post-Fukushima improvements, spent fuel pool SAMG, spent fuel pool accident analysis with the MELCOR code simulation, validation for all guidelines by the full range simulators, the e-SAMG system to implement the guidelines on computers.

Topical area

OPERATION OF NPPS WITH CHANNEL AND FAST REACTORS

Calculated forecast of deflections of RBMK-1000 channels at the stage of graphite cracking

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The issues of software development and application are considered to perform design estimations of kinetic of deflections of RBMK-1000 columns structure aimed at forecasting the guaranteed operation time of power units till the deflections of the columns reach the critical value and, consequently, their safe operation. The main mechanism of deflections accumulation is provided by the fact that the graphite, when increase of the accumulated neutron fluence, transfers from the shrinkage stage to the swelling, which, due to irregular distribution of temperature and fluence on the structure area on the whole and local irregularity of the graphite units volume will cause longitudinal cracks therein. When further irradiation the cracked graphite units are «opened», which will result in significant growth of their overall dimensions, this is the reason of additional deflection of neighboring graphite columns and such deflection is accumulated while moving from the center of the core to the periphery areas.

The description of the detailed three-dimensional model of RBMK-1000 graphite structure containing 3 levels as well as two versions of engineering models– deterministic and stochastic – are given. The first level of the detailed model of structure is a volumetric finite-element model of graphite unit, which enables to perform the calculations of stress and strain state (SSS) taking into account actual geometry and sizes of graphite units, impact of fields of neutron fluence and temperature irregularly distributed on the graphite units, anisotropy of mechanical-and-physical properties of graphite, change of mechanical-and-physical properties of graphite induced by the operation factors: temperature, irradiation, radiation creep and cracking.

Second level – model of graphite column consisting of graphite units of various height, which enables to simulate a contact interaction of end surfaces of graphite units and graphite units and pipes of the process channels occurring when significant deflections of columns as well as conditions of process channels secure.

Third level – model of RBMK core structure containing of graphite columns, which enables to simulate a contact interaction between graphite columns when significant deflections of columns that exceed initial design clearances between them.

The issues of enhancement of effectiveness of finite-element and super-element algorithms implemented in software complex UZOR 1.0 for calculations using the developed models the number of which degree of freedom exceeds 10^8 are considered.

The results of estimations of maximum deflections of graphite columns of power unit 1 of Leningrad NPP and power unit 2 of Kursk NPP for the time of shutdown for repair and after 1, 2 and 3 years after repair are given.

Impact of dynamic effects on integrity of channel pipes and graphite structure stability of RBMK power units after repair

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Computer code FEMGR used for simulating the deformation of graphite columns of RBMK core under the conditions of dynamic loads, as well as results of the code verification are presented.

Mathematical model of the code takes into account the specific features of column design of fuel channels and control and protection system channels, impact of fixtures in the routes of pipe channels as well as post-repair state of graphite units.

Impact of the weight of structural elements of columns and channels on dynamic of deformation, pressure inside channel pipes and weight of bottom water utilities resulting in bending and axial compression of channels are taken into account. Impact of friction on deformation of columns, contact interaction of channel pipes with graphite elements of columns and graphite units of neighboring columns among each other are simulated.

Verification of the code was performed by the experimental data received from full-scale mechanical tests of RBMK column assemblies with channel pipes conducted on test bench in EREC and on the vibration bed of TSNII Mash.

Results of justification of refurbished structure stability of power unit 1 of LENNPP are given and justification results of channel pipes strength under seismic impact 7 score MSK-64 on structure as well under dynamic impact resulting from a single channel rupture are also presented.

Design analysis was performed with account of structure ageing and its changes related to refurbishment.

It is demonstrated that under the considered conditions of dynamic impacts the stability of graphite structure is ensured. Condition of FC and CPS pipes strength is met.

Analysis of graphite structure deflection by probabilistic model

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Operation resource of graphite structure of RBMK reactors is limited by the effects related to changed characteristics of graphite when irradiation by fast neutrons. As neutron fluence with energy more than 0,18 MeV grows up the graphite first compacts and then swells, which is induced by accumulation of defects and formation of micro cracks. Due to irregular distribution of fluence on the radius of graphite units the stresses therein grow up, which, ultimately, will result in formation of through longitudinal cracks and progressive increase of the units overall sizes in directions perpendicular to cracks. These processes are most intensively going on in the area of maximum high fluences, which results if deformation of graphite structure that attains a barrel-shaped form. Distortions of channels can hinder normal stroke of control rods and refueling.

Nowadays the most critical parameter for further operation of the reactor is maximum deflection of channels. To maintain this parameter within the permissible limits the activities were conducted at Leningrad and Kursk NPPs for restoration of RBMK long-term performance.

In order to assess the distortion of channels within further operation of deformed structure, as well as effectiveness of repair conducted the engineering methodology was developed based on probabilistic approach to determination of the moment of cracks formation and their arrangement on the facets of graphite units (*probabilistic model).

Brief description of probabilistic model implemented in the form of computer program and some calculation results are given in the report. Comparisons are made with the measurement results of channels deflection and consideration is given to contributing errors of forecast calculations and effectiveness of various schemes of structure repair.

Complex of softwares GEFEST 800 for conduct of operation calculations of neutronic characteristics of RP FB-800

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The developed software GEFEST 800 is intended for conduct of operation neutronic calculations of fast breeder reactor with sodium coolant FB-800, namely: efficient breeding ration; maximum reactivity margin; effectiveness of control and protection system single and group rods; full, specific and linear power of energy release in FA; coefficients of irregular energy release in FA and in reactor; injurious exposure of FA radiation; fuel burnout; reactivity coefficient (temperature, power, density) and reactivity void effect; effective portion of delayed neutrons; transient characteristics for normal operation modes; residual energy release, activity of spent fuel (RP SNFA).

It is planned to make calculations of normal operation modes (steady state modes and non steady state modes at minimum controllable power level) and at power when rods operate in transient modes, fuel burnout).

As constant software it is recommended to use software of constant preparation CONSYST with library БНАБ-93.

This version GEFEST800 is a development of its previous versions for RP FB-600. the software has a modular structure. The main modules of SW are: modules of preparation of neutron constants; modules of neutronic calculation of reactor in three-dimensional geometry in multi-group diffusion approach both with a point on FA and with a point on fuel element; module on the base of Monte Carlo for nuclear safety assessment; module for energy release calculation; module for calculation of fuel and absorber burnout, module for calculation of neutron fluence and radiation load on structural materials; module of heat hydraulic calculation for accounting of feedback in neutronic calculations; module for temperature calculation of structural materials in spent fuel assembly in the coolant medium or gas medium; module for calculation of residual heat removal; module for calculation of function of value and estimations of reactor parameters by the perturbation theory; module for calculation of effective functionals of reactor kinetic; module of non steady calculation (direct and reverse task) in quasi-static approximation; module for assessment of design errors and block of modules for work with fuel archive.

The plan is that the work of the complex will be implemented in software casing to enable user's services (monitoring of parameters, schedule, preparation of data, analysis of calculation results) when work with it in interactive mode.

Simulating of accident type Black Out for RBMK-1000 and use of mobile equipment for preventing severe consequences

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Accident at Fukushima Daiichi set the task of defining the technical and organizational measures, which implementation will enable decrease fission products release to atmosphere under the extreme impact of natural and man-induced origin. Within the frames of this task solution the decision was made on equipping the power units of Russian NPPs with mobile pump stations and diesel generators that will enable to ensure fuel cooling when long lasting house power loss.

The stages of Black Out accident type evolution are considered in the report for RBMK – 1000. Three main stages of accident evolution were considered:

1. Stage of coolant reserve boil-off in reactor and subsequent warm up to temperature ~ 650 °C. At this stage of accident evolution no additional damage of fuel and channel pipes is expected.

2. Stage of start of core degradation when warm up of dried core in the temperature range from 700 to 1450 °C. At this stage of accident the damages of fuel elements cladding and FC are evolved, rupture of FA frame rods is possible. When temperature of CPC rods reaches 1450 °C cladding of absorbing rod of cluster control rod starts to melt and absorber can be dropped out of the core.

3. Evolution of reactor core degradation up to fuel melting: heat up to temperature of zirconium melting, formation of fusion and its movement to the lower part of the core and then to lower baffle and lower plate of the reactor. Calculations show that such a state is subcritical but restoration of the core cooling in such a state is nuclear hazardous.

The report presents the calculated investigations of the reactor cooling restoration due to mobile pump stations, ways of use the «internal» reserves of the power unit: water stock in deaerators, water stock in main tank of reactor cooling system. The need of pressure decrease in reactor is demonstrated when there is a risk of temperature increase in channel pipes up to 650 °C. Time estimations of accident evolution are given for every stage. Based on time estimations of duration of the first stage of accident the recommendations are given regarding the time of mobile systems deployment and fulfillment of actions for opening of pressure reducing valves of the power unit to decrease pressure inside reactor.

Ways of the accident management at the second stage of accident are discussed and overview of hazards when cooling restoration at the second stage of accident is given.

For the stage involving fuel melting the task is set to retain fuel within the boundary of reactor metal structures and assessments are given to the technical measures required for this task implementation.

Forecast assessments of neutronic and thermal hydraulic characteristics of RBMK-1000 after performance of work on restoration of long-term performance of graphite structure

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Activities conducted on RBMK-1000 power units for restoration of long-term performance of graphite structure destructed as a result of long time operation require an independent design analysis of reactor parameters. When performance of repair-recovery activities the part of graphite structure mass is lost due to cutting of graphite units, which causes the change of uranium-graphite ratio, which, in its turn, impacts the neutronic and thermal hydraulic characteristics of reactor. Reactors of the first power unit at Leningrad NPP are operating now after completion of activities on restoration of long-term performance, therefore, the forecast of reactor parameters change obtained on the base of design codes of better analysis seems to be urgent.

Forecast assessments of RBMK-1000 neutronic and thermal hydraulic characteristics change when operation of the RBMK-1000 first power units after completion of activities on restoration of long-term performance of graphite structure given in justifying documents of JSC «Concern «Rosenergoatom» that underwent expertise were received with the help of large-grid diffusion programs SADCO and TROYKA. Based on experience, diffusion models in some cases, particularly when probable power tilt and uranium-graphite ratio change by reactor core due to random cutting of graphite columns cannot describe the exact distribution of energy release in reactor. Therefore, it seems to be important to obtain the assessment of RBMK-1000 parameters based on better assessment codes that is free from the specified drawbacks.

The goal of the work is an independent forecast analysis of neutronic and thermal hydraulic characteristics of reactors with restored graphite structure. To fulfill the stated goals the independent design investigations were performed on the base of software complex of better assessment BARS. Weak sides were detected that require corrective measures to be taken.

Investigations for justification of process heat reliability (departure from nucleate boiling) of fa operation in RBMK-1000 FC of larger diameter with regard of possible distortions

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Deformation of graphite structure observed at RBMK NPPs and respective changes of geometrical characteristics of fuel channels (FC) set an issue about process heat reliability of fuel operation within the final period of RBMK operation. The main parameter that features safety of heat removal from fuel element is the departure from nucleate boiling, showing heat flow critical power ratio (i.e. power when reach of critical heat flow becomes possible) to its current capacity.

Based on the evolved situation, JSC «NIKIET», NITS KI and JSC «EREC» developed the program for justification of heat reliability (departure from nucleate boiling) of fuel assemblies (FAs) operation in fuel channels of larger diameter taking into account distortions during extended life time (ELT). The program was approved by the management of JSC «Concern Rosenergoatom» and taken for implementation.

Need in experimental study of departure from nucleate boiling under the conditions of changed geometry of RBMK channels was caused by the fact that nowadays the ratio for the critical heat flow can only be obtained based on generalization of the experimental data and the experimental data for departure from nucleate under the conditions of changed geometry of RBMK channels were absent.

It was decided to conduct experimental study on large-scale heat hydraulic test bench of RBM-1000 – PSB RBMK located in JSC "EREC" (Elektrogorsk, Moscow region). Test bench of RBMK PSB simulates one loop of the circuit of multiple forced circulation of RBMK (MFCC) and includes the models of all main elements of reactor circuit. RBMK PSB test bench is intended for both direct simulating of emergency modes at RBMK-1000 and for conduct of experiments related to research of separate processes and phenomena specific for RBMK-1000 including for determination of conditions of departure from nucleate boiling occurrence. In order to investigate the conditions of departure from nucleate boiling occurrence one of six channels of the test bench was involved in FC of RBMK-1000 (full-scale models of FC RBMK-1000) re-equipped specially for such experiment.

Experimental study was also conducted for conditions of departure from nucleate boiling occurrence in full-scale model of FC RBMK-1000 with rated (80 mm) and enlarged (84 mm) inner diameters, under coaxial and eccentric arrangement of FAs regular and step-vise heating of dummy fuel elements, with simulation and without simulation of channel deflection. The tests were carried out in the following range of scheduled parameters change:

pressuree about 7 MPa, water temperature at input and output from 230 to 265^{0}C, flowrate from 10 to 30 t/h.

The obtained experimental data were included into the library of thermophysical data bases of JSC «NIKIET» and were used for verification of design code BUNDLE BM DF, according to which thermo physical reliability of FA is justified (departure from nucleate boiling) at enlarged diameter of FC and account of distortion during ELT.

Solution of scientific and technical problems of the final stage of RBMK operation

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1 Brief history of RBML operation.

1.1 Shrinkage of openings in graphite units and radial swelling.

Replacement of channels (single-type and stage-by-stage).

1.2 Axial and radial shrink of the fuel cask second set, exhaustion of fuel cask bellows travel, growth of fuel cask inner diameter.

1.3 Vertical shrink of graphite columns (problem of route telescope joint).

1.4 Secondary swelling of graphite units, their deformation and cracking, distortion of setting and channels.

2 Repair-recovery activities for decrease of distortion and rate of setting distortion growth (technology, effectiveness, ways of optimization, main outcomes and tasks).

3 Justification of safety and criteria of operability of reactor internal units.

3.1 Strength analysis and dynamic processes.

3.2 Thermal dependability of FAs.

3.3 Operability (mechanical, overloading) of FAs and CPS rods.

3.4 Analysis of neutronics characteristics and thermal mode of setting.

3.5 Analysis of design based and beyond design based accidents.

3.6 Additional SR&EDAs and verification of developed design codes.

4 Activities on upgrading of FA and CPS rods (CCrods, EP).

5 Tasks resulting from the analysis of beyond design based accidents at RBMK power units.

Ensuring of RBMK-1000 fas serviceability at the final stage of power units operation

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Nowadays deformation processes related to beginning of the secondary graphite swelling and, hence, cracking of the units of graphite structures were already recorded at 4 power units RBMK-1000 of the 1-st generation (power units №№ 1 and 2 of Leningrad and Kursk NPPs). The above defect appeared in the form of distortion of fuel (process) channels (FC) and control and protection system (CPS) channels detected during performance of routine inspection of their geometrical characteristics within the period of planned maintenance of power units.

In the long term, as graphite structures accumulate the damaging fluence of fast neutrons, the deformation processes similar to those recorded at the first generation power units at Leningrad and Kursk NPPs, particularly, distortion of FC and CPS channels, will appear at other power units with RBMK-1000 reactors.

The evolved situation required to carry out a complex of experimental studies aimed at definition of the boundaries of FAs serviceability in new, from the view of FC geometrical characteristics, operation conditions with regard to:

- possibility and conditions (efforts) of performance of FA loading to and unloading from FC;
- change of FA operation conditions in FC at power due to re-distribution of clearance between the cladding of fuel elements of external row and inner surface of channel pipe.

As a result of the executed work, serviceability of RBMK-1000 FA of standard and advanced structures (with central fixing of fuel elements) under the conditions of graphite structure deflection was demonstrated.

In 2013 for the first time since 2000 reduction of fuel duty was recorded at RBMK-1000 power units.

Urgent issues still remain that are related to:

- possibility of FA failure with leak of fuel element like «plug tearing-off», with a large portion of probability that fuel will spill to MFCC coolant;
- elongation of fuel elements during operation process, which stipulates reduction of clearance between the upper and lower bundles that hinders spent fuel assembly preparation for putting for dry storage.

Under these circumstances it would be advisable to resume the work on implementation of advanced fuel assembly (FA) (with central fixing of fuel element) with fuel enrichment profiled by height to ensure the following:

- further increase of fuel duty;
- elimination of fuel spilling to the coolant when fuel elements leak;
- unhampered preparation of SFA for putting for dry storage.

Topical area

MAINTENANCE, REPAIR AND INSTALLATION OF NPP EQUIPMENT

Optimization of process operations using RPS during power units repair at Smolensk NPP

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- Goals for reduction and optimization of work duration
- Analysis of critical route of repair schedule
- Schedule computing in SW Primavera
- Combination of work
- Search for possible ways to reduce the critical work
- Rational sequence of work in the critical way
- Elaboration of corrective measures
- Mitigation of delay between the partially loaded time intervals
- Coordination of work on the level of shift assignment
- Computer aided process flow control system
- Set up of complex teams for prompt mitigation of defects detected during equipment tests.
- Mitigation of wait related losses
- Mitigation of possible losses and reduction of the number of additional operations
- Extra income from electric energy sail due to reductions
- Additional reduction of repair deadlines
- Optimization of process flow

Equipment and welding technique of pipelines Dnom 850 of MCP of VVER-1200 in narrow bevels

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Till now the main method of connecting the joints of MCP bimetal pipelines Dnom850 mm has been manually operated arc welding.

Taking into account the growing number of nuclear power units to be mounted the lack of qualified experts for manually operated welding is becoming an urgent problem. Efficient solution of this problem when VVER main circulation pipeline mounting is transfer from manually-operated to automatic welding.

The technique of automatic argon arc welding of pipelines of VVER main circulating pipeline Dnom 850 in narrow bevels and also in traditional bevels has been investigated and developed. Welding was performed using orbital automatic welding machines Polysoude(France) and Arcmachines (USA). For argon arc welding of steel 10ГН2МФА specially developed welding wire Св-10Г1СН1МА was used.

Comparative study of the process capabilities of welding equipment has been made. Real-size mock ups produced from bimetal pipes Dnom850 were welded. Non-destructive tests were performed and mechanical properties determined in the scope of qualification tests. Metallographic studies of the fusion zone of pearlitic metal of joint and austenite buildup depending on the technological methods of welding were conducted.

Development of norms of inventories for operation and maintenance needs of nuclear power plants

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Norms of inventories are an integral part of effective operation management of nuclear power plants as an important part of production and commercial activity. Both safety and reliability of NPP operation depend on proper calculation of these norms.

For justified introduction of norms for inventories the document MT 1.2.6.2.0112-2012 «Introduction of norms of inventories to ensure operation and maintenance needs of nuclear power plants. Typical model» was developed and put into force.

According to the methodology the norm of inventories is a planned value (measure) of level of inventory needed for ensuring a steady production process at NPP with compulsory observance of all requirements to safe operation. For this purpose the entire stock is subdivided to the below types:

- main (standard) stock covering operation, maintenance and repair inventories by its separate groups;
- minimum level of stock including separately inventories for maintenance and repair, in-service needs, inventories stock for civil defense and emergencies;
- repair stock of inventories for NPP;
- centralized stock of JSC separately including reserve stock and centralized repair exchange stock.

Operation experience data including statistic of deviations and disturbances in NPP operation, expert appraisals of nuclear power plants and CO of JSC «Concern Rosenergoatom», as well as information about inventories spent within the period since 2002 through 2013 were the basis for introduction the norms.

After development of norms to all above types of inventories the differential calculation of their costs was made as well as recommendations for improvement of stocks management were prepared.

Prevention of origination and evolution of intergranular corrosion cracking formed under stress conditions in welded joints of austenite pipelines Dnom 300 at RBMK-1000

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This report presents the main results of application of technologies for prevention of origination and evolution of intergranular corrosion cracking formed under stress conditions (ICCSC) in the weld-affected zones of welded joints of austenite pipelines Dnom300 (CC Dnom300) at RBMKs-1000 when emphasis is made on the results and perspectives of application of technology of mechanical re-distribution of residual welded stresses (technology MSIP).

Taking into account that till now the number of CC Dnom300 of «pipe – pipe» type at all power units with RBMK-1000, whereto the technology MSIP was applied, amounts to 9090 pieces (tentatively 50% of all CC Dnom300), it would be advisable, from the view point of the authors, to considerably speed up solution of the artificially supported problem of ICCSC for CC Dnom300 by way of distributing MSIP to CC Dnom300 of types «pipe - nozzle equipment/valve», «pipe – cone reducer», «pipe – T-joint nozzle», to which till now only measures of organizational nature have been applied through the entire operation period.

In their report the authors presented the results of experimental validation of the use of MSIP technology for the specified types of CC Dnom300, confirming the effectiveness of MSIP technology use as a counter measure for prevention of origination and evolution of intergranular corrosion cracking formed under stress conditions.

Based on the analysis results of ten-year experience of the use of MSIP technology for CC Dnom300 at RBMKs-1000 the recommendations are presented in the report for enhancement of effectiveness of the use of MSIP technology results addressed at trouble-free operation of austenite pipes Dnom300 towards process, operation and organization directions.

Experience of solution the task of equipment development and supply for NPP new power units with the set of repair documentation

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Below are knowledge required for prompt and high quality performance of equipment maintenance and repair:

- Strategy of maintenance and repair performances (TASK);
- Technique of maintenance and repair performances (PROCESS);
- Requirements for equipment technical condition (RESULT);

History of repair documentation development for NPP equipment.

Contemporary requirements for the composition and content of maintenance and repair documentation.

Order issued in JSC «Concern Rosenergoatom» on solving the decision of development and supply of equipment for new power units of NPP with the set of repair documentation.

Set of repair documentation developed by JSC «VNIIAES» for steam generator PGV-1000MКП for engineering company «ZIOMAR» (Podolsk) performing manufacturing and supply of equipment for new power units with VVER-1200.

Brief characteristic of developed documents – Programs for maintenance and repair, technical specification for repair and set of process documentation.

Implementation of pneumohydroimpulse method of NPP turbo-generator oil-duct cleaning

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Results of analysis of disturbances in operation of automatic regulation and protection systems of the turbines became the background for implementation of pneumohydroimpulse method of NPP turbo-generator oil-duct cleaning.

The main reasons of disturbances and corrective measures indicate the deficiencies of the existing methods of NPP turbo-generator oil ducts cleaning. With the aim of enhancement the efficiency of the existing methods of turbo-generator oil ducts cleaning JSC «VNIIAES» proposed to use the pneumohydroimpulse method.

When pneumohydroimpulse method of cleaning the compressed air is delivered to the washed circuits, which facilitates oil foaming, increase of its volume in the point of air delivery and enables better quality washing of the head oil ducts and bigger surface of discharge headers as compared to hydrodynamic method. Supply of hydraulic and pneumatic impulses also facilitates the expansion of washed surface of discharge oil ducts during cleaning.

Assessment of effectiveness of pneumohydroimpulse method, based on the results of experimental washing at the NNPP allowed for making a conclusion on the favorable impact of the implemented cleaning method on the quality and reliability of hydraulic regulation systems operation and reduction of the number of failures of automatic regulation and protection systems due to poor quality of turbo-generator oil ducts cleaning.

The main ways of effectiveness enhancement and reduction of labor intensity of pneumohydroimpulse method of cleaning

The main ways of effectiveness enhancement and reduction of labor intensity of pneumohydroimpulse method of cleaning address the automation of washing devices. JSC «VNIIAES» performed development of design documentation of automatic washing devices taking into account the detected defects during trial washing.

Technique of pneumohydroimpulse method of turbo-generator oil ducts cleaning provides for only oil ducts cleaning and is not related to cleaning of «washing» oil. This study contains the recommendations for post-washing cleaning of in-service («washing») oil and requirements imposed on the quality of turbine oil.

Development and implementation of the complex of special-purpose equipment for NPP equipment and pipelines repair

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Set of special-purpose equipment (SSPE) is intended for power operated restoration repair of the unit for coolant header welding to nozzle Dnom1200 of steam generator PGV-1000 at VVERs-1000.

The purpose of SSPE development is ensuring high quality repair, reduction of labor intensity of repair activities and reduction of time of the personnel staying in the radiation exposure zone.

The below aspects were submitted in the engineering proposal prepared by JSC «GMZ «CHIMMASH» in 2012:

- assessment of quality picture of damages of the unit to be repaired in the area of welded joint №111 and description of size characteristics of local damages;
- statistical analysis of size characteristics of damages, which enables to detect the number of regularities of arrangement of local damage centers in the unit;
- quantitative justification of selection the design bevel of repair joint using the results of statistic analysis of size characteristics of damages

and comparison with traditional bevel used when repair in the area of welded joint №111;

- design process flow of power operated repair of the unit;
- assemble solutions of SSPE main elements including fitting for mechanical treatment, automatic welding, support-mounting system and special-purpose tooling for coolant header unfastening.

Original elements of the developed set are support-mounting device for mounting the driven equipment on the unit being repaired and pipe cutter with updated drive that can be equipped with a special inserted blade milling cutter.

The set enables to perform a circular cut out of the stripe of damages and form the facets for repair bevel by lathe machining, cutout the local damages by milling. Filling of repair bevel is performed by automatic argon-arc welding by the nonconsumable electrodes and filler wire heating. Welding apparatus is equipped with video system for remote monitoring of the welding process. Welding apparatus and welding process are managed from the programmed source of welding current.

Equipment for SSPE control is installed in the normal access area remote from the unit being repaired and is connected with the driven equipment through the compact cable cord. Control cabinet has the block for registration of welding parameters in real time with their graphics delivery to display, and also video monitors for depicting the welding area both before and after electrode.

In 2013 all components of SSPE were manufactured and delivered on Novovoronezh NPP site whereon, based on the decommissioned steam generator PGV-1000, the laboratory bench structure was prepared for SSPE test and experimental run of repair technique of this unit.

Currently, in cooperation with the specialists of Novovoronezh NPP maintenance department the experimental run of repair technique is going on. JSC NPO «TSNIITMASH» and JSC OKB «GIDROPRESS» are involved in this work.

Earlier a similar welding complex was developed and delivered to Kola NPP for repair of welded joint №23 on steam generator PGV-440.

Information system «Portal of pre-operational tests support»

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JSC «Atomtechenergo» is s business-unit of electric power division of JSC «Atomenergoprom» and performs all types of pre-operational tests (POT) for all ranges of NPP systems and equipment. POTs incorporate development of coordination plan and pre-operational documentation, development and keeping of budget and financial documentation, supervision of tests conduct and submission of reporting- acceptance documentation and other types of activities. In order to provide computer aided support of the part of the main activity of the plant, to increase effectiveness of POT monitoring and control, increase the activity and encouragement of the personnel in the area of innovation activity of the organization as well as to reduce the time period of the projects implementation inside JSC «Atomtechenergo» the development of information system «Portal of pre-operational tests support» was initiated.

The portal represents a complex of soft and hardware ensuring a single point of access (portal solution) for the consumers to use the set up functions in compliance with the assigned access rights.

The portal ensures the solution of the following tasks:

- set up of the unified system for projects control;
- ensuring information support of the main activity of the plant;
- supervision of work performance deadlines fulfillment;
- organization of the unified information space for the territorially allocated company units;
- computer aided support of the part of the routine processes in order to reduce the time period of the work performances;
- enhancement of activity and effectiveness of innovation activity;
- ensuring the possibility for the users to promptly receive information about the subject of their interest;
- providing a unique tool for communication of the users.

Soft and hardware complex of Portal consists of a central server including a cluster of two server processors, central file storage and storage for duplicated copies as well as associated servers of the structural units of the company and client workstations. The whole soft and hardware complex of Portal will undergo certification for compliance with class 1Г of the system protection against unauthorized access.

Portal consists of 12 subsystems, seven of which are directly connected to POT performance (for example, subsystems for POT support, documentation storage, planning and management etc.), and remaining five subsystems implement the support and additional functions (administering, communication, informing and others).

Within the time period from March through July 2014 in one of the affiliated branches of JSC «Atomtechenergo» within the frames of the project of production system «Rosatom» «Optimization of the process of pre-operational documentation issuance for power unit №1 of NV NPP-2» one of the modules of the Portal - «Pre-operational documentation management» underwent an experimental and industrial operation. Results of experimental and industrial operation of the module showed a considerable improvement and enhancement of effectiveness of the process of pre-operational documentation issuance. Currently, the active development of the functional of other subsystems and modules of Portal is going on.

Simulators of maintenance processes for NPP equipment based on virtual reality

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Training of maintenance personnel plays an important role in operation and maintenance of nuclear power plants. Traditional methods of personnel training using full-scale mockups of equipment become inaccessible when we deal with large-size and expensive equipment.

To solve such tasks the computer interactive dynamic training systems are developed using virtual reality technologies based on which simulators of maintenance processes for Leningrad NPP are being developed. The main task of such simulators is obtaining, learning and test of knowledge on maintenance technologies of various equipment.

The main training element – introducing of training situations using three-dimensional graphics. Graphic models of equipment parts have been developed in a strict compliance with design documentation of real equipment. Maintenance processes and operations are introduced by animations simulation in three dimensional space that reflect visually the entity of a specific element of process flow.

Simulator enables to depict additional textual, graphic, audio and video information. To ensure simulation of man's actions with the objects of virtual space the man-machine interaction system was developed.

Several training methods are supported in the simulator; they can be used depending on a final objective of training and on the level of personnel qualification as well.

Nowadays the training systems created on the base of technology of virtual reality became an integral part of the process of NPP maintenance personnel training; they help to receive and learn the necessary knowledge by method of active interaction of trainee with graphic images of parts, tooling and equipment in three-dimensional virtual space and, thus, to advance the qualification of personnel and, consequently, NPP safety.

Industrial technologies for restoration of RBMK-1000 graphite structure long-term performance

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Large-scale overhaul was performed at power unit 1 of LNNP within January-August 2013 for straightening of graphite structure columns of RBMK RP -10000.

At power unit 2 of Kursk NPP in autumn 2013 the task was set to transfer the technique of graphite columns straightening to the industrial level: accelerate repair rates not less than two times and reduce repair cost of long-term performance restoration in 3-4 times.

This task was solved by way of establishing the complex of simple and reliable repair equipment and organization of complex team from the personnel of NPP involving the personnel of developing company. Experience gained at LenNPP was used in a full volume.

Comparative analysis showed the absence of significant variations of neutronic characteristics, thermal mode of graphite structure and other parameters of reactor before and after long-term performance restoration. Currently, the power unit is operating at rated power.

This report views the solution of the task for principal speed-up and cheapening of repair of RBMK-1000 graphite structure from the point of the tool that was used for graphite structure repair.

A complex of special-purpose repair equipment was developed:

- device for graphite cutting;
- device for calibrating the opening in graphite column;
- system of removal of graphite cutting products,
- device for powerful straightening of graphite columns.

The main principle of development was to solve the problem by a maximum simple and advisable method. Apart from the minimum deadlines of work performance such an approach will also enable to minimize the costs.

Training of repair personnel was conducted on the special test bench that was manufactured at Kursk NPP specially for this purpose. All activities related to graphite cutting and other process operations were carried out by the personnel of Kursk NPP. Specialists –developers of JSC «Prolog» continuously witnessed the work for prompt solution of problems arising during repair of graphite structure.

In the course of work the equipment was upgraded for enhancement of its efficiency and reliability and for simplifying the work on this equipment as well.

In the long view we see the possibility of further perfection of repair technique and repair equipment to reduce the time period of long-term performance restoration to the time period of routine PPM and reduction of costs.

PLACE OF M&R IMS IN TODAY'S SYSTEMS OF ASSET MANAGEMENT. STANDARDS ISO 5500X

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SpetsTech is a professional consultant in the area of assets management of enterprises, a lead Russian developer of software and approaches for management of assets and processes of their maintenance and repair (M&R).

In the area of its speciality NPP «SpetsTech» masters the methodology international and national standards defining requirements for the assets management systems, safety engineering and health protection, ecological aspects, quality and has more than 20 years of such systems implementation. Since 1994 NPP «SpetsTech» deals with development, evolution and supply of EAM-systems TRIM and implementation of information management systems on its base.

On the basis of «Scientific-production enterprise «SpetsTech» the technical committee for standardization №086 was established « Asset Management » (TK86). Committee TK86 was established as an analogue of TK251 «Asset Management» of International Standardization Organization (ISO).

Today's information asset management systems provide collection, storage, processing and analysis of information about assets, which is required for the organization to perform management of assets during their life cycle.

In April 2005 the information support system of the plant operation management was handed over for commercial operation at Smolensk NPP. This system was built on the base of software complex TRIM developed by NPP «SpetsTech» and was named «Desna-2». In the end of 2009 the system was expanded due to the start up of inventory subsystem.

More than 1500 users have been registered now in «Desna-2» and this system includes data base of equipment, planning of work on regular inspection of technical condition, management of the processes related to planned or out-of-schedule repairs, management of inventories, analysis and assessment of effectiveness (quality) of work, informational support of managers and experts of NPP with respect to the issues of safe and cost-effective operation of NPP.

When analyzing the above specified standards, one can state that «Desna-2» should be developed in the following directions:

- Set up interconnections between strategic measures of organization and M&R activities performed;
- Planned expenses for resources to be specified in periodical works;

- Annual requests shall be produced on account of planned expenses for work and information about the balance in warehouses;
- Integrate with the contract department system for obtaining the information about execution of supply requests;
- Integrate with the systems of APCS and SCADA implementing the registration and monitoring of parameters of assets and processes;
- Collect and analyze the data on the technical condition delivered from various sources, which will enable to point out the groups of types of equipment according to service strategy: performance of repair requests wherever it is possible and advisable or preventive types of service are required with account of condition included.

Subsection 1.2
ENGINEERING SUPPORT TO NPP OPERATIONS

1.2.1. DIAGNOSTICS, ENHANCEMENT OF THERMOMECHANICAL EQUIPMENT RELIABILITY, MODERNIZATION AND LIFETIME EXTENSION OF NUCLEAR POWER UNITS

Intelligent systems of nondestructive testing and technical diagnostics – the basis for NPP safe operation

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Improvement of reliability of nuclear fuel utilization facilities under conditions of extension of service life of facilities requires evaluation of remaining service life of equipment at all stages of its life cycle, and the same evaluation us required for all of its components. Specific nature of nuclear power industry has determined the wide use of monitoring and diagnostics means, which possess a number of special features. Wide range of recent achievements in the area of equipment, arrangement and design decisions and technical means of registration is used for improvement of reliability of heat-mechanic equipment, materials study and metal control, diagnostics and extension of NPP service life.

Radiography and ultrasonic methods are widely used at monitoring of welding joints and junctions of nuclear machine building. NDT means developed in RDI of introscopy for oil and gas and air-cosmic industries can be applied at construction and operation of NPP: Ultrasonic analyzers and tomographs, which are widely used for control of welding joints of oil and gas pipelines. Meters of speed and time of ultrasonic longitudinal waves transmission in concrete, intended for concrete strength control and assessment of crack depth.

Safety can be viewed on the basis of the scheme of NTD and TD system as intelligent, into which the following is included:

- NDT and TD means,
- Personnel including training and certification,
- Regulatory documents including metrology and certification.

Such approach to sensors construction even today allows implementing tendencies manifested in the recent years and improving them.

Implementation of system of automatic control of remaining life (RLACS) at the stage of power unit extension of service life

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Power unit extension of service life is connected to performance of range of activities of its preparation for further operation.

Requirements used for development of the program of modernization of the power unit are recorded in regulatory documents NP-017-2000 «Main requirements to power unit extension of service life» and STO 1.1.1.01.006.0327-2008 «Organization standard. Power unit extension of service life». Among them there is a requirement on necessity to develop the Program of management of long-term performance characteristics, the content of which is regulated by the standard of Concern «Rosenergoatom» «management of long-term performance characteristics of power units of nuclear power plants». CTO 1.1.1.01.007.0281-2010.

In order to «manage equipment operation life», predicting possibility of repair/replacement/modernization of element in advance, it is necessary to have the prognosis of remaining life, which can be obtained by the use of the methods of continuous analytical diagnostics of metal condition, which together with periodical application of NDT will allow to manage long-term performance characteristics. Despite the fact that nondestructive testing (NDT) is direct physical method of detection of equipment metal damage, it has a number of peculiarities. For one thing, NDT detects the consequences of metal damage in the form of cracks of different kinds and can not give quantitative evaluation of the size of accumulated fatigue damage. Besides, NDT can only be performed during PPM, and stress concentration areas for RP of VVER type are located on the internal surfaces of equipment and pipelines, the access of personnel to which is restricted, and correspondingly there are a great number of areas not suitable for control.

System of automatic control of remaining life (RLACS) is suggested for use as the method of continuous analytical diagnostic of metal condition.

When implementing RLACS at the power units, which are under operation and subject to service life extension, it is necessary to take into account the following problems:

- RP equipment metal has partially worked out its service life, and it is necessary to quantitatively evaluate the value of accumulated damage;
- there are initial imperfections (sores, corrosion, case of damage with subsequent repair, etc.);
- the power unit has sufficient scope of modern monitoring of operation parameters;
- the necessity to monitor stressed state and remaining life of damage areas during operation (for example area №111).

Monitoring of engineering structures, buildings and facilities of NPP

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At present JSC «VNIIAES» carries out operating support of industrial diagnosis system (IDS) of JSC «Concern Rosenergoatom», to which diagnostic information arrives from 1, 2 and 3 units of Kalinin NPP.

In the current year it has been planned to extend the capabilities of IDS on account of:

Connection to IDS of unit №4 of Kalinin NPP and completion of creation of prototype model of IDS at VVER RP NPP.

Creation of prototype model of IDS at RBMK RP NPP (Smolensk NPP).

Also, one of the most prospective trends of extension of IDS capabilities is its addition to the system of geotechnical monitoring (GMS) and systems of engineering facilities monitoring (IFMS) of NPP (the basis for establishing such systems are GOST 22.1.12-2005 and GOST R 53778-2010). The main purpose of these systems is execution of integrated monitoring of condition of engineering structures, buildings and facilities of NPP. The issue of creation of such systems is relevant; it has already been started at some engineering facilities of RF (NPP, TPP, HPP). Different sensors (strain, power, pressure in the soil, tilt meters, piezometers, extensometers, fiber-optic strain sensors, etc.) are included into such systems. They allow monitoring values and speed of change of controlled parameters to a high precision. Information received from sensors is subjected to evaluation of conformance to pre-established operating parameters, and serves as the basis for determination of technical condition of monitored building, facility and/or facility of NPP (both at the stage of construction and operation).

At present JSC «VNIIAES» carries out works on development of technical assignment and technical requirements for establishing and implementation of integrated monitoring and diagnostics system of containment of NPP with VVER-1000 RP. Development of integrated system will allow to receive information on containment current technical state and other information required for development of routine preventive measures as per results of measurement of set-up parameters in selected points and areas during NPP unit operation.

JSC «VNIIAES» develops feasibility assessments for development of monitoring systems of turbine unit foundation and geotechnical monitoring of NPP and HPP.

Measurement information from GMS and IFMS transmitted to IDS will enable to carry out analysis of technical state of equipment, buildings, facilities, elements of construction of NPP facilities in a more prompt and qualitative manner.

Integrated use of monitoring and diagnostics systems of equipment, buildings and facilities, and geotechnical monitoring integrated into IDS will increase efficiency of NPP power units operation.

Technical diagnosis as the element of NPP life cycle management

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In this report concept suggested by the authors is presented – «Technical diagnosis as the element of NPP life cycle management». Creation and development of this concept meets the requirements of the «Federal law of nuclear power utilization» of the Russian Federation, corresponds to aims and tasks of development of diagnostics, requirements of other regulatory documents, rules, etc.

Increased design service life of new NPP power units requires more meaningful substantiation of provision of conditions of safe operation. There is a set of IAEA recommendations on provision of long-term operation, which can be effectively used at commissioning of each NPP power unit.

The benefit for new NPPs from the use of diagnostics means at all stages of life cycle management consists in provision of uninterrupted operation, switchover to maintenance based on technical condition and, ultimately, to drop of costs of possession of key assets of operating organization and receiving maximum profit.

The concept «Technical diagnosis as the element of NPP life cycle management» considers the place of technical diagnostics in management of NPP life cycle, substantiates purpose and scope of application of integrated technical diagnostics, limits of its functionality, and gives evaluation to expected technical-and-economic results from diagnostics implementation at each stage.

In course of work on suggested approach, analysis of necessity of diagnostics of equipment, systems and facilities at all stages of management of life cycle was carried out. Stages of creation of «Digital dossier of NPP unit» with account for diagnostics data at all stages of life cycle management were presented.

Three-dimensional information model of NPP power unit as the basis for problem solving of creation of industrial system of diagnostics and management of service life characteristics of NPP power units

V.L. Tikhonovskiy

The task of management of service life characteristics of elements of NPP power units throughout all stages of life cycle with development of industrial system of diagnostics and management of service life characteristics is becoming more actual and problematic with each passing year. Above all things, this is related to the main tendencies currently existing in the nuclear power industry:

- in accordance with the «Program of activities of State enterprise of nuclear power «Rosatom» for long-term period (2009–2015)», the main equipment of NPPs will operate at enhanced modes;
- gaining additional power production is mainly realized by implementation of activities of increasing installed capacity and increase of NPP power units LF (load factor);
- number of power units which have exceeded design lifetime increases;
- existing competitive struggle at the market of electric power forces to search for cost cutting reserves regarding operation of NPP power units (particularly, for their maintenance, restoration, service life extension).

In spite of seemingly increasing part of tasks of management of service life characteristics, currently there are a number of problems in the area. Among others – lack of centralized information system of accounting and analysis of all aspects of activities in the area of LTPM. Besides seemingly increasing part of such system, its development is also demanded by national regulatory documentation and IAEA recommendations. Complexity and multidimensionality of tasks of management of service life characteristics (detection and study of ageing processes; evaluation of technical condition of different types of elements of power units; accounting and analysis of the history of maintenance, modernizations, replacements; cooperation of different organizations and informational exchange between them, etc.) will require participation of different organizations when solving issues related to creation of industrial information system of diagnostics of power unit elements and support of LTPM. Besides NPP and operating organization, participation of chief designers and general designers of NPP power units will also be required; organizations performing scientific-research support of power units operation, as well as other key specialized organizations.

Besides brief review of requirements of standard of management of service life characteristics of elements of NPP power units, currently existing problems of LTPM, report also contains premises for automation of tasks of information support of LTPM, goals and tasks of industrial information system of diagnostics and management of service life characteristics, as well as organizational and technical proposals on its creation on the basis of three-dimensional informational models of NPP power units, containing data about topological characterization of power units (with information required to solve many practical tasks).

New diagnostic properties in industrial diagnostics system of Concern Rosenergoatom

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Industrial diagnostics system (IDS) of Concern RosEnergoatom, which allows to:

- detect mutual diagnostic information distributed at different local diagnostics systems;
- apply unified calculation procedures for accumulated DB in industrial diagnostics;
- perform remote integrated diagnosis in automatic and automated mode, was under experimental operation during 2013.

By way of analysis of long-term statistics of time and assembly of NPP units:

- neutron-thermohydraulic sources supplying information on reactor core under operating conditions were formalized;
- new numerical measures (diagnostic properties) with familiar names: «energy intensiveness», «local channel-wise flowrate of coolant», «total general circuit flowrate of coolant» were engineered;
- noise measurement method of boric acid concentration in coolant was developed.

It was confirmed by experiment that:

- different fuel cycles have significantly different fields of highest types of standing acoustic waves (SAW), therefore undesirable triple matching of harmonics of RCP rotation frequency, free-running frequency of element of FA construction and frequency of the highest type of vessel SAW is possible;
- two main factors – vibrational and thermohydraulic – determining operational reliability of FA, change from one fuel cycle to another, as well as in the course of one cycle, which requires continuous monitoring.
- emergence of gas space in the upper part of SG head, under RP lid and, especially, gas space (explosive mixture) under cases of CPS CR significantly decreases the frequency of corresponding SAW.

Integrated control system of VVER RP coolant leak

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Main provisions of integrated system of automated leak control (ALCIS) are stated. Set of ALCIS means is intended for control of equipment and pipelines leak tightness at technically complex and potentially dangerous industrial facilities, on-line detection of leaks, assessment of their size and coordinates based on analysis of data coming from autonomous subsystems of leak control. ALCIS contains five independent separate subsystems of leak control, which operate based on different physical principles, complementary to each other.

Development of algorythmic support of CLMS of pipelines and equipment of NPP pressure circuits

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Timely leak detection allows to prevent spreading of accidents caused by loss of RP equipment integrity, and thus, enhances NPP safety.

In order to detect coolant leak and determine its parameters (mass flowrate and location) automatic coolant leak detecting system (ACLDS), based on analysis of several mutually independent physical quantities, was developed and implemented (currently implemented at 11 power units with reactor plant of RBMK type and at 2 power units with reactor plant of VVER type). The system allows to detect a leak at the early stage of its occurrence and control; its development, giving exhaustive information to operating personnel for making decision on taking corresponding timely measures. However, taking into account operating experience and putting into effect of federal law 102-ФЗ «On ensuring uniformity of measurements», it was required to ensure stricter requirements to the system, namely:

- expand the range of measurement of the value of mass flowrate of leak to the value of 10÷1140 kg/h;
- reduce the time for detection and measurement of mass flowrate of leak from the moment of leak appearance within the measurement range of 20 minutes;
- ensure 50% limit of allowed relative accuracy of measurements of the value of mass flowrate of leak;
- limit absolute error of measurement of leak location by each coordinate to 2 meters.

In order to meet the above requirements algorithms using methods of monitoring based on different physical principles, combination and mutual complementation of which allow to satisfy all requirements imposed on leak detection systems were developed. Methods of monitoring used in ACLDS include: monitoring of acoustic pressure in controlled room, monitoring of moisture content of air medium, monitoring of temperature field of air medium and monitoring of volumetric activity of aerosols in controlled room.

Arrangement of sensing devices for each subsystem of leak control was carried out upon results of experimental-calculation justifications in co-operation with GNTS FEI and thermophysics department of NIKIET. Calculations were carried out with certified program codes (Ansys, KUPOL-M, COCOSYS) and were confirmed both by experiments at test base of NIKIET and with the help of information on real operation of systems at nuclear power plants.

Enhancement of diagnostics systems using operating experience feedback from NPPs with VVER reactors

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JSC «Scientific-engineering center «Diaprom» is one of the main suppliers of on-line diagnostics systems of NPP reactor plant with VVER.

Starting with 2004 JSC «SECD» has developed, manufactured and maintains support of operation of the following diagnostics systems of VVER RP NPP:

- vibration monitoring system (VMS, VNDS);
- loose parts monitoring system (LPMS);
- travel control systems (TCS);
- integrated diagnostics systems (IDS);

VMS, LPMS systems are operated at Russian and foreign NPPs:

- VMS, LPMS at units 3,4 of Kalinin NPP, unit 2 of Rostov NPP, TCS at unit 1 of KLN NPP;
- VNDS, LPMS at units 1,2 of NPP «Tianwan» (China), VMS, LPMS at units 1,2 of NPP «Kudankulam» (India), unit 1 of «NPP «Bushehr» (Iran), unit 4 of Rovno NPP, unit 2 of Khmelnitski NPP (Ukraine).

Over systems operation time great experience was accumulated. It allowed to:

- detect deficiencies in operation of systems and eliminate them in the course of operation;
- develop industrial technical requirements to VMS, LPMS systems;
- develop diagnostic procedure of vibronoise diagnostic systems;
- develop instructions on personnel actions at detection of eliminations from design conditions.

Main lines of works on improvement of systems:

- improving diagnostics quality by development of stability criteria and diagnostic passport of channels for recording their actual quality.
- development of new trends in diagnostication of RP equipment and pipelines.

Extension of service life of power units of Russian NPPs

V.A. Gilev

Extension of service life of power units of operating NPPs after exhaustion of designated service life period is still one of the most relevant tasks at the current stage of development of nuclear power industry in Russia and is the most economically efficient way to invest money into production of nuclear energy due to retaining of generating capacities of NPP power units upon condition of assurance of acceptable safety level.

As of March 30 of 2014 works on SLE of 19 NPP power units of total installed power of 11 802 MW were performed. Performed works have substantiated the possibility of safe operation of power units beyond assigned operation life. Rostechnadzor licenses for operation during additional period were obtained in the established order.

Power units with extended service life have produced more than 350 bln. KW.h of energy, and their total capacity is 46,7% from installed capacity of operating power units of NPPs of the Russian Federation.

In the long term before 2023 JSC "Concern Rosenergoatom" has set not less difficult and ambitious task of keeping operable of 13 more NPP power units of total installed power of 10 737 MW.

Currently programs of preparation for additional service life period are under way at 10 power units: unit № 4 of Kursk NPP; unit № 4 of Kolskaya NPP; unit № 2 of Smolensk NPP; units № 1, 2 of Kalinin NPP; units № 1 - 4 of Balakovo NPP, unit of № 4 of Novovoronezh NPP. 3 SLE investment projects are at the stage of development: units № 1,2 of Kolskaya NPP, unit № 3 of Smolensk NPP.

Results of extension of service life of power units NPPs provide energy preparedness and socioeconomic stability in the country due to guaranteeing minimum tariff burden, maintaining power balance of regions before the start of commissioning of new power units, preserving scientific and technical and industrial potential in Russia.

Extension of service life of VVER NPPs up to 60 years

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Extension of service life of VVER NPPs is currently relevant in many countries which utilize nuclear energy. The relevance is primarily determined by the fact that significant part of NPP power unit has finished (or close to finishing) its established design service life.

JSC «VNIIAES» under the frameworks of performing works on service life extension (SLE) of NPP power units with VVER-1000 successfully performs the following types of works:

- development of regulatory-engineering and procedural documentation on management of long-term performance characteristics of elements;
- development and implementation of modern means and procedures of diagnostics and non-destructive testing of elements of power units;
- integrated inspection systems of power units;
- assessment of technical state and remaining life of equipment, pipelines, engineering structures of buildings and facilities;
- development and maintaining of data bases of long-term performance characteristics of elements;
- development of management programs of long-term performance characteristics of NPP elements during additional service period, taking into account the main mechanisms of ageing of materials and stress history;

For all power units of VVER-1000 NPPs operating life is extended to 30 years. By now JSC «VNIIAES» has accumulated experience of evaluation of technical state and remaining life of more than 250 units of different types of heat-mechanic equipment (heat-exchanging equipment, tanks, pumps, filters, etc.) and 50 systems of pipelines.

JSC «VNIIAES» has implemented the model of industrial diagnostics system (IDS) of JSC «Concern Rosenergoatom» at Kalinin NPP. The system automatically collects information coming from diagnostics systems installed at power units №№ 1÷3 of Kalinin NPP and sends it to industrial diagnostics center. Works on expanding IDS by connection of diagnostics systems installed at other NPP power units to it are carried out.

The following local data bases of long-term performance characteristics of NPP elements have been developed and maintained:

- Extension of service life of elements of NPP power units;
- on technical state and operating conditions of SG.

Local data base of service life extension is intended for storage of documents on SLE of power units and SLE of NPP elements, as well as information on long-term performance characteristics and conditions of extension.

In order to increase efficiency of execution of works on LTPM, execution of works on SLE of NPP power units work to create Industrial information system of support of LTPM of NPP power unit was started in JSC «Concern Rosenergoatom».

Improvement of reliability of operation of equipment, pipelines, and engineering structures of buildings and facilities of NPP requires development and implementation of on-line diagnostics systems, allowing to perform fast evaluation of current state of metal, determine its remaining with account for conditions and operating life terms.

Integrated testing of actual state of equipment and facilities of power units № 5, 6 of NPP «Kozlodui»

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At present 5 and 6 power units of NPP «Kozlodui» of overall capacity of 2 GW are operated in the Bulgaria grid. 1-4 power units of NPP «Kozlodui» were shut down upon demand from EU in the course of entrance of the Republic of Bulgaria into European Union.

5 and 6 power units of NPP «Kozlodui» were put into commercial operation in 05.11.1987 and 29.05.1991 correspondingly. At present design service life of the power units is limited by 2017 for the 5th power unit, and by 2021 for the 6th power unit. After Bulgarian government rejecting the project of NPP «Belene» construction the issue of retaining of available capacities has gained strategic importance for the Republic of Bulgaria. Therefore NPP «Kozlodui» has planned a set of activities subdivided into stages on extension of service life of 5 and 6 power units of NPP «Kozlodui». The first stage stipulated integrated testing of actual condition and evaluation of remaining life of equipment and facilities of 5 and 6 power units. Based on testing results Programs of preparation of power units for additional service life period are developed.

Upon results of open international tender performed by NPP «Kozlodui» the winner of tender for integrated testing execution (hereinafter – IT) of actual condition and evaluation of remaining life of equipment and facilities of 5 and 6 power units of NPP «Kozlodui» was declared consortium including JSC "Concern Rosenergoatom" and «Electricite de France» (France). On 19.04.2012 signing of contract for execution of works on IT between NPP «Kozlodui» and the Consortium took place.

The subcontractors of Concern became two leading Russian companies: JSC «Atomtechenergo» and JSC «OKB Gydropress», as well as Bulgarian company «Risk engineering» EAD. A range of Russian company was engaged into performance of works on integrated testing of NPP «Kozlodui»: JSC

«OKBM im. Afrikantova», JSC «TSKBM», JSC «Atommashexport», FSUE «NIIP», JSC «NPF «TSKBA», JSC «VNIIAM», OOO «Resurs».

Performance of IT works on equipment and facilities of 5 and 6 power units of NPP «Kozlodui» was carried out stage-wise in accordance with time schedule of the contract. During the first stage a list was drawn up and analysis of regulatory-engineering documentation (RED) of the Republic of Bulgaria, Russian Federation, France (EDF), as well as IAEA recommendations used in the process of IT was performed.

With the purpose of description of general approach to performance of works integrated testing of actual state and evaluation of remaining life of equipment and building structures of 5 and 6 power units of procedure of IT execution was developed. When developing the procedure experience of JSC "Concern Rosenergoatom" and experience of EDF I the area of management of ageing of equipment, facilities and extension of service life of NPP power units was used.

Based on the procedure development of IT programs as per the types of investigated equipment was performed.

In the process of IT execution 5863 structures, systems and components (SSC) of 5 and 6 power units of were inspected. Upon results of integrated testing, as well as experience of extension of SSC life at Russian NPP, specialized organizations have developed recommendations, which were presented in IT reports and reflected in the program of unit preparation for additional service life. At the stage of IT preliminary evaluation of service life of RP equipment and pipelines was carried out.

The results of IT have revealed that technical condition of SSC, engineering structures, buildings and facilities of 5 and 6 power units of NPP «Kozlodui» complies with requirements of operating, design-manufacturing and regulatory documentation effective NPP «Kozlodui» and allow to make a conclusion on possibility of extension of service life of power unit №5 to at least 17 years beyond design terms and power unit №6 to at least 20 years beyond design terms. Amended operating life of 5 and 6 power units of beyond design terms will be determined after execution of planned activities stipulated in the programs of preparation of 5 and 6 power units of NPP «Kozlodui» for additional service life.

This project has become a large-scale project in the filed of nuclear power engineering, realized by Russian companies on the territory of European union and the first commercial project, jointly put into life by JSC "Concern Rosenergoatom" and EDF company (France) over the whole of 18-year history of cooperation of the two companies.

Service life extension of spent nuclear fuel storage (SNFS) of Leningrad NPP

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Over a period 2010 – 2012 a complex of works on extension of service life (hereinafter - SLE) of spent nuclear fuel storage, bld.428 (hereinafter - SNFS) with the purpose obtaining license for its operation during additional service life period was performed at Leningrad NPP for the first time.

Due to lack of experience of SNFS life extension as independent nuclear fuel utilization facility, at the first stage main required scopes of work, deadlines, financing and responsibility for execution were specified in accordance with corresponding organizational documents of JSC "Concern Rosenergoatom", Moscow.

General approach at work performance related to SLE SNFS stipulating obligatory observation of requirements established by Federal norms and regulations in the area of nuclear energy utilization «Requirements to substantiation of possibility of extension of specified service life of nuclear energy utilization facilities» NP-024-2000, with the possibility to be guided by standards and regulatory documents of JSC "Concern Rosenergoatom", Moscow related to area of activities of «Service life extension of operating NPP power units» during execution of works was developed.

The following main works were performed within the framework of SLE of SNFS:

- Execution of integrated test of SNFS of Leningrad NPP;
- Development of Program of SNFS preparation to service life extension based on results of integrated testing and safety evaluation;
- Performance of the set of works in accordance with the Program of SNFS preparation to service life extension;
- Performance of procedures of obtaining license for SNFS operation during additional service life period, including expert analysis of substantiating documents;
- Issuance of Report on completion of works as per the Program and Decision on SNFS service life extension.

As per results of work in accordance with approved by JSC "Concern Rosenergoatom", Moscow and SE «Rosatom» «Decision ЛЕНАЭСР-494K(04-08)2012 on continuation of operation of spent nuclear fuel storage (SNFS, bld.428) Leningrad NPP», additional service life term of SNFS, bld.428 of Leningrad NPP was established— 30 years from August 30, 2012. Rostechnadzor license for operation of SNFS of Leningrad NPP till 2032 was received.

Positive results of performed works allow to make a conclusion on the possibility to use the experience of Leningrad NPP SNFS service life extension in affiliates of JSC “Concern Rosenergoatom”, Moscow.

Program of improvement of safety level of Ukrainian NPPs

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The report contains information on implementation of measures on improvement of safety of Ukrainian NPP power units and site territories related to service life extension.

Safe and moreover accident-free operation of NPP has always been and is an absolute priority of the work of SE NAEK «Energoatom».

Giving the highest priority to the issues of safety improvement conforms to provisions of Convention on nuclear safety, which was ratified by the law of Ukraine as of December 17, 1997.

It should be noted that after the accident at NPP «Fukushima Daiichi» Ukraine has been implementing the so-called «post-Fukushima» measures developed upon results of «stress tests». Implementation of «post-Fukushima» measures is under constant supervision by regulatory authority, IAEA and international community. Among other the following stages of these activities implementation can be noted:

- TWS «Strategy and technical means of overcoming the consequences of beyond design accident "Long blackout of NPP site with loss of ultimate heat sink " for the power units with RP B-320» was developed and approved.
- DDW «On measures of prevention of early loss of LSS-HES integrity in case of reactor vessel failure and release of molten masses of reactor core beyond its boundaries» was developed and approved.
- DDW «On implementation of system of monitoring of hydrogen concentration in SV for beyond design accidents at power units with VVER-1000 (B-320)» was developed and approved.
- Concept decision «Development and implementation of activities on reduction of hydrogen concentration in SV for beyond design accidents » was developed and approved.
- Technical requirements to equipment of the system of emergency hydrogen removal from accident localization area by localizing safety system were developed and approved.
- DDW «On implementation of system of forced pressure drop from HES of power units by NPP SP with RP of B-320 type» was developed and approved.

Safety enhancement is carried out within the framework of safety improvement programs.

As of today the main program is Integrated (summary) program of safety improvement of power units of nuclear power plants of Ukraine, which was initially put into effect by joint order of Mintopenergo and Gosatomregulirovanie in 2010. Almost immediately after the events at NPP «Fukushima Daiichi» in Japan, the status of the program was upgraded to governmental. The corresponding order of putting the program into effect was signed on December 7, 2011. At that the program was supplemented by the measures as per results of unscheduled task-oriented deep safety reevaluation in terms of power units' resistance to external extreme conditions (the so called «stress-tests»).

According to the order of CMU IPSI is underway from 2011 to 2017.

Since the majority of power units of Ukrainian NPPs was commissioned during 1980s and has design service life of 30 years, one more priority of the Company is the preparation of NPP power units to operation beyond design service life, which is closely connected to safety improvement. In accordance with requirements of Gosatomregulirovanie implementation of IPSI is an essential condition of continuation of NPP power units operation.

1.2.2. ENHANCEMENT OF ELECTRIC AND I&C EQUIPMENT RELIABILITY

Experience of implementation of MB RPAD in electric installations of NPP

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Microprocessor-based NPP technical means are installed for implementation of the below-mentioned functional secondary devices.

Relay protection devices – relay protection of power lines, collecting busses and elements of main circuit of power output, auxiliary SG-6 kV.

Devices of electric automatics - automatic controllers of turbine generators excitation, control of position of OVR switch of power transformer, switch control.

Emergency automatics devices – nodal and terminal devices of system emergency automatics (EA) and EA at PTL, transceivers of commands of system emergency automatics.

Automated devices of measurement, control and display of teleinformation DPDS ASSO and AIIS KUE. Communication devices.

In the course of execution of the program of RPA devices based on electromechanic and electronic element base replacement with Microprocessor-based relay protection and automatics devices (MB RPAD), drawbacks in part of organization of personnel training and supplies of technical means (terminals) were revealed.

Even if under general assessment installed microprocessor technical means did not affect the reduction of reliability level, some incorrect actions revealed problems related to their implementation.

With many positive features of microprocessor technology, a stricter approach to observation of logical sequence of elements and integrity of the process on the whole is required: design – equipment purchase – installation and commissioning.

Long-term performance management of cables at nuclear power plants: certification and diagnostics

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To implement modern methods of management of long-term performance characteristics (LTPM) of cables new revision of technical document "Provision. Determination and technical condition and management of ageing of cables at nuclear power plants" (PO 1.2.1.02.999.0184-2013) was developed.

Long-term performance management (LTPM) of cables at nuclear power plants – types of activities, operational activities, and methods used to maintain serviceable state of cables within acceptable limits for safe and economically efficient operation of NPP. Certification and diagnostics are the main types of activities. The new revision of Provision presents requirements and recommendations for execution of typical certification testing and nondestructive testing of cables at NPP. The provision was developed on the basis of long-term experience of execution of works on LTPM at NPP and IAEA recommended practices.

The Provision includes:

- main long-term performance characteristics of cables;
- tasks solved when performing works on LTPM;
- procedure of execution of typical certification testing with detailed description of all stages, activities and recommendations for proper execution;
- the reasons for conventionalism of typical certification testing and uncertainties occurring during their execution;
- LTPM measures performed directly at NPP units;
- brief description of cable condition diagnostics and control methods which passed verification and validation;
- recommendations on informational support of technical diagnosis;
- recommendations on development and implementation of cable LTPM programs at a power unit.

The Provision is intended for both NPP personnel and specialists of organizations rendering technical assistance to NPP within the framework of works on LTPM of cables.

Diagnostics of technical condition of power cables with impregnated-paper insulation at nuclear power plants

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Diagnostics procedure of technical condition of power cables with impregnated-paper insulation (IPI), which takes into account occurrence and development of typical defects at long-term operation of cables at NPP power units, was developed.

The procedure suggests execution of periodical fast diagnostics by personnel of electrical department (ED) and in-depth diagnostics of cables by specialized organization. To perform fast diagnostics the following indicators of condition (CI) are used: K_a absorption coefficient, PI polarization coefficient and insulation resistance standardized for the cable of 1 km $R_{\text{from 1 km}}$ length. It is recommended to combine fast diagnostics with the performance of standard tests by increased voltage of power cables. Purposes of in-depth diagnostics: determination of defective areas of cable routes and evaluation of service lives of cables at the moment of CI management. It should be noted that in-depth diagnostics allows to reveal defects of "insulation shrinkage" type, which can often be detected even at execution of regular cable tests by increased voltage, which bear the signs of marginal state. During execution of in-depth diagnostics the following activities are performed: thermal imaging testing, insulation resistance measurement, restored voltage measurement, measurement of internal discharges at oscillating die out strain (OWTS method), space-time reflectometry, frequency-dielectric spectroscopy (FDS). Restored voltage measurement method is the main method of integral estimate of condition of IPI insulation, OWTS method – the main method of determination of defective cable sections. FDS method is the additional control method (control of water content) for confirmation of probable cause of local insulation leaks: dampening or paper carbonation.

The developed methodology, initial and limit values of CI are given in technical document MP 1.2.02.0168-2013 "Diagnostics of technical condition of power cables with impregnated-paper insulation at nuclear power plants".

Results of monitoring of technical state of turbine generators and power transformers with measurements and location of vibration and electrodischarge defects at non-live grounded parts under operating conditions by portable diagnostics means

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Results of technical diagnostics of NPP turbine generators will be presented in the report. Interrelation was determined between:

- vibration events conditioned by availability of defects at units of stator fastening due to loosening pressing and opening of end packages, fastenings of coil-heads and embedded coils, fastenings of stator core to the body;
- results of measurements and location of electrodischarge events in stator winding (surface partial discharges) and in the core (sparking at crowns of cogs).

The said approach was also used for power transformers, for which peculiarities of development of defects in the package of active steel were registered as per the following stages: occurrence of increased vibration at harmonic waves (200-400 Hz), then occurrence of sparking in package, and, finally, occurrence of thermal gases dissolved in oil.

Experience of implementation and operation of insulation monitoring system «EKRA-SKI» in automatic direct current systems of Russian power facilities

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From 2006 OOO NPE «EKRA» has been conducting works on development and implementation of insulation monitoring system, which would allow to automate the process of insulation monitoring and search of damaged connections, and not cause false actuation of RPA and EA devices, into the networks of automatic direct current of Russian power facilities (HPP, NPP, SS and TPP). The first prototype of insulation monitoring system «EKRA-SKI» was submitted into experimental operation in 2008 at SS «Bugulma-110». In 2009 the first two «EKRA-SKI» systems within direct current boards were supplied to Kaliningrad TPP-2. From 2010 «EKRA-SKI» were supplied to Kursk NPP (4 power unit, OS), Beloyarskaya NPP (OS), Balakovo NPP (OS). Currently approximately 100 «EKRA-SKI» terminals and 3000 sensors of differential current have been installed into networks of automatic direct current of three NPPs, three HPPs, four TPP and more than 40 SS. The advantage of «EKRA-SKI» system is the fact that it operates simultaneously with existing traditional circuit of insulation monitoring on the basis of two resistors connected between poles of grid and relay of PH51/32 type. With the help of «EKRA-SKI» insulation monitoring systems, not only insulation monitoring and search for damaged connections were automated, but during implementation connections with damaged insulation were detected.

Availability of sensitive sensors of differential current in «EKRA-SKI» system allowed to find defects in connections, which haven't been found earlier by manual search of earth connections, namely – galvanic connection by one of two poles of storage batteries, sections of direct current board or several loads. Such defects in connections lead to occurrence of «ring» currents not related to insulation damage. Detection of places of galvanic connection is a difficult task, which requires solution in accordance with, for example, STO of JSC «FSK EES» «Systems of automatic direct current of substations. Technical specifications», which allows to lower the possibility of accidents in automatic direct current of substations network.

In the course of adjustment of «EKRA-SKI» insulation monitoring systems at some NPPs presence of alternating differential currents in connections was detected in automatic direct current network. The occurrence of these currents is related to the use of some types of thyristor charging-recharging devices with increased level of output voltage ripples, as well as defects at laying of cables in automatic current system when positive current lead of one load lies near negative current lead of another load. At that the form of differential current in connections has variable component of 300 Hz, the value of which reaches several hundreds of milliampere at the impulse. Such differential current is not related to insulation resistance and can be classified as interference. The curve shape of this interference and its value are different from those regulated, for example, by standard of JSC «FSK EES» by electromagnetic compatibility.

Differential current sensors, which have enhanced interference immunity, were developed in RPE «EKRA». However, it should be noted that it does not cancel the necessity to use charging-recharging devices with low level of voltage ripple in automatic direct current network, as well as skillfully lay cables in channels.

In 2013 SPE «EKRA» completed development of portable device of insulation control working together with «EKRA-SKI». It allows to simplify the search for connections with damaged insulation at those connections where stationary differential sensors are absent. First prototypes of «EKRA-PKI» portable devices of search for feeders with damaged insulation were handed over to Kursk NPP and Kalinigrad HPP-2 for experimental operation.

Groundwall protection of electrical equipment

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One-phase short circuits are a very complicated type of damage and, if possible, shall be disconnected quickly. Current protection meets the main requirements imposed on relay protection, - selectivity, fast response, sensitivity, reliability, in the best way. However, requirement of selectivity of current protection – ability to immediately disconnect only damaged section of network at short circuiting – is not always realizable. There are so-called «dead zones», for example, the section of circuit between circuit breaker and remote-mount current transformer, protections in which do not work selectively, which is a significant disadvantage of known current protections. Short circuit in these areas can only be eliminated by breaker failure redundant installation (BFRI), which disconnects all connections of the section with the damaged area. BFRI actuates after time delay, which can result in significant damage of electric equipment in the damage area.

Suggested groundwall protection of electrical equipment allows to disconnect one-phase failures with absolute selectivity and at maximum speed, including those at the sections of the so called «dead zones» where known protections do not work selectively. Groundwall protection device includes current sensor registering the current of electric equipment ground fault with relay element actuating disconnection of circuit breaker of the damaged line. Sensitive element of current sensor implemented in the form of Rogowski coil or optical fiber embraces the foundation (support structures) of electric equipment and does not require modernization of existing foundations (support structures) of electrical equipment of 110 kV OS and higher.

This design is patent-protected and can be used for protection of electrical equipment at failures resulting in short circuiting to the ground, which is the most frequent type of failures in the networks with dead-earthed neutrals (voltage of 0,4 kV, 110 kV and higher). This protection system can be easily integrated into effective system of NPP electrical equipment protection, making it an integrated solution, which can lead to significant decrease of damages of electrical equipment and, correspondingly, to increase of general safety level of NPP operation.

Specification of HFC of RCP, determination of additive corrections of loop thermal control at nominal power (determination of mass-average temperature of coolant in RCC loops) for exact calculation of RP thermal power as per primary circuit parameters

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One of the main parameters of VVER NPP power unit is weighted average RP power, to determine which calculation of RP power by primary circuit parameters is applied. The main and the most important thing at calculation of RP power by primary circuit parameters is most precise determination of values of flowrate and temperature of coolant in hot and cold legs of RCC loops.

The main problems at calculation of RP power by primary circuit parameters at nominal power are:

- accuracy of determination of weighted average coolant temperature in hot RCC legs due to stratification (thermal stratification) along pipeline cross-section,
- accuracy of determination of RCP by method with the use of HFC (head-flowrate characteristic) obtained at benchmark tests.

Based on data received at execution of pre-operational tests at NPP «Kudankulam» the report shows:

- high accuracy of determination of coolant temperature in ICIS with the help of 6 RT (resistance thermometers) in each RCC leg,
- insignificant stratification within the limits of $\pm 0{,}2^0$C at the power of ~50% $N_{rated.}$ in each of loops at operation of all RCPs and in two loops with working RCPs when one RCP is disabled.
- insignificant stratification within the limits of $\pm 0{,}2^0$C at the power of ~75% $N_{rated.}$ in two out of four loops when all RCPs are operating.

High measurement accuracy of coolant temperature RT in RCC legs and insignificant stratification of coolant in cold legs at the power of ~50% $N_{rated.}$ when all RCPs are working, as per individual loops at the power of ~75% $N_{rated.}$ and at different combinations of working RCP allows to adjust HFC of RCP by using the method of OJB GP described in 320.00.00.00.000ПМ1. GKAE OKB «Gydropress» Appendix 3.

The method is based on application of heat balance between primary and secondary circuits in individual loops. For precise calculation of loop power by secondary circuit parameters at measurement of SG feedwater flowrate it is necessary to use ultrasonic flowmeter «FLUXUS», which provides measurement error of not more than 1%. Regular flowmeter devices provide measurement error of ≈2% at rated flowrate (at decrease of power the error increases).

At precise calculation of coolant flowrate at nominal power, by using clarified HFC RCP, with a help of balance tests it is possible to calculate additive corrections of RT (weighted average temperature) of hot legs at nominal power and perform precise calculation of RP power as per primary circuit parameters.

Measurement of feedwater flowrate with fluxus attachable ultrasonic flowmeters

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Recently the issue of improvement of reliability of nuclear power plants has become more relevant. For this purpose new engineering designs are engaged, including instruments and meters of the latest generation. One of such devices is attachable ultrasonic flowmeter Fluxus. These flowmeters allow to measure flow rates of different liquids, including feedwater without disturbing pipeline integrity. Therefore, additional risks related to cutting-ins in pipes are gone. Also non-availability of contact with the medium increases service life of flowmeters, reduces the risk of failure. Servicing of attachable flowmeter (for example, calibration), as well as installation/demounting can be performed without production shut down, which enables to perform process modernization works under operating reactors.

Fluxus attachable flowmeters have relatively high precision, which allows to increase reliability of some processes at the plant. In order to increase measurement stability two sets of primary ultrasonic transducers connected to one electronic unit can be simultaneously mounted onto the pipelines. Spaced placing of sensors and electronic unit allows to optimize allocation of components of the flowmeter for convenience of operation. Electronic unit may contain different types of output signals and digital protocols, which allows to integrate Fluxus flowmeters into any control systems. Also the device has internal energy-independent archive, which also records parameters of self diagnostics into the memory, enabling to assess the quality of flowmeter operation. To assess metrological characteristics special algorithm of error calculation, allowing to evaluate real error of each measurement point was developed and certified. A number of distinctive aspects, such as the use of two measurement methods (time-impulse and doppler) with the possibility of automatic switchover, compensation of difference of temperatures between sensors (according to ANSIASME standard), the possibility of measurement of flowrate of liquid mediums with temperatures of up to + 600^0C make Fluxus flowmeters the world leader among attachable ultrasonic flowmeters.

First these flowmeters were used at nuclear power plants of France and USA. Currently Fluxus flowmeters are the only ultrasonic flowmeter recommended by EDF concern for use at nuclear power plants, and to be used at all NPPs in France. As of today in Russia a number of tests to confirm the possibility of application of Fluxus flowmeters for measurement of feedwater at different NPPs in Russia have been carried out, particularly at units №3 and №4 of Leningrad NPP and unit №4 of Kursk NPP.

Magnetic resonance flowmeters – relaxometers for monitoring of heavy water parameters in cooling system

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Development of scientific-and-technological advance has led to necessity to widely implement automatic process control, including nuclear power plants (NPPs). In this case the role of precise and reliable control and measuring equipment significantly increases. Different liquid mediums (water, heavy water, etc.) are used in NPP production processes. Flowrate and sometimes even the composition of these mediums shall be monitored to a high precision. The biggest difficulties occur when measuring flowrate and parameters of heavy water condition, which can have different temperature, quickly change its flow and contain bubbles in it. Regular contact devices using traditional measurement methods, when used under such conditions for a long time, start to generate great measurement error, primarily due to destruction of measuring elements owing to temperature cavitation. Necessity to develop new ways of measurement allowing using non-contact instruments of high-quality accuracy occurs. One of these instruments is nuclear magnetic measuring devices, operating principle of which are based on the phenomenon of nuclear magnetic resonance (NMR).

Label NMR flowmeters – relaxometers are successfully used for monitoring of flowrate and composition of flowing medium. These completely non-contact measuring devices do not require preliminary calibration and allow to perform measurements on a real time basis. Information about measured medium parameters arrives in the form of electric tension, which allows to space apart magnetic system of the instrument and detecting device from electronics operation of which is more subject to influence of different factors, to significant distance.

The main parameters of any liquid medium, which can be used to immediately determine the change of its physical state (temperature increase, occurrence of other substances in it, dissolved or not, impact of radioactive radiation on it), are longitudinal T_1 and transverse T_2 relaxation times. Currently temperature control does not constitute a problem, and two other parameters can be monitored by measurement of constants of relaxation as per the form of registered NMR signal, as well as flowrate of liquid medium.

Apart from various design decisions and electronic circuitry, the authors have developed a procedure of processing of NMR registered signals with the use of wavelet – transformations allowing to determine: occurrence of foreign elements in liquid medium (because of internal destruction of product pipeline) and radiation impact on medium.

Monitoring of operational characteristics of current ionization chambers within CPS of reactors of RBMK-1000 type

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This scientific-engineering report is devoted to monitoring of operational characteristics of current ionization chambers and research of stability over time of readings of chambers per se and equipment processing the signals from these sensors.

Ionization chambers are ex-core sensors of signals in the systems of monitoring, control and protection. They are gas ionization detector for monitoring of neutron flux density.

At NPPs ionization chambers are used within so called suspensions of ionization chambers. Suspension of ionization chambers is constructional unit which fixes the position of ionization chambers in reactor and protects it from external environment impact and from electromagnetic blasts.

This work suggests continuous monitoring of electrical and radiometric parameters of current suspensions of ionization chambers within CPS of RBMK-1000 type reactors on the basis of readings received from PPDMS (physical power density monitoring system) equipment and equipment monitoring parts of KSKUZ.

Regarding the way of obtaining diagnostic information, this technical diagnostics is classified as functional, that is, it is only performed on operating equipment.

In the course of work, connections between characteristics of diagnostic parameters and the state of facility were established, and diagnostic algorithms required for determination of type of technical condition were determined.

Application of this method of non-destructive testing of current ionization chambers and primary equipment of signals processing will allow:

- to detect operational defect of detachable connections of suspensions of ionization chambers and primary equipment of processing of signals from current ionization chambers;
- to detect approach of the moment of marginal technical state of ionization chambers suspensions;
- to monitor burning-out of radiator of suspensions of ionization chambers in time for determination of possibility of their further operation upon expiration of specified lifetime;

- to predict failures of suspensions of ionization chambers;
- to reduce stock resources in accordance with the ideology of Production system of «Rosatom» by determination of optimum quantity of stock of suspensions of ionization chambers;
- to accumulate statistical data useful for developers of suspensions of ionization chambers.

Thus, application of this method will allow to increase reliability and efficiency of operation of suspension of ionization chambers within CPS of RBMK-1000 type reactors.

Control of ageing process of CPS electrical equipment

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At extension of equipment service life equipment ageing plays an important role. At the same time, increased impact of production risks of different nature and consequences brought by them are almost completely ignored by production and management practices.

Up to present the task of service life extension of specific equipment has been considered in isolation from:

- the risks which have not been formulated at all and not described;
- economics.

At that it is required to not only describe the risks, the prognostication, but the methods of their management.

This is related to objective difficulties of description of connection of scientific apparatus with the theory of decision making when it comes to the object which is a complex system.

Currently mainly quantitative methods of evaluation of reliability characteristics are applied. At the same time, modern methods of nonnumeric mathematics of prognostication of qualitative properties, for example, electrical equipment of NPP CPS have not been spread widely due to difficulties of establishing of connection of scientific apparatus and qualitative estimations.

Methods which allow to move from quantitative to qualitative estimations of monitoring of ageing process of NPP CPS equipment include:

1. Mathematical model of analysis of equipment reliability indices
2. Analysis of SPTA sufficiency
3. Power law model for forecasting change of equipment reliability indices with account for risk-management
4. Logic-correlation analysis
5. Analysis of consequences of failures (ranking of failures)

Logic-correlation analysis combines all the above-mentioned methods and allows to present the most precise assessment of reliability indices change, which was demonstrated by the example of electrical equipment of NPP

CPS. In comparison with analysis performed by separate characteristics, logic-correlation analysis describes reliability of CPS electrical equipment in the most concrete and realistic manner, since in logic-correlation analysis variables and criteria are not only interrelated, but also complement the properties of each other.

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Building diagnistic system of condition of information-measurement channels of VVER NPP ICIS on a real-time basis

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The specific feature of in-core instrumentation system (ICIS) is the fact that its operation not only directly affects safety, but economic efficiency of NPP power unit operation. Mainly, ICIS is used in most cases to calculate parameters monitored as per conditions of safe power unit operation and hand-over to shift operating personnel; it also generates signals of preventive and emergency protection as per in-core local parameters of energy release. Besides that, ICIS specifically determines average-weighted reactor heat power, which affects the calculation of technique-economic indices of power unit operation. Due to the above-mentioned circumstances, as well as taking into account development of informational technologies and accumulated experience, the task of expanding of capabilities of automated diagnosis of failures of ICIS information-measurement channels has become highly relevant.

Currently information on development of automated system of diagnostics of failure and unreliable information based on algorithms and methods used at execution of physical tests of ICIS during commissioning and industrial operation of power units with VVER is presented. These algorithms and methods have been successfully tested during commissioning of almost all power units of different projects with VVER-440 and VVER1000. Currently pre-operational documentation for power units with VVER-1200 is being developed on their basis.

Implementation of the described system is scheduled for commissioning of power unit №1 of Novovoronezh NPP-2.

Implementation of diversity in neutron flux monitoring equipment

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Due to development of technologies and broadening of spectrum of requirements, as well as owing to tendency for improvement of operational characteristic, the extent of application of software means for execution of safety functions and functions important for safety increases in modern NPP APCS equipment.

Recently great amount of regulatory documents establishing requirements and recommendations for stages of life cycle and components of software – hardware means was developed and implemented.

One of the main recommendations is the use of as the main instrument of protection from failures due to general cause related to software application.

At that regulatory documentation describes two types of diversity – software-hardware (when to reach one goal physically different means realizing the same function are used), or functional (when to reach one goal the same means performing different functions are used).

The concept of implementation of diversity in modern equipment of neutron flux monitoring equipment (NFME) developed for NPP power units of VVER -1200 type and VVER TOI is considered in this work.

The main decisions allowing to provide resistance to failures due to general cause by application of diversity in two NFME sets, not resulting on the other side to excessive complication of designing and servicing of the system, as well as not resulting in great appreciation of the system were considered.

Application of diversity at various levels of the system was considered –starting from devices of processing of signals of ex-core sensors of neutron flux, and complete with modules of generation of signals of protection and management and modules of main parameters calculation.

Integrated decisions for NPP APCS projects

OOO «Moscow plant «PHISPRIBOR»

1. Company experience

1.1 Stages of development of software-hardware means: from power sources and technical means based on «rigid logic» to digital management and control systems.

2. Company place at the market of automated systems

2.1 Development and manufacture of software-hardware complexes of safety controlling systems of nuclear power plants.

2.2 Product line. Completed projects: Balakovo NPP, Novovoronezh NPP, Rostov NPP, Kalinin NPP, «Kudankulam» NPP, NPP «Bushehr».

3. Investment policy of the company.

3.1 Advanced developments of technical means, software, certification of finished products. Development of processor modules (master controllers) of PowerPC architecture and x86 IntelAtom architecture.

http://ru.wikipedia.org/wiki/X86

3.2 Creation of digital control system of different level of backup as per input and output discrete and analogue signals, controllers, with the use of RS485 and Ethernet networks at the lower level and provision of 5 msec scanning cycle with accuracy of time mark assignment of 1 msec and analogue converting error of 0,1 %.

3.3 Implementation of diversity principle.

4. CADS, SCADA. Choice of optimum design medium.

Operating system and CADS of own design.

4.2 The medium of "through" designing and modeling of production processes – SimInTech. Certified in the system IT code generator.

4.3 Real-time protected operating system – RTPOT KPDA 10964.01 «Neitrino».Ministry of Defense Certificate (NSD, NDV).

4.4 Certification of software for use in safety systems and systems important for NPP safety.

4.5 Assurance of cyber security. Availability of source codes of software – transparence of design – possibility of independent verification.

5. Integrated approach to design and manufacture of equipment of control and management systems.

5.1 Process simulations – creation of control algorithms – virtual control boards – adjustment with the use of mathematical models – controllers setting – verification of control algorithms. Unified development medium.

5.2 Issues of equipment completing. Pirate products – decrease of I&C reliability. Incoming control of component parts. Uniform data base of suppliers of radioelectronic element components.

6. Sufficiency of company's process equipment.

- Two lines of surface mounting.
- Workshop for simultaneous assembling of up to 200 control cabinets.
- Minimum engagement of third-party organizations for manufacturing component parts (turning and cutter lathes with NPC, laser-beam cutting, bending press, powder spraying chamber)
- Coating with moisture-protection (automated line of two-sided lacquer coating).

7. Company test base.

- heat and moist chambers.
- Sand chamber.

- Salt spray chamber.
- Vibrostand (up to 300 kg).
- Equipment of execution of tests of electromagnetic compatibility EMC.

8. The test ground for adjustment and execution of integrated tests of systems of management and control. Equipping of test ground. Burn-in of I&C equipment.

Deterministic approach to evaluation of safety and risks of informaiton management systems for nuclear power plants

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At present development of new and modernization of existing nuclear power plants (NPPs) is one of priority lines of activities of all structures of SE «Rosatom». This requires creation of new complex information management systems (IMS) important for safety, on which strict requirement to provision of safety and efficiency of their operation are imposed.

Distinctive aspect of IMS creation for NPP is its development and implementation under conditions of risk occurring due to ambiguity of factors of internal and external environment of the project, which can lead either to deterioration of qualitative indices of developed IMS, or to excess of budgetary provision and/or violation of project deadlines, or to its complete failure. Inefficiency of probabilistic-statistical methods when designing many complex objects in engineering, economics, biology and medicine calls for necessity of change from solving of problems of building probabilistic models to creating deterministic models. As Director General of SE «Rosatom» S.V. Kirienko has emphasized: «It is necessary to avoid probabilistic approach when designing NPPs ... deterministic approach to safety assurance shall be used in the industry». In order to guarantee achieving high reliability and functional safety of IMS, it is required that during whole IMS life cycle, especially at the initial stages (feasibility analysis of IMS project, assessment of safety and risks of IMS) formulation of basic requirements, development of structure and arrangement of the system different approaches and methods are applied.

In spite of great number of works of many national and international authors devoted to development of IMS for NPP, the issue of project feasibility analysis, as well as assessment of safety and risks of IMS is still open.

In this works it is suggested to consider some aspects of this problem and bring special attention to relevant issues of analysis of project feasibility and assessment of safety and risks of IMS. Fuzzy set and fuzzy-interval approaches to feasibility analysis of IMS project on the basis of formation of comparative multiple criteria analysis of possible options and selection of acceptable alternatives under conditions of ambiguous source data have been considered.

Another prospective trend, which is suggested for consideration, is the method of precedents based on experience of prior situations, which allows to solve new, yet unknown task by using or adapting the decision of already known task under conditions of vague, non-precise source information.

Suggested approaches will allow to economize budget funds, as well as reduce administrative mistakes of project manager under conditions of vague, non-precise source information, and thus to increase the quality and efficiency of made decisions at development of new hardware means of IMS important for NPP safety.

Application of wireless passive sensors at piezoelectric elements of radiation-resistant piezoquartz to increase reliability of NPP moniroting and control systems

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The possibility of application of wireless passive sensors at piezoelectric elements of radiation-resistant piezoquartz for designing, development and implementation of system elements after emergency monitoring is considered in this work.

Possibility to use wireless passive sensors, which are able to work in the area of ionizing radiation, for collection of physical information for recording parameters of temperature, pressure, constructional material stresses performed on the basis of radiation-resistant piezoquartz is considered. Also principles of development of informational system of continuous flow registration and event processing based on the data base with open source code are considered.

The use of wireless passive sensors not requiring electric power supply allows to determine after-accident environment parameters, such as temperature, pressure, mechanical damages in elements of constructions. Measurements are performed with device based on radiation-resistant piezoquartz, operating on the basis of the principle of surface acoustic waves (SAW) in the area of high ionizing radiation and transmitted to remote registering device (radar), which is moved beyond high-radiation area.

Information flow registered by the radar is processed and then used as the basis for operation of system of support of operating decisions making.

Purposes and results of work, development and implementation of Instrumentation technologies at:

- reduction of terms and costs (prime cost) of system installation;
- reduction of the scope of process equipment;
- reduction of number of cables;
- increase of signal processing speed and reduction of general speed of decision making during post-emergency situation.

Results of analysis of reliability and safety of NPP CPS electrical equipment operation

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In order to ensure reliable and economically efficient production of energy operation of NPP power unit systems shall be accompanied by the management of long-term performance characteristics.

In this work experience of assessment of indices of reliability, remaining life of CPS electrical equipment at execution of works on extension of equipment service life, and methods of long-term performance management during additional service period was summarized.

Modern approach to reliability management recommended by IEC standards and applied for performance of works on extension of service life of CPS electrical equipment is based on risk analysis.

Risk – a combination of probability of occurrence of process equipment and its consequences. For example, failure of some component or subsystem can lead to failure or deterioration of equipment function, to necessity of a power setback or power unit shutdown.

The process of risk analysis shall include both qualitative identification of risk and quantitative assessment of risk value, and probabilities of failure of technical means with consequences of specific level.

When assessing equipment reliability and developing method of control of performance characteristics, risk analysis method adapted for CPS electrical equipment – «Analysis of type and consequences of failures» was applied. It consisted in:

1. Assessment of probability of failures occurrence
2. Ranking as per the severity of consequences of failures
3. Analysis of failures which occurred over operating period

These methods of risk analysis were applied by the authors when performing works on extension of service life of CPS electrical equipment of Kolskaya NPP, Bilibino NPP and «Tianwan» NPP.

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Innovations increasing safety and quality of tests of primary equipment

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The article describes mobile software-hardware complexes Omicron, which enable to automate testing of primary electric equipment and increase safety of test execution.

Brief description of three systems is given:

1. Portable system for tests and calibration TH Omicron VOTANO 100 intended to take characteristics and evaluate operability of inductive VT of up to 245 kV and capacitive VT of up to 750 kV.
2. Portable 3-in-1 unit for integrated check of high-voltage circuit breakers Omicron CIBANO 500, enabling to perform tests of three-phase power circuit breaker with two tears, grounded from both ends, for electrical, temporary and electromechanic characteristics.
3. Portable multi-channel system Omicron MPD 600 for measurement and localization of PD power transformers, electric machines, high-voltage cables and circuit-breakers, enable to measure and localize PD when strong electromagnetic interferences are available in the field.

Innovative approach to tests of primary equipment allows reducing the time of tests execution, increasing the safety of works and quality of assessment of power equipment condition.

1.2.3. MATERIAL SCIENCE AND METAL INSPECTIONS

Improving methods and means of operating non-destructive testing of coolant header welding to PGV-1000 steam generator nozzle DNOM1200

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Ultrasonic testing of the coolant header welding to VVER-1000 steam generator nozzle Dnom 1200 applying automation and technologies of original data record and startup visualization has been performed involving AVGUR system since 1999. From that time and up to the present time this technology underwent substantial changes, which concerned all main aspects of testing designation: beginning from accuracy and detectability to performance measures and operational comfort.

This report also considers tested and well-reputed methodological and hardware decisions for testing result automation and visualization concerning this unit, involving the up-to-date ultrasonic testing using antenna arrays. First results of applying new technologies using antenna arrays have been analyzed.

The report describes main current problems related to in-service inspection of the coolant header welding VVER-1000 steam generator nozzle Dnom1200; it considers ways of further UST improvement related to applying additional UST methods, such as 3D-digital focusing performed by ultrasound antenna array (C-SAFT), recording of transferring wave modes and design features of the unit under testing, including repair welding (Multi-SAFT), increasing of testing operating frequency, recording echo signal phases, taper surface monitoring, application of two-dimensional antenna matrixes.

Application of new testing technology aimed at increasing its sensitivity will inevitably result in increasing the number of registered imperfections. This factor is an additional argument requiring to apply quality evaluation norms for this welded joint with imperfection dimensional criteria.

Information analysis system analytical level subsystem of non-destructive testing of equipment and NPP pipeline metal

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Creation and implementation of the information analysis system for operating non-destructive testing of equipment and pipeline metal, developed according to the updated "Integrated activity program..." approved in 2012, was designated to solve the following problems:

- Information support of monitoring (performing operating non-destructive testing of equipment and NPP pipeline metal),
- Creation of unified information space by test results and analysis of defect creation and development mechanism, including causes for various nuclear power plants, units, process systems and structural parts (equipment and pipelines), and support of application software systems after the state estimation and residual life prognosis
- Information support of unit operation control procedures.

Presented target directions form a three-level structure of information analysis system for operating non-destructive testing of equipment and pipeline metal:

- *First level* – Basic (plant level) is oriented to solve tasks of the plant level and generation of information streams for the analytical and management level;
- *Second level* – Analytical level is oriented to acquire data from all nuclear power plants and development of recommendations and corrective measures to improve reliability of equipment and pipelines during operation;
- *Third level* – Management level is oriented to information support of NPP and CA management to taking management decisions on NPP unit operation.

Considering the set goals, IAS analytical subsystem should ensure implementation of the following processes:

- Acquisition and consolidation of data from all nuclear power plants, including accumulation, processing and loading of data from DB sources stored at NPP under the basic level IAS control, creation and keeping of the system joint directories;
- Processing of revealed NPP defects, including the classification of revealed imperfections and defects, generation of arrays based on the principle of control element uniformity and their detection, statistical analysis;
- Data exchange with external systems for predicting equipment and pipeline service life;
- Prediction of kinetics of equipment and pipeline defect development.

The report considers implementation of these processes for IAS analytic level used in computational infrastructure of JSC "Concern Rosenergoatom".

Ensuring operating erosion-corrosion resistance of NPP power unit welded joints

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Accrued NPP operating time and erosion-corrosion damages of equipment and pipeline base metal caused inadmissible thinning and fracturing of welded joint metal more frequently.

Beginning from 2012 JSC "Concern Rosenergoatom" has been implementing the actualized integrated program No. AES PRG-62K(04-03) 2012 on the problem related to erosion-corrosion, which provides for measures on improving NPP safety by preventing simultaneous damages and inadmissible thinning of welded joints of main pipelines related to the secondary circuit and feedwater control elements.

Accumulation and analysis of data on damageability and results of the metal in-service inspection performed within the framework of the above mentioned program allowed revealing facts of inadmissible thinning of welded joints and adjoining areas of NPP pipelines. Also there occurred some cases of worm-holes near welded joints caused by their erosion-corrosion thinning.

Analysis of damageability and investigation of physical and chemical features of the local erosion-corrosion thinning mechanism of welded joint elements allowed to detect their root causes. It was proved that the main influence on intensification of the local erosion-corrosion was caused by the chemical composition of the welded joint and adjacent areas, including generation of local near-wall hydrodynamical dithering in the area of welded joints.

Consequently, principles of design experimental justification of increasing the efficiency of in-service inspection of welded joint metal related to main pipelines of condensate-feed and water-steam train of NPP units блоков АЭС were developed.

Range of works was performed on design analytical justification of intensity and areas of local wear related to main pipelines of pilot NPP units with VVER-1000, VVER-440 and BN-600. There were also detected areas of welded joints classified under the risk group of intensive erosion-corrosion thinning. Currently works are being performed on developing and implementing software system of personnel support on the issue of welded joint erosion-corrosion considering pilot NPP units.

Development and implementation of service life estimation procedure related to coolant header welding to PGV-1000 steam generator nozzle DNOM1200.

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The suggested procedure is intended to estimate the service life the coolant header welding to PGV-1000 steam generator shell that has slit-shaped defects in the area below the welded joint 111 in the area of transferring from the header “pocket” to Dnom1200 nozzle (fillet with the radius of 20 mm).

While estimating the service life of the specified unit it is necessary to consider crack extension applying both fatigue mechanism, and mechanism of slow deformation corrosion cracking (SDCC) – specific fracturing characteristic of 10ГN2MFA steel grade in conditions of the secondary circuit water chemistry in case there are sludge deposits with increased copper oxide content. The procedure is based on results of direct experiments with static and cyclic low-speed loading of 10ГN2MFA steel, data of which were used to determine conditions of revealing SDCC (range of temperature and deformation rate) with determined parameters of equation of crack extension rate.

Service life estimation is performed for the defect, sizes of which shall be defined by ultrasonic test. Calculation of the stress and strain state in the area of the unit under consideration shall be performed in the linear-elastic area of constructive material behavior. Apart from load factors, which shall be considered according to PNAE G-7-002-86 (pressure, thermal actions, seismic load etc.), the procedure also provides for registering local and general tensions from technological actions (welding and thermal treatment). It was defined that the greatest contribution to damageability in conditions of SDCC have startup (heating) and shutdown (cooldown) modes. Recommendations were provided on mode schematization, considering temperature change in the loading cycle.

Calculation of kinematics of defect extension in the area of coolant header welding to PGV-1000 steam generator shell was carried out using this procedure. It was proved that if the original crack depth is 5-20mm, then its extension to through crack in case of SDCC requires 4-6 cycles of heating and cooldown or 4-6 years respectively if this cycle lasts for one year. Defect depth shall be 400 mm to be extended to the through crack for one service life.

The described damage mechanism in case of SDCC and received design estimations shall be checked against analysis results of defect extension in the area of welded joint No.111.

Improving methods of calculating strength and remaining life of VVER-1000 internals in case of life cycle extension up to 60 years

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Beginning with 2002 TSNII KM "Prometey" in cooperation with other industrial companies рфы been performing a range of works on extending service life of VVER internals. The result of these works was creation of "Methods of VVER-1000 internals strength calculation in case of life cycle extension" 1.6.1.08.0018-2007/2011, which then was the only normative document that allowed estimating VVER internals strength and remaining life, considering damage and degradation mechanisms typical for the material that was exposed intensive neutron irradiation.

However, as a result of insufficient number of experimental data, the developed method did not cover the range of destructive doses that was necessary for justifying the beyond-design-basis life cycle of VVER-1000 internals up to 60 years and more. Besides, it did not consider some dominating mechanisms of irradiation-induced swelling effect on materials of internals and some other mechanisms of corrosion cracking of irradiated materials.

In order to solve the specified issues within “Material testing works on extending service life of VVER-1000 and internals up to 60 years and more” it was suggested to study the effect of neutron radiation on physical and mechanical properties and mechanisms of internals material destruction. The performed large-scale investigations allowed to issue a revised and updated “Method of VVER-1000 internals strength calculation in case of life cycle extension up to 60 years” RD EO 1.1.2.99.0944-2013, which differs from the previous one in the following:

- Experimentally confirmed linear connection of physical and mechanical properties for the material of internals were specified up to the injurious dose of 150 dpa.
- Recording of irradiation-induced swelling effect on durability, ductility and cracking resistance of internal materials, and resistance to crack initiation and extension was introduced according to the fatigue mechanism;
- Approach to estimation of material corrosion cracking was reviewed and experimentally justified;
- A procedure that allowed minimizing the estimation conservatism of irradiation-induced swelling and deformation for internal components of the certain power unit was introduced by results of measuring the geometry of its reflection shield;
- Recording of area formation with critical thinning that can occur in the internal material as a result of phase $\gamma \rightarrow \alpha$-transition;
- Recording of possible crack extension in internal components caused by irradiation creep.

The report will include general provisions of the developed RD EO 1.1.2.99.0944-2013 “Method of VVER-1000 internals strength calculation in case of life cycle extension up to 60 years”.

Technology of recovery annealing VVER-1000 shells

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Next after annealing the accelerated irradiation of welded joint metal and base metal coupons was performed to confirm efficiency of annealing to restore the structure and и properties of VVER-1000 shell steels. Investigation of structural condition and mechanical properties proved extendibility of service life of VVER-1000 shell steels that underwent annealing for one more service life period.

Annealer to implement recovery annealing of welded joints 3 and 4 of VVER-1000 shell was designed along with work design documents for heater, thermal control and recording system. All taken technical decisions ensuring implementation of the intended annealing technology were calculated and experimentally justified. Auxiliary devices and mechanisms were designed and tested, heater was produced and underwent factory tests.

Considering the uniqueness and responsibility of annealing procedure it was decided to check and test the equipment using the full-scale model of VVER-1000 vault and shell.

Creation of the full-scale test stand with VVER-1000 model of vault and shell will greatly intensify works on scientific and technical support of reactor shell operation and implement the program of extending NPP service life working with VVER-1000.

Two-parameter criterion for heat exchanging tube plugging in VVER NPP steam generators

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One of the measures applied to reduce the number of plugging heat exchanging tubes (HET) and therefore to preserve heat-exchange surface of NPP steam generators (SG) is transition from one-parameter criterion of plugging HET considering defect depth (parameter is the phase of respective eddy current signal) to two-parameter criterion (phase and amplitude of signal). This will save a part of HET from plugging during periods of planned preventive maintenance (PPM).

Analysis of amplitude-phase criterion applicability during NPP SG HET plugging was performed in signals of operating eddy current control during PPM-2011 of Kola (VVER-440) and Balakovo (VVER-1000) NPP steam generators. The analysis was performed using the software-hardware complex PIRATE, which was capable to estimate main geometrical parameters of defects – their depth and axial length.

Geometrical parameters of maximum permissible defects were defined by data received by OAO OKB "GIDROPRESS" during testing (relative to curves of length and depth of permissible defects). For all possible combinations of defect geometry parameters (depth - 70%, 80%, 90%, axial length – 5mm, 10mm, 20mm) eddy current signals of through-type differential transducer were predicted using mathematical modeling by finite element method at respective working frequencies of every of the standard eddy current machines.

Then the curve of allowed length-depth ratio of critical defects and the crossing point of this curve with fixed depth coordinate lines (70%, 80%, 90%) was applied to the amplitude field to define required amplitude values. As a result, permissible signal amplitudes from SG HET defects while receiving and processing of operating control data, for example by ZETEC MIZ-70 installation (main operating frequency 100 kHz, normalizing 10 V) are displayed as follows:

Permissible depth, % of wall thickness	Permissible signal amplitude on the open area, V	Permissible signal amplitude from defect under lattice, V
<59%	Tube cannot be plugged	
60–68%	<3.63	<3.16
69–77%	<3.33	<2.90
78–87%	<2.21	<1.92
88–100%	Tube should be plugged irrespective of the amplitude	

Testing of signal indication determination algorithm, meeting two-parameter criteria, was performed using operating control data of SG HET PPM-2012 of Kolsk and Kalinin NPP. Total amount of tubes is 6040 (out of which 1397 are VVER-440 tubes and 4643 are VVER-1000 tubes). There were 967 indications detected (600 and 367 respectively). Tubes that should be plugged only by depth criterion - 260 pieces, and tubes that could be “saved” because of insignificant signal amplitude – 89, i.e. 34% of pipes may not be plugged.

Development of directions for use of safety concept «leak before break» applied to pipelines of active NPP

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A new document was developed for practical use at nuclear power plants working with VVER-440/1000 and RBMK-1000: RD EO 1.1.2.05.0939-2013 “Directions for use of safety concept ‘Leak before break’ applied to pipelines of active NPP”, where the safety concept “Leak before break” (LBB) was presented as a component part of general safety concept “break elimination” (BE) and was intended for justifying the impossibility of BCT simultaneous break of active NPP working with water coolant.

Prerequisites for applying LBB safety concept as a component part of general BE safety concept are based on performing main principles of: A) High quality; B) Under control operation and additional principles: C) Boundary loading conditions and D) LBB calculation and experimental evidence.

The document contains LBB methodology, technical recommendations and technical evaluation details for every LBB calculation procedure, requirements to leak detection systems, experimental validation programs and format of technical documentation, and the list of pipeline systems of active nuclear power plants working with VVER and RBMK reactors, which are potentially suitable for implementation of LBB concept.

Changing of VVER-1000 shell material peroperty during the operation of up to 60 years

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Neutron irradiation and high temperatures cause the degradation of VVER-1000 shell material properties, which increases the probability of the reactor shell brittle fracture in emergency situations, when filling cold water. Possibility of reactor shell safe operation beyond the period of design service life requires justification of prediction dependences, adequately describing changing of reactor shell material properties under the influence of damaging factors.

From the point of view of monitoring the reactor shell metal state, test results of irradiated and temperature witness samples (WS) that are kept in conditions similar to those of the reactor shell are more representative. Simultaneous analysis of WS data allows detecting not only changes of reactor shell metal properties, but also sharing the influence of irradiation embrittlement mechanism and temperature aging. Nevertheless, prediction of changing reactor shell properties under irradiation during long periods of operation, which do not have WS examination results, also requires using of forward accelerated irradiation of representative materials. Meanwhile it is necessary to consider actual structural changes occurring in metal under the influence of damaging factors.

This paper contains the prognosis of VVER-1000 shell material embrittlement during the operation beyond the design service life period on the basis of WS test database analysis, considering investigation results of changing the fine structure of reactor shell materials under irradiation.

As opposed to previously developed dependences, prediction estimate of general shift of critical brittle temperature ΔT_K (F, t) for VVER-1000 shell metal under irradiation is performed on the basis of physically justified presentation in the form of the sum of hardening or nonhardening component:

$$\Delta T_K(F, t) = \Delta T_K^{\text{Упр}} + \Delta T_K^{\text{Неупр}}. \qquad (1)$$

Hardening component in (1) is defined by radiation defects (dislocation loop formation and thickening) and nickel-enriched precipitates, thickness of which rises with the increase of fast neutron fluence. Nonhardening component in (1) is related to the formation of thermally and radiatively accelerated phosphorous segregation along grain boundaries зерен and interface boundaries of carbide matrix.

Correlation of WS metal forward radiation results with the database in WS testing of unirradiated samples allowed to justify the possibility to use the received dependences for the base metal and welded joint metal up to values of fast neutron fluence ~ 8×10^{23} neutron/m^2.

Using of accelerated irradiation results allowed estimating the conservatism of prediction for newly received dependences, additionally considering effect of thermal aging and influence of neutron flux density on irradiation metal embrittlement.

Development and certification of ECT system of heat exchanging tubes of NPP steam generators working with VVER using matrix sensors

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Currently the main method of flaw detection in metal of heat exchanging tubes (HET) of steam generators (SG) at NPP working with VVER is eddy current testing (ECT). This method allows receiving ф significant amount of information on HET condition, and its results form the basis of making a decision of HET plugging.

SG is one the facilities that ensures safe operation of the nuclear power plant and therefore recently matrix probes came into use worldwide to increase accuracy of HET testing along with axial probes (probes of the bobbin type), the former having a number of advantages before axial ones, such as:

- possibility of dividing defects into longitudinal and lateral, pits and cracks;
- possibility of detecting various defects in one section with HET; in the area of HET rolling, including the transition zone;
- visualization of presenting ECT results in 2D and 3D surfaces, which minimizes the risk of analyzer's error.

Considering the importance of improving accuracy of the performed SG HET ECT and high market prices for foreign-made equipment for implementation of this task, management of JSC Concern Rosenergoatom has taken a decision to develop and certify home-made SG HET ECT system using matrix probes.

The developed system was named "Politest-PG". The system included all elements necessary for an integrated approach for carrying out of SG HET

ECT – multifrequency, multichannel eddy current flaw detector; manipulator; probes, including matrix probes; data accumulation and analysis software.

Functional capabilities of the flaw detector allow applying it with all types of probes, including rotating and matrix probes. The flaw detector is included in the instrumentation registry.

Probe reaching the specified HET and moving inside HET is carried out by manipulator in the manual and automatic mode of remote operation. Manipulator weight and dimension characteristics provide for its mounting and dismounting at SG header neck without using NPP lifting machines.

All manipulator functions and flaw detector parameters are monitored via ETHERNET with the communication line up to 200 m.

The probe movement speed in the process of data acquisition: axial – up to 500 mm/s, matrix – up to 200 mm/s.

Visual observation of manipulator operation is carried out using two video cameras.

Software of data accumulation and analysis is operated by Windows operating system via user-friendly interface that allows configuring windows at operator's discretion. All information related to data accumulation (manipulator and flaw detector control, ECT signals, video picture) is displayed on one monitor screen.

A special program was developed for data analysis providing for tuning out disturbing factors and their combinations, such as spacer grid, bending, rolling.

According to results of acceptance tests, methods of carrying out SG HET ECT using "Politest-PG" system were approved by the federal agency "Rostechnadzor".

Applying of calculation and analytical approach of OKB «GIDROPRESS» to justify extension of time between tests of equipment and pipeline metal of the RP primary circuit working with VVER

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Currently, base metal test (MT) period was defined as once every four years according to requirements of the normative document /1/ during the operation in operating typical metal state test programs of RP equipment and pipelines working VVER. At the same time, the actual task is justifying the transition to more extended metal test periods (from 4 to 6 years). The main difficulty of performing this transition is the absence of corresponding methodic base.

To justify feasibility of time extension from 4 to 6 and more years between metal tests of VVER-1000 RP equipment and pipelines, OKB “GIDROPRESS” has developed an approach for equipment and pipelines of groups A and B according to the classification /1/ applying exclusively to periodical metal tests (in terms of /1/).

In general, the approach to justification of feasibility to extend time periods between metal tests consists of the following stages:

- stage 1: analysis of production and operating experience of RP equipment and pipeline elements to determine those that were degraded during the operation;
- stage 2: performing calculation for more critical RP equipment and pipeline elements (reactor shell, MCP welded joints etc.), which do not have operating degradation (cyclic growth of defects described by Paris equation is taken as a design degradation mechanism):
  - — dimension calculation of critical, potentially dangerous, potential defects;
  - — probable fracture analysis aimed at defining changes of fracture probability in case of changing metal test period;
- stage 3: conclusion on the possibility to increase metal test period.

Main results of work:

1) there is a number of RP VVER equipment and pipeline elements, where an unpredictable operating degradation is observed – for these elements the metal test period is set within the specially developed programs, considering the specific character of the situation;

2) design methods were used to define dimensions of potential crack defect (length and depth) for a number of the most critical elements of equipment and pipelines. It was proved that changing of metal test period from 4 to 6 years does not sensibly influence on dimensions of potential defect, and probability of destruction with the 6th year period of metal test does nor virtually (1-2 %) exceed the probability of destruction of the 4th year period;

3) the approach for justifying extension of metal test periods developed by OKB “GIDROPRESS”, and based on the operating experience and calculation methods, allows making recommendations on extending metal tests period for RP VVER equipment and pipelines of the primary circuit.

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1.2.4. RW AND SNF MANAGEMENT. NPP POWER UNITS DECOMMISSIONING

1.2.4.1. Topical session on « RW management; NPP power units decommissioning»

Preparation to decommissioning of Leningrad NPP units

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Four 1000 MW units with channel type (RBMK) reactors are operated at Leningrad NPP. The first unit was commissioned in 1973, the fourth unit – in 1981. Design service life of these units is 30 years.

Currently all the four units of Leningrad NPP were modernized and their service life extended.

Unit	Startup date, year	Operation end date, year	Date of extended service life, year
1	1973	2003	2018
2	1975	2005	2020
3	1979	2009	2024
4	1981	2011	2026

According to the "Object concept of decommissioning Leningrad NPP", variant of decommissioning "Immediate dismantling with postponed disassembly of the reactor" was considered as the basic one.

Basic reasons of choosing this variant are as follows:

- Absence of approved and justified method of handling with spent reactor graphite;
- Reduction of the amount of accumulated radwastes and radiation doses of the personnel.

The program of decommissioning should be developed not later than 5 years prior to the expiry of NPP design service life. Decommissioning programs have been developed for all the four units of Leningrad NPP.

Complex of the spent fuel container storage was put into pilot operation.

Works on commissioning of solid radwaste treatment complex are completed.

Currently, works on creating the complex of liquid radwastes (LRW) are performed.

Main high-priority tasks on preparation to decommissioning of Leningrad NPP are as follows:

- Integrated examination of epy current state of the first unit;
- Preparation of the technical assignment for development of decommissioning project;
- Preparation of documentation for getting a license for operation of Unit 1 that has been shut down for decommissioning;
- Development and implementation of activities related to handling with the nuclear fuel, including reusing on active power units;
- Justification and maintaining of resource characteristics of systems, equipment units, buildings and constructions left in operation at stages of preparation and commissioning of power units;
- Further creation of database on decommissioning of power units.

Below are possible perspectives of using Leningrad NPP site – Creation of experimental and demonstration center (EDC) on decommissioning of RBMK power units:

- Development, testing, implementation and improvement of new technologies;
- Spreading the experience of Leningrad NPP to other nuclear power plants
- Training for the personnel on topics related to new technologies and operating methods, including operation on robotic equipment.

Approaches to the solution of radwastes certification problem at Russian NPPs

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Radiological danger of radwastes and, consequently, the methods of handling and disposal are defined by radionuclide composition and specific activities of separate radionuclides. Specific activities of radionuclides and category of radwastes by radiation factor shall be stated during their certification according to Russian and international practice in force (international term – characterization) – id est, statement of package or radwastes batch characteristics and their recording into passport.

In Russia like in majority of countries the responsibility for measurements and documenting of wastes characteristics is laid upon manufacturing plant, particularly on nuclear power plant (NPP). In accordance with Federal law № 190-ФЗ adopted on 11.07.2011 «Radwastes handling...» (hereinafter - FL) and Decree of RF government dated 19.10.2012 № 1069 «Criteria of solid, liquid and gaseous wastes to be referred to radioactive wastes ...», officially the characterization of radwastes shall include control of specific activities of 275 radionuclides. Bigger part of these radionuclides refer to so called «complicated for detection» radionuclides, direct measurement of which activity is related to cost consuming procedures of sampling, preparation and analysis of the sample (^{3}H, ^{14}C, ^{55}Fe, ^{59}Ni, ^{63}Ni, ^{90}Sr, ^{99}Tc, ^{129}I, row of trans-uranium radionuclides and others). Currently there are no methods or measuring instruments at the NPPs of concern Rosenergoatom required for performance of such comprehensive control of radionuclide composition and specific activities of radionuclides in radwastes. Procedure of radwastes certification at NPP is not unified, specific activities of only those radionuclides are recorded in the passports that are considered by this or that NPP to be worth of recording or can be recorded. Mostly these are easily detected gamma irradiating radionuclides with high percent of decay output and energy of gamma- quanta (^{60}Co, ^{137}Cs, ^{134}Cs). The situation evolved, according to FL statements, will with a high level of probability result in refusal of the national operator for radwastes handling from accepting radwastes from JSC «Concern Rosenergoatom» for burial.

To fulfill the requirements of regulatory and legislative documents and prevent refusal of national operator from accepting radwsates from NPP for burial it is advisable to justify and comprehensively agree upon the necessary and sufficient list of radionuclides under control in radwastesw of JSC «Concern Rosenergoatom». Availability of such list will enable to bring the process of radwastes characterization to measurement of specific activities of necessary and sufficient number of radionuclides. For comparison, the list of radiologically significant radionuclides recommended by IAEA for characterization of NPP radwastes (IAEA Nuclear Energy Series NW-T-1.18) contains 30 radionuclides.

Increase of efficiency and effectiveness of control of radionuclides complicated for detection from the content of the stated list can be achieved by way of stating the stable or conservative relations between specific activities of radionuclides that are commonly called radionuclide vector for various types of radwastes. Stating of radionuclide vector for every type of radwaste in accordance with international standard ISO 21238-2007 will enable to bring control of radionuclides in radwastes to measurement of specific activities for only separate easily detected radionuclides.

Variants of ion exchanging resins handling at NPP

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Free volumes for storage of spent ion exchanging resins (IER) have been practically exhausted at almost all Russian NPPs. Thus, at every NPP with RBMK the volumes of IER are measured by thousands of m^{3}. Set up of methodologies and installations in Russia is more than evident. Methods of IER treatment that are used abroad can be divided into two groups:

- Destructive;
- Non-destructive (including to matrix).

Below some of them are discussed.

1. Thermal

1.1. Pyrolysis

Pyrolysis is a low temperature thermal process that is carried out within the temperature range 500–700 °C. In the course of pyrolysis the resins are destructed in the inertial atmosphere, polymer chains of resin materials are interrupted, thus forming solid remainder (<1/3 of resin weight) and gas steam fraction (>2/3 of resin weight). Advantages of pyrolysis consist in considerable reduction of waste volumes and almost complete absence of radionuclides carrying away. Experimental and industrial treatment of IER with the help of pyrolysis is carried out in Japan, Canada, Germany and Sweden.

1.2. Drying

IERs are dried under atmospheric pressure and rarefaction in many countries. Water steam (thick) and air in various apparatus are used for drying, including apparatus with «boiling layer» and rotor-film evaporators. IER are dried till the dry state and placed to containers. All functions of protective barrier are imposed on container.

2. Including to matrix materials

Before including into matrix materials IER are usually subject to one or several preliminary treatments. But sometimes just delivery water is removed.

2.1. Cementing

IER swelling can be referred to disadvantages of matrix, which can cause matrix damage, low filling of IER, volume increase when engagement and high speed of radionuclides desalination.

2.2. Including into polymer matrices

When IER is included into polymer binding substance the thermoplastic materials are used (polystyrene plastics, polyethylene) and thermal reactive materials (epoxy, poly resins). Conditioning is carried out in apparatus of periodical and continuous action in Argentina, France, Germany, Japan and other countries.

3. Decontamination (washing)

Washing is similar to IER regeneration, except for the fact that the purpose of the process is removal or elution of activity to water phase and transfer of this activity to inorganic sorbent, which can be easily immobilized.

4. Wet oxidation

Wet oxidation consists in IER interaction with oxide at presence of catalytic agent under temperature 100 ^{0}C and atmospheric pressure. As oxide 50 % of hydrogen peroxide solution is used. This method is widely studied in England, USA and Japan. Wet oxidation can be carried out under higher temperatures (200–550 °C) and pressure using hydrogen peroxide or air as oxide.

Comparative economic assessment of the above methods was performed and directions of activity for IER treatment accumulated at NPPs are proposed.

Techical and economic assessment of LRW handling variants at Beloyarsk NPP

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Treatment systems of Russian NPP radioactive media and wastes were constructed mainly according to design decisions, developed in 1960-70. According to requirements and processing capabilities existing at that time, a scheme of radioactive waste treatment was adopted, which allowed receiving concentrates – vat residues (VR) and ion-exchange resins (IER), intended for storage in the liquid state or pulp in the liquid waste storage (LWS). Currently LWS-1,2 of Beloyarsk NPP has accumulated 3800 m^3 of VR and 360 m^3 of IER. Main radionuclides are $^{134,\ 137}$Cs (with activity exceeding 80%), ^{60}Co, ^{54}Mn. The average salt content of VR is 325 g/l, VR activity - $3{,}5\times10^{14}$ Bq, IER activity - $7{,}2\times10^{13}$ Bq. Annual LRW supply in LWS-1,2 makes ~ 100 m^3 with activity ~ 10^{13} Bq. It is necessary to construct LRW treatment and conditioning complex (LRWTCC) in order to empty containers and bring LRW to the condition suitable for burying.

Technical and economic analysis of LRW handling technologies was performed in 2013 to choose the optimal LRW conditioning technology at Beloyarsk NPP. The following variants of VR and IER treatment and conditioning:

I. VR concentration up to salinity ~ 800 g/l. Conditioning of CVR, sludge from concentration of VR and IER by cementing.

II. Ion-selective cleaning of VR that results in salt fusion, spent sorbent and sludge. Sludge from ion-selective cleaning of VR and IER is conditioned by cementing.

III. Ion-selective cleaning of VR that results in salt fusion, spent sorbent and sludge. IER Pyrolysis .

IV. VR concentration up to salinity ~ 800 g/l. IER pyrolysis that results in ash formation. Conditioning of CVR, sludge from concentration of VR and ash in barrels by cementing. It produces a cement compound with VR, sludge and ash mixed in it. Stack gas filters are pressed.

V. Preliminary treatment of VR with the sorbent. Conditioning of CVR, sludge from concentration of VR and IER by cementing.

Economic estimation was performed in prices effective in the 2nd quarter of 2013. Below are results of economic assessment results:

Cost	Variant				
	1	2	3	4	5
Cost per unit, thous.rub./m^3	1 499	685	471	1 283	637
Total, mln. rub	7 015, 8	3 206, 8	2 202, 5	6 003,4	2 978,3

According to results of the performed technical and economic assessment, the most economic is the variant No.3. This variant was industrially tested at Kolsk NPP considering ion-exchange cleaning. Implementation of this variant requires adaptation of this technology for LRW of Beloyarsk NPP; R&D development; mastering the technology of IER pyrolysis at the level of industrial plants. Besides, it is necessary to create an industrial waste range on the territory of Beloyarsk NPP site to store/bury the salt fusion.

Creation of information system of governmental accounting and control over radioactive materials and wastes in SC "Rosatom" and integrated information system of radwaste handling support in JSC "Concern Rosenergoatom"

- In order to implement the information system of governmental accounting and control over radioactive materials and wastes:
  - The project implemented by State Corporation "Rosatom" is aimed at implementation of the information system of governmental accounting and control over radioactive materials and wastes (IS GA&C RM&W) on the basis of modern information technologies. The project shall be implemented together with ZAO "NEOLANT" (general contractor) and FSUE "RosRAO" (subcontractor).
  - IS GA&C RM&W is created in conditions of changing the normative base (making changes in the Order No.600 and setting of requirements to the system), which puts some claims to the possibility of IS GA&C RM&W adjustment.
  - IS GA&C RM&W is created according to requirements imposed by Russian regulators and those of the State Corporation "Rosatom" on information safety.
  - IS GA&C RM&W is the information basis of EGS RAO. Data coming to IS GA&C RM&W will be used to draw up analytical reports required for functioning of IS GA&C RM&W and ЕГС РАО.
  - Within the project implementation a central part of IS GA&C RM&W shall be created to process data by State Corporation "Rosatom" organizations and Russian organizations using remote communication channels with the possibility to use digital signature to sign these IS. Concerned parties can also view data via remote communication channels in compliance with access rights to data.

- Organizations that cannot use the central part of IS GA&C RM&W have a free access to the autonomous part of IS GA&C RM&W that is integrated with the central part of IS GA&C RM&W for data processing.
- During the verification of reports and other participants, a unified information field is formed for cooperation between RIYATS/VIYATS/TSIYATS to grant access to data and reports according to established access rights with in the framework of creating IS GA&C RM&W. Data go through the unified access point, which allows processing actual reporting data accessible to all participants of data processing.
- IS GA&C RM&W is integrated with the information system of SC "Rosatom" to be used in industrial and all-Russian references and classifiers, and using user authentication mechanisms implemented in IS landscape of SC "Rosatom".

• Information system of radwaste handling support is created in JSC "Concern Rosenergoatom" that allows the following:
  - Identification/labeling of accounting units (AU) for reading the information using technical means and entering AU number.
  - Description of the infrastructure of radwaste handling, locations of performing process operations by the system configuration.
  - Integration with spectrometric equipment regarding radwaste properties for getting reliable information on radwaste physical parameters.
  - One-time data logging on AU and repeated use of information while replacing AU to areas of process operations. Getting the information on locating AU in the storage.
  - Registration of operations related to AU, generation of AU creation history (what certain radioactive wastes of what accounting units were assigned to this AU).
  - Drawing up of supporting documentation, radwaste handling logbooks, papers on the basis the system data.
  - Possibility to use the system with channels, exchange of information using data exchange files.
  - Accounting of costs related to radwaste handling by entering data and integration with technical means of data accumulation.
  - Usage of mobile devices to receive information on AU in situ, registration of information on AU (during the formation /drawing up of inventory).
  - Drawing up of industrial reporting documents.
  - Providing actual information on radioactive wastes to the central management of JSC "Concern Rosenergoatom" for scheduling the activity on radwaste handling, drawing up of analytical reports on required topics.
  - Data generation for submission to IS GA&C RM&W.

Complex of radioactive waste plasma treatment of Novovoronezh NPP. Installation and putting into operation

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During several recent years a project of the complex of radioactive waste plasma treatment of Novovoronezh NPP was developed, and in 2013 installation and construction works began on installation of the complex at NVNPP 1, 2 units. The technological complex was based on using plasma heating sources and shaft loading of solid radioactive wastes. Technology of waste treatment was tested and beginning from 2007 it has been successfully used for radioactive waste treatment by experimental industrial installation "Pluton" of FSUE "Radon", which serves as a prototype during the development of the complex design.

Design capacity of the complex makes 250 kg per hour of solid radioactive wastes with the wide morphological composition. Together with combustible materials (paper, cloth, wood) van be processed – construction waste (not more than более 35%), heat insulation materials (not more than 20%), ion-exchange resins (not more than 5%), plasticate, rubber and polymer materials (not more than 3%), metallic scarp (not more than 3%). The complex consists of more than 23 basic and supporting process systems, including control and management system, power supply and radiation control systems.

Operation mode: waste treatment by life cycles for the period of 360 to 720 hours and subsequent carrying out planned preventive maintenance during 360 hours. The resulting product of radioactive waste treatment (slug) possesses properties that allow safe and reliable storing and burying it for a long period of time. The resulting product will be placed in disposable unitary containers, which in their turn shall be placed in the last-trip shielding container of NZK-150-1,5P type. Certified LTSC can be places for temporary storage or radioactive storage point.

Customer and operating organization – Branch of JSC "Concern Rosenergoatom" "Experimental demonstration engineering center of decommissioning". Installation and construction works have been performed since December of 2013. Work schedule provided for the following stages: equipment installation completion in August of 2014, completion of pre-operation tests – October 2014 года, getting a license for operation and putting into experimental operation – November of 2014 года. Currently construction works, installation of main complex equipment have been completed. Works that are being performed: shaft furnace lining, pyrolysis gas combustion chamber and evaporator heat exchanger, piping works, installation of supporting system equipment.

Mathematical model of SRW treatment in the shaft furnace has been developed using plasma sources of heating. Model application allows defining optimal modes of carrying out a technological process of high-temperature waste treatment, and define parameters of waste treatment processes with different morphology in order extend the treatability of such wastes on the created complex. The important component of the developed model – calculation module of cesium isotope distribution that allows optimizing processes by minimizing the entrainment of radioactive isotopes with effluent gases.

SGRW and LGRW burners at NPP and NFC organizations

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Issues and problems of creating burning complexes of SGRW and LGRW are considered on the basis of existing experience of engineering, constructing and producing burning equipment for FSUE "GKHK" and modernization of burner at Kolsk NPP.

Burners are designated for burning low-activity SGRW and LGRW that are accumulated in companies to reduce the volumes of wastes and their transition in the form safe for storage, transportation and burial.

Design of SRW burner equipment modernization of Kolsk NPP provides for replacement of SRW burner, SRW loading facilities, ash removal and conditioning, except for the gas purification system. Capacity of burned SRW makes 35 kg/h. Design coefficient of reducing SRW volume is 100. Specific activity of initial SRW is max. $3{,}7\times10^5$ Bq/kg.

According to the taken technical decisions the following parts are included in the modernized burner equipment of Kolsk NPP: semiautomatic unit of SRW water-cooled gates and slide valves, electrically driven roll tables; burner with rotary upper and lower fire bars; afterburner; ash removal unit with dosing screw and knife gate valve; unit of cement mortar preparation; cementation unit; equipment for preparation and air supply to the burner and afterburner, and also for cooling of equipment.

Peculiar feature of the burner is using the technology of cementing the ash in 200-liter barrel with high permeability cement mortar using special vibration devices.

Burner for FSUE :GKHK" was redesigned.

Unlike the burner of Kolsk NPP the design also provides for burning LGRW. Design capacity of the burner is 60 kg/h for SGRW and 20 kg/h for LGRW, meanwhile LGRW generally consists of burnt oils from equipment operation. Specific activity of initial SRW is max. 10^5 Bq/kg for α-nuclides and max. 10^6 Bq/kg for β-nuclides.

The most difficult and critical system considering designing was gas purification system that purifies stack gases from the solid phase, radioactive nuclides and harmful chemical materials down to sanitary standards.

Cementing of ash residual and spent radioactive salt solution, forming in the system of gas purification, is performed in the bulk mixer. Unloading of the cement compound is performed in 200-liter container.

Burning of SRW is carried out if there is excessive air. To maintain the required temperature mode in the burner and afterburner of stack gases, both burners use nozzles working on diesel fuel.

All design and engineering works were performed using the experience of operation and servicing of similar equipment in FSUE "Radon", and in active SGRW burner in other projects. All technical decisions were approved by calculations and results of mathematical modeling of current processes.

According to this project, burner equipment was supplied and installed at the site of FSUE "GKHK".

Radiation survey experience of shut down commercial uranium-graphite reactors

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Currently all commercial uranium-graphite reactors (OAO "SKHK", FSUE "GKHK" and PO "Mayak") and power reactors of AMB type (Beloyarsk NPP) have been shut down in Russia. Works on preparation to commissioning and decommissioning are performed on such reactors. After exhausting the service life it is suggested to perform the final shutdown of RBMK reactors (Leningrad NPP, Kursk NPP, Smolensk NPP) and EGP-6 (Bilibinsk NPP). Preliminarily the final shutdown of the first RBMK reactor of Leningrad NPP is scheduled on 2018.

After the final shutting down of reactors during their preparation to decommissioning it is necessary to estimate radiation condition of reactor constructions and prove their nuclear safety. Results of inspections are used as source data for developing a package of documents required for preparation of decommissioning project and getting a license for decommissioning.

This report describes the experience of OAO "ODTS UGR" specialists related to performing radiation survey of uranium-graphite reactors: OA "SKHK", FSUE "GKHK" and PO "Mayak". OAO "ODTS UGR" was founded on the basis of the reactor plant of OAO "Sibirsky Chimichesky Kombinat", Seversk, according to the decision of the state corporation "Rosatom".

Specialists of OAO "ODTS UGR" together with leading institutes of FSU "GNTS RF-FEI" n.a. A.I. Leipunovsky (Obninsk), IFKHE RAN n.a. A.N. Frumkin and NIYAU MIFI (Moscow) developed methods and equipment, designated for defining the nuclear and radiation state of graphite stacks of uranium-graphite reactors.

The report considers results and methods of carrying out experimental investigations on scanning the graphite stack cells, selection and radiochemical analysis of graphite samples to define activity of long-lived fission products and areas of their distribution in graphite stacks.

Tested methods and equipment of reactor radiation survey are supposed to be used during RBMK and EGP reactor surveys.

Objective limitations of advanced treatment of radioactive concentrates technology and ways to solve the problem

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Requirement of minimizing the amount of radioactive wastes is determined by reasons of radiation safety and economy. Significance of the latter factor grew immeasurably high after adopting the law No.190-FZ and corresponding subordinate acts [1] in 2011. In this regard tasks on improving efficiency of LRW treatment become especially actual, and accordingly, tasks on reducing the amount of secondary wastes to be buried.

For this purpose this paper contains a "sensitive" composition analysis of radioactive concentrates generated at NPP. Results of analytical studies and experimental verifications proved that:

1) Contrary to the common perception, presence of ion oxalates and citrates in EDTA vat residues in coordination active form is unlikely [2] and thus a relatively low efficiency of LRW ion-exchange polishing cannot be explained as related to binding of activated corrosion ions and some polycharged fission products in sorption-inert complex compounds by organic acid anions;

2) Practically all LRW with salt content exceeding 1 g/dm^3 contain nonorganic acid anions (SO_4^{2-}, PO_4^{2-}, CO_3^{2-} etc.), which in case there are actual proportions of concentrations "central atom-ligand" can bind the prevailing part of transition metals into neutrally charged (inert in ion-exchange processes) sulfate, phosphate and, probably borate complexes;

Thus, in order to increase the level of radioactive concentrate treatment it is necessary to remove complexing agents from solutions or inactivation of nonorganic complexing agents.

The report describes experimental and theoretical validity proof of the hypothesis about presence of radionuclides in NPP radioactive concentrates and the possibility of improving the technology of LRW treatment within the proposed concept.

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Experience of using SRW methods and equipment of ultrasonic decontaminaiton during NPP decontaminaiton

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OOO “Aleksandra-Plyus” has accumulated a considerable experience in development and usage of ultrasonic technologies in the nuclear industry. For the period of long-term cooperation of OOO “Aleksandra-Plyus” with OAO “NIKIET”, OAO ‘Concern Rosenergoatom” and a number of other nuclear companies, they managed to salve many tasks owing to the development and implementation of unique equipment using ultrasound technologies. The report provides analytical review of successful examples of various ultrasound technologies in domestic nuclear industry, including decontamination of metal radioactive wastes and grounds.

Usage of ultrasound technologies ensured efficiency of decontaminating metal parts, including claddings of spent fuel assemblies (SFA). Main methods of cleaning large parts include usage of large ultrasonic baths, submerged ultrasound modules and usage of contact methods of treatment. As a result of ultrasound decontamination the treated metal becomes suitable for repeated usage and remelting. Implementation of ultrasonic treatment in the decontamination process allows decreasing the technology process ten-fold and reduce the amount of generated LRW.

Developments of OOO “Aleksandra-Plyus” are patented for inventions, authors of which number more than 50 leading specialists both in the sphere of ultrasound technologies and other spheres where they are being implemented. Specialists of the company have developed more than 100 ultrasonic transmitters, on the basis of which there were more than 400 models of equipment created. Also there is a great amount of supplies abroad. Differential characteristic of the company developments is the wide application of resonance effects in the ultrasonic oscillations, which can improve the efficiency of the created equipment with the significant decreasing of power consumption.

OOO “Aleksandra-Plyus” holds an active cooperation with leading Russian scientific centers, supplying them with laboratory and testing ultrasonic equipment.

Leading industrial Russian companies and companies of neighboring countries are among customers of the company. Major partners of OOO “Aleksandra-Plyus” are FSUE PO “Mayak“, JSC “Concern Rosenergoatom” and its branches, including Novovoronezh, Kalinin NPPs, and nuclear fuel cycles.

Automated system of accounting and control of radioactive materials and wastes of Leningrad NPP

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Unified system of automated accounting (USAA) and control of radioactive materials and wastes is designated for automation of internal production activity of LNPP subdivisions related to accounting, control and inventory of radioactive materials and wastes at all stages of handling with them. The unified automated system includes:

- Program subsystems:
  - Accounting and control of containers with SRW (SRW-ACC);
  - Accounting of LRW;
  - Accounting of radioactive isotopes, including those generated by irradiation in the reactor;
  - Radio inventory;
  - Three-dimensional visualization of SRW storage structure.
- Server and automated workplaces of users.

Unified automated system provides for the following:

- Accounting of SRW and LRW;
- Centralized data acquisition on handling with radioactive materials and wastes immediately from workplaces of job foremen into the common database;
- Continuous accounting of operations performed from radioactive materials and wastes at all stages of life cycle;
- Automated control and inventory of packed SRW on the basis of using bar-code and radio labels;
- Automated drawing up of certificates and reports on handling with radioactive materials and wastes for submission to regulatory and supervisory authorities.

Structure of implemented operations, references and reports is adequate to the process cycle of handling with radioactive materials and wastes, which ensures continuity of accounting, timely recording of operation results and detection of violations during the handling with accounting units.

Reports and certificates enable a prompt detection of actual presence of radioactive materials and wastes, receiving all actual information on the current previous condition of accounting units in different section, both in the form suitable for the current operation, and in the form of reporting documents to be submitted to control organizations.

Access of users to all system elements is controlled via the management system.

Subsystem of radioinventory allows keeping a continuous monitoring of status, presence and movement of containers with wastes without the immediate human part.

Wide functional capabilities, easy and intuitive interface, meeting the requirements of the normative base make the Unified automation system a reliable and comfortable instrument of performing the whole cycle of works related to accounting and control of radioactive wastes in branches of Concern Rosenergoatom.

Simulation model for cost estimation for decommissioning of PWR-TOI NPP unit

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Simulation model has been developed, considering cumulative costs for decommissioning of VVER-TOI NPP Unit on the variant of "Elimination with immediate dismantling" on the basis of defining and summing up discrete cost estimates related to performance of required works and process operations.

The structural scheme of the developed model specifies the number of objects, types of works and process operations, performed on these objects, considering layout and design features of the unit.

Functioning of the model is maintained according to adjustable calendar of events, featuring the list and consequence of work performance, including process operations on every object.

In order to perform cost estimation of every type of works or process operations, performed on simulation models, a special formula mechanism has been developed.

Initial data for the model are as follows:

- weight and dimension characteristics of equipment, systems and engineering structures of VVER-TOI Unit;

- radiation characteristics of equipment, systems and constructions of VVER-TOI Unit;
- specific engineering and economical performance of process operations, including handling with radioactive wastes and other required data.

The model allows estimating costs for work performance both on every object of the model and for the unit as a whole, and estimate the required number of containers with radioactive wastes of different activity classes submitted to the national operator.

This model can also be used for the comparative cost analysis of different layout, construction and process decisions of decommissioning, taken during engineering of VVER NPP units.

Creation of the system of radioactive wastes charcterisation and putting of radioactive wastes out of regulatory control in JSC «Concern Rosenergoatom»

Taking the Federal law No.190-FZ "Handling with radioactive wastes and making changes in certain legislative acts of the Russian Federation" allowed joining the previously accepted normative and regulatory framework and harmonize Russian legislation considering the radioactive wastes safe handling with the legal system of leading foreign countries, and with requirements of the international legal regulation on handling with radioactive wastes. One of the key provisions related to handling with radioactive wastes according to the federal law No. 190-FZ is creation of unified state system of handling with radioactive wastes (RW USS). The purpose of its creation was organization and ensuring safe and economically effective handling with radioactive wastes, including their burial.

Instrumental reviewing of radiation characteristics and composition content of RW accounting units is the source of data for creating the system of safe handling with radioactive wastes. Characteristics of radioactive wastes should be measured to define their chemical, physical, and radiation properties, which makes it possible to build up a classification to select the most safe and economically effective variant of handling with wastes.

The bigger part in the general amount of solid wastes make wastes containing radionuclides with the specific activity of up to 10^4 kBq/kg (by beta-irradiating radionuclides, excluding tritium). According to the existing classification rules such wastes are subdivided into the following groups: low-activity radioactive wastes (LARW), too low-activity radioactive wastes (TLARW); for NPP wastes – industrial wastes containing radionuclides (LARW), for other organizations and companies – limited use wastes; unlimited use wastes without additional requirements to radiation safety. Transition of LARW to the category of TLARW, LARW and industrial wastes define a significant reduction of the recyclable LRW amount leads to cost reduction for their handling and reduction of required storage amounts and consequently to the general cost reduction.

According to the prognosis of creating all radioactive waste categories at NPPs of OAO "Concern Rosenergoatom" as of 2014-2025, the amount of potential radioactive waste quantity for releasing makes ≈ 6,000 m^3. Experience of handling with LARW (Czech Republic) using the release system (1000 m^3/year), shows that implementation of this system allows reducing the amounts of LARW by 4 times, where one half of LARW and TLARW amount is transferred to the category of industrial wastes.

Calculation of annual costs for creation of the release area with the productive capacity of 250 m^3/year in one shift mode of operation allows estimating the release cost. The release cost makes about 84 and 40 thous. of rubles/ m^3 with one and three-shift modes of operation respectively. On the basis of the prognosis of accumulating potentially releasable wastes at NPP of OAO "Concern Rosenergoatom" and using the conservative approach to cost estimation for releasing and handling with LARW, TLARW and industrial wastes estimation of economic effect from implementation of the release system prove the economy up to 2,5 bln. rub/year.

In order to perform works on releasing from regulatory control of wastes generated at NPP of "Concern Rosenergoatom" it is necessary to create at least 10 release areas. These works imply significant investments in the amount of not less than 500 mln. of rubles, and organization a lot of organization works. Besides, considering prognoses of waste generation at certain NPPs, an incomplete loading of equipment is foreseen considering created passportization areas, which also defines degradation of economic parameters.

Effective mechanism of reducing investment costs on releasing radioactive wastes is outsourcing, i.e. transferring works on releasing NPP wastes from regulatory control to specialized organizations.

Investigation of influencing factors and accuracy estimation while measuring solid radioactive wastes containing plutonium using neutron coincidence counter

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Purpose of operation: Validation of methods for measuring solid radioactive wastes, containing plutonium by detecting neutron radiation. Investigating factors influencing on accuracy of measurements and estimation of overall uncertainty when measuring wastes containing plutonium by neutron coincidence counter.

The main task of the performed work was examination and selection of analysis methods suitable for measuring the mass of plutonium in SRW.

Neutron coincidence counter produced by Canberra Company, NMCS model was used as measuring equipment. INCC program was used as the data processing program. Samples of SRW were modeled using standard samples of plutonium dioxide placed in 100 l. barrels with the filling consisting of different materials for simulation of waste matrixes.

Table 1. CO properties .

CO No.	Mass of CO, gr	Δ, gr
19	4,940	0,007

Table 2. Source properties.

Source	Intensity	Certification year
^{252}Cf	2,5 10^6 n/s	2002

Table 3. measuring results.(m= 4.70 gr.)

Matrix	Methods			
	Additional source (m, gr)	Passive calibration curve (m, gr)	Known alpha (m, gr)	Multiplicity (m, gr)
Paper	3.55±0.18	1.84±0.10	1.89±0.10	1.95±0.14
Concrete	**4.92±0.25**	4.90±0.25	4.72±0.24	4.44±0.23
Wood	**2.46±0.14**	–	0.551±0.035	0.91±0.05
Polyethylene	**6.33±0.35**	2.30±0.12	2.37±0.13	2.63±0.15
Sand	**5.11±0.26**	5.09±0.26	4.83±0.25	4.52±0.23

The received results allow to make a conclusion that accuracy of measuring plutonium mass for measuring and analysis of basic approaches is worse. Method ‘additional source” is more universal and suitable for operation with waste matrixes of any composition.

Decontamination of high saline liquid radioactive wastes, containing organic complexons

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Nuclear power plants of Russia and some of the adjoining countries have accumulated chemically complex liquid radioactive wastes (LRW). Vat residues are most difficult for treatment being high saline solutions, containing anions of boric acid and organic complexons, fission products, radionuclides of corrosion origin and substances used to maintain water chemistry and decontamination of equipment. The main activity is defined by cesium and cobalt radionuclides. Problem of cesium radionuclide treatment is solved, meanwhile cobalt radionuclides in such solutions form fixed complex compounds, which will not allow to remove cobalt using traditional methods.

Methods of oxidative destruction of complexons are used for the treatment of such wastes: ozonation, pressure oxidation in the presence of hydrogen peroxide etc. However these methods are expensive.

New approaches to technology of LRW decontamination, containing cobalt, cesium radionuclides, and organic complexons in the form of oxalic acid, citric acid, ethylenediaminetetraacetic acid etc. without preliminary oxidation of organics have been developed to improve efficiency and make the LRW treatment cheaper.

Decontamination technology includes inclusion of decomplexing cobalt reagent in the initial radioactive solution to maintain the specified value of pH solution, inclusion of collector (precipitator) to remove cobalt and cesium radionuclides from the solution, mother liquor elimination containing organic complexing agent and residual quantity of cobalt and cesium radionuclides. Advanced treatment of mother liquor is performed to achieve the required residual content of radionuclides in treated liquid wastes.

This method provides for:

- Effective decontamination of LRW, containing organic complexons;
- Significant reduction of energy consumption;
- Usage of cheap accessible material (reagent) and standard chemical equipment produced by Russian industrial companies;
- Concurrently removal of cesium radionuclides allows reducing the flowrate of specially used sorbent by more than 98%;
- Simplicity of the process implementation (using standard chemical equipment);
- Excluding the necessity to buy expensive oxidizing equipment: ozone generators, oxygen preparation equipment, autoclave etc.);
- Absence of chemically aggressive gas generation that leads to equipment damaging;
- Carrying out of the process under atmospheric pressure and at temperature max. 80 °C.

Main decisions on cementing LRW in NZK-150-1,5P container with mixing mechanism

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One of the ways to improve efficiency of cementing liquid radioactive wastes (LRW) is reducing investment and operational costs by simplification of technology and solidification equipment, reducing the total structure volume of the building intended location of solidification installation and reducing the amount of secondary wastes during the operation of installations.

Module installation of LRW solidification in NZK-150-1,5P container with non-recoverable mixer [1] developed by OAO Head Institute "VNIPIET" and OAO "345 mechanichesky zavod" wholly serves this goal.

The man element of module installation is NZK-150-1,5P container with non-recoverable mixer, which can be used separately from the module by integration in existing or newly created constructions.

Biological shielding of NZK container allows solidifying average activity wastes, and the package of radwastes will surely meet criteria of acceptability for long-term storage and burial in radioactive waste burial points, которые which will be developed by the national operator.

An example of using LRW cementing in NZK-150-1,5P container with non-recoverable mixer can be the installation, a part of single housing LRW treatment system of radiochemical building No.6 of FSUE SRC "Radiyevy institut im. V.G. Khlopina", Gatchina.

The installation is going to be located in the reconstructed part of the building No. 6 in the transport entry-additional building at elev. +1,000 between axes. Peculiarities of locating the installation is confined area and height of the building, which determined the selection of equipment for cementing of LRW.

The process of cementing was performed remotely, 1st class o performed works, according to OSPORB-99/2009.

The empty container is delivered to premises for cementing on a tracked type carrier. The carrier moves the empty container under the position of plug removal/installation, then the plug is remotely by electric hoist using a special cross arm for plug, then the container is transported to the position of cementing.

LRW supply is carried out by the pump unit through LRW supply nozzle located on the process slab. Supplying the required amount of LRW is followed by supplying of cementing mixture by the packing plant with the worm feeder. The amount of cementing mixture equals to a single portion, which is required for one cycle of cementing. The process of cementing is performed with continuous mixing of the medium with the built-in mixer and air purification from the cement kiln dust on the line of breath blowdown.

Taken decisions on the method of cementing LRW of the radiochemical building FSUE SRC :Radiyevy institut im. V.G. Khlopin" allowed getting into the minimum required bulk of building meeting all norms of ensuring radiation safety with minimal costs for purchasing equipment and installation operation.

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New material for immobilization of VAT residues

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Currently, immobilization technology of LRW received during the distillation or vaporization is their concreting. Howeverб creation of compounds meeting requirements of GOST R 51883-2002 during cementing vat residues, concentration of radionuclides and salts in which their content in initial LRW is exceeded by 60-300 times, refers to the number of problem and not completely certified technologies. Significant differences of salt content, presence of various organic substances and surface active agents in the composition of LRW of nuclear power plants working with reactors of different types of "RosRAO" companies etc. also do not allow applying a universal technology for cementing such vat residues.

It is proposed to use matrix compositions on the basis of magnesia cement to solidify high saline LRW concentrates, containing organics and surface active agents to prepare them for burial. Selection of magnesia composition materials is based on exceptional properties of solidified compound, specifically:

- High mechanical stability, exceeding that of concretes by 1-2 times;
- High adhesion to mineral and organic substances that allows using virtually any substances for filling;
- High density as compared to concretes (2900-3200 kg/m^3), thus the thickness of the layer of half attenuation of gamma-radiation is lower by 14–18 %;
- High content of chemically bound water – more than 40 %.

In order to provide experimental validation of using matrix materials on the basis of magnesia cement magnesia-mineral-salt composition (MMSC) was used. Its main components are as follows:

- Caustic magnesia powder – magnesia cement;
- Magnesium chloride – solidifying agent;
- Ash, ground metallurgical slag, barite etc.

It should be noted that usage of ash as a filler will not significantly change physical and chemical properties of solidified compound. Since including up to 40 mass. % of ash ensures a standard (GOST R 51883-2002) durability of solidified compound.

Various sorbents for cesium, sequence of adding components into the compound, and correlation of main components during the preparation of compounds were examined to developed the technological process that allows preparing compounds meeting requirements of their quality according to GOST R 51883-2002. Quality of compounds considering their compliance with normative requirements was defined by their mechanical stability and leach rate for ^{137}Cs.

Vat residues were used as radioactive wastes from LRW distillation installation of active laundry with the salt content of 600 g/l, activity of which is equally determined by ^{137}Cs and ^{90}Sr with insignificant presence of ^{60}Co (1–2 %). Ferrocyanides were used as cesium sorbent, and calcium chloride was used to increase the degree of filling compounds with salts for binding phosphates, oxalates and silicates in sparingly soluble compounds.

By results of experiments a reliable solidification of vat residues was attained (leach rate of ^{137}Cs $2 \cdot 10^{-5}$ g/cm$^2\cdot$day), containing up to 30 % of organic substances, with the including of dry radioactive salts 35-37 %.

Technologies and equipment for conditioning LRW of BN-reactors with heavy coolants (HC)

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The report contains analysis of main types of liquid radioactive wastes (LRW) accumulated in the process of operating reactor plants with heavy coolants differing by performance specifications of accumulation, radiochemical composition, physical and chemical properties.

Handling with LRW of this type reactors is considered in the logics of the full process cycle from LRW accumulation to their conditioning by solidification in the stable form - sparingly soluble compound suitable for the long-term insulation from environment.

Treatment of nonspecific LRW types (drainage waters from decontamination of premises; active laundry and air locks; decontaminating solutions from treatment installations for radioactive media, cooling pool water etc.) can be implemented by standard methods, according to known technologies used for LRW of thermal reactors, however it will be reasonable to implement new economic technological processes.

During the operation of reactors working with HC there arises a necessity to clean and decontaminate equipment contacting with products coolant activation, corrosion products of equipment material and fission products of fuel. Primarily it concerns steam generators, mass-tarnsfer apparatuses, reloading mechanisms, RCPS, filter housings etc.

Handling with LRW waters from cleaning coolant residues and decontamination of the primary circuit equipment presupposes development of new technological methods, specifics of which is defined by increasing the content of Pb (Pb-Bi) salts in LRW composition – chemically toxic substances, regulated by new NP "Criteria of radioactive waste acceptability for burial".

Feasibility of the new economic sorbent-membrane method of LRW treatment and conditioning (cleaning and decontaminating solutions, active laundry waters) is experimentally confirmed on physical and chemical processes: reagent-filtration separation; catalytic thermal-oxidative recovering destruction of organic components, sorbtion removal of radionuclides; coagulation and sedimentation; micro-filtration compartment of suspended particles and immobilization of concentrates and sorbent in the slag-alkaline cement matrix.

The list of main radiochemical parameters of LRW composition has been accepted for implementation of the process, procedure of treating every type of LRW in compliance with the developed method. Optimal parameters of LRW treatment (cleaning and decontaminating solutions (on simulators), active laundry waters) were approved. Flowrate of initial reagents was defined along with the number of accumulated solid insoluble products ("slurries") from spent reagents. The amount of LRW reduction can be achieved by 50 times depending on the type of treated LRW.

Technological process was refined on mockup samples of installations ensuring the full cycle of handling with LRW.

A new process method was recommended concerning perspective source materials for carrying out a process, parameters and formulas for slag-alkaline cementing of slurries, spent sorbents.

Compound samples were synthesized with the filling of radioactive slugs and sorbents – up to 35 and 30 mass % respectively, main quality factor of which comply with NP-019-2000: mechanical stability for axial compression– 9-20 MPa; ^{137}Cs leach rate in the water– $\leqslant 10^{-4}$ g/cm^2.day.

The technological process was refined and optimized to create an industrial installation for LRW treatment of the reactor plant working with HC.

Elaboration of single stage vitrification technology by induction melting in the cold pot at low current frequency

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Current radwaste vitrification installations by induction melting in the cold pot and installations designed in Russia use high frequency current, causing the occurrence of some phenomena that significantly reduce the melting pot service life. Usage of low frequencies allows solving this problem cardinally. The report specifies results of works, performed in FSUE "NITI im. A.P. Aleksandrov" in cooperation with St. Petersburg State Electrotechnical University on elaboration of single stage vitrification technology by induction melting at the installation with low current frequency of inducer.

Samples of phosphate glass have been synthesized with inclusion of LRW during a number of preliminary laboratory experiments. Also radiocesium washout rate was examined in these samples. Glass has permissible rate of ^{137}Cs washout according to NP-019-2000 for vitrified high-activity LRW. Synthesized phosphate glass contains up to 30% of LRW natural concentrate solid residual.

Induction melting pot was tested successfully with the cold pot of large diameter, produced from stainless steel, with reduced inductor current frequency. Also main processing characteristics of phosphate glass melting in it were defined.

Designing IPS600 vitrification installation was performed on the basis RASPLAV platform considering the received experimental and estimation results. IPS600 installation engineered for elaboration of induction furnace design, LRW concentrate vitrification technology and its subsequent implementation pilot installation was created and commissioned in NITI in 2013. Specific feature of the installation is that the steel melter has large dimensions (up to 600 mm in diameter), high performance considering the solution and reduced current frequency that ensure a significant improvement of electrotechnological equipment reliability. A number of cold pot sections was added to reduce the probability of high-frequency disruptive discharges between sections. The installation consists of the following components: vacuum-tube oscillator, induction melting pot with the lid and drain bottom assembly, LRW concentrate preparation container equipped with the controlled heater, glass melt container. Vitrification installation is designed for immobilization of LRW concentrate at a single stage with periodical or continuous pouring of glass. Optimization of melter parameters during melting of phosphate glass was performed using electro-hydrodynamical 2D models of the induction melting pot in ANSYS package developed in the laboratory of SPbSETU “LETI”.

Estimations and the received experience show that when using a generator with higher power output the designed installation is capable of processing the known types of boiled off LRW with the capacity of glass output reaching 0,72 tons per day and calcinated radwaste with the capacity of glass output up to 2 tons per day.

1.2.4.2. Topical session on «SNF management»

Perspectives of producing in Russia shipping packaging sets with housings made of high-duty cast iron with globular graphite

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Currently the worldwide practice recognizes containers with housings made of high-duty cast iron with globular graphite to be the most advanced construction of shipping packaging sets (SPS) for transportation and storage of spent fuel (SF) of nuclear power plants and other facilities.

OAO “Petrozavodskmash” in partnership with specialists of OAO SRC “TSNIITMASH” and OAO “KBSM” produced a pilot SPS on the basis of the cast-iron container for transportation and storage of spent fuel assemblies (SFA) for nuclear power plants working with VVER-1000. In this regard they approved SPS construction, housing of which is performed by monolithic cylindrical casting from high-duty cast iron on the basis of analyzing various structural schemes of containers described in various sources of information and produced by OA “KBSM”, and the neutron shield is placed around the housing in the form of cylindrical jacket that is externally covered by stainless steel.

Designing of new SPS-146 construction allows the following:

- Creating an SPS park that satisfies up-to-date safety requirements;
- Mastering the production of new types of equipment and innovation technologies, including those related to casting and modifying of large masses of high-duty cast iron with globular graphite and production of bulk high-duty castings out of this advanced material.

Also a new effective technology of thick-wall casting the container body out of high-duty cast iron with globular graphite was developed using the mechanical property data of cast irons with globular graphite in the body of industrial bulk castings produced by different manufacturers. Meanwhile a special casting mold was used that ensures fast cooling of hot metal during the mold curing period to form graphite of regular globular shape and slow cooling to form ferritic metal base of cast iron as cast condition. Casting weight was 78 tons, and flowrate of liquid high-duty cast iron was 90 tons. Quality of casting was inspected along with the structure and mechanical properties of high-duty cast iron (see table below).

Location of specimen cutoff		Mechanical properties				
		σ_w, MPa	$\sigma_{0,2}$, MPa	δ, %	KCV, J/m^2	K1C, N/mm^2·m$^{1/2}$
Tidal sample		424	282	21,4	20,4	–
Casting	20 °C	388	274	20,0	20,0	–
	250 °C	338	217	12,8	–	–
	Minus 40 °C	–	–	–	8,8	73

The pilot sample of SPS-146 was produced in premises of OAO “Petrozavodskmash”. According to IAEA regulations, its certification tests were performed successfully.

Taking into consideration new design, material science and technological developments, and existing production facilities it is possible to say that there are all prerequisites for producing containers of various designs using solid-cast SPS housings made out of high-duty cast iron with the weight up to 120 tons in our country.

Defining the spent fuel burnup depth, isotope composition and residual heat of VVER-1000 reactors. Using VVER MKS-01 and VVER MKS-03 installations

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Installations of MKS series for industrial measuring of VVER fuel burnup were developed in Russia. MKS installations allow determining the isotope composition and residual heat of the spent fuel on the basis of the measured burnup value. Installations are certified and registered in the state registry of measuring instrument and admitted for use in the Russian Federation.

Measuring methods (MM) using MKS installations have been certified and recorded in the Federal registry of measuring methods.

VVER MKS-01 installations have been implemented at NPP working with VVER-440 reactors (Kolsk and Novovoronezh NPP), VVER-1000 reactors (Kalinin NPP). VVER MKS-03 installation was produced and shipped to FSUE "Gorno-khimichesky kombinat" to be used for measuring VVER-1000 fuel burnup.

The core of the method used for determining the burnup level by MKS installations is that burnup is calculated in the basis of results of measuring SF radiation characteristics and design coefficients. Design coefficients have been tested using the results of destructive analysis of fuel burnup and isotope composition in OAO "GNTS NIIAR".

VVER MKS-01 installations are used to measure the fuel burnup in the water environment, and in air environment - using VVER MKS-03 installations. Nuclear fuel burnup time of one SFA makes 3-8 minutes.

Limits for relative accuracy in measurements using VVER MKS-01 installations and VVER MKS-03 installation are as follows (with the confidence probability 0,95): 10 % - for fuel burnup; 15 % - for the total weight of uranium and plutonium isotopes; 10 % - for residual heat.

Nuclear fuel burnup measurements of 19 SFA were performed at Kalininsk NPP. The maximum difference of measured and design (accounting) data on burning up makes 7 %. Thus, accounting data on burning up of measured SFA lies within the confidence interval of measured data.

Analysis of vacuum drying time required for the reinforced conrete container designated for spent fuel transportation and storage

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Recently the problem of spent fuel storage became especially actual for Russian nuclear power industry since wet type storage capacities for RBMK spent fuel are virtually exhausted, and the period of spent fuel storage under water in many cooling ponds verges to the maximum allowable time (30 years).

Alternative for the wet storage is the technology of dry storage of the spent fuel that had previously been kept in water in order to reduce radioactivity and heat emission. Dual-purpose reinforced concrete containers (RCC) TUK-109 have been specially designed for RBMK spent fuel (storage and transportation).

Considering that technologies of dressing SFA before loading into container do not provide for special operations for water removal from their component parts, then a certain amount of water will get into RCC internal part. In order to remove water vacuum drying technology is used, that is creation of sufficient expansion pressure in RCC internal part, which will cause intensive water vaporization, and generated vapor is removed from the container by vacuum pump.

The report primarily describes methodology of water calculation in the reinforced concrete container when it is loaded with the spent fuel, which considers the following factors:

- Surface wetting effect;
- Capillary effect in clearances, crevices;
- Retention of water in cavities that do not have a natural drainage.
- Dynamic loads during transportation and processing operations;
- Fluid evaporation from SFA surface in the hot dome.

Methods have been developed for the conservative estimation of vacuum drying time of UKKH-109 container loaded with spent fuel based on hydraulic model of vacuum drying. The program code CVDD-109 was developed to implement these methods.

Calculation model has been verified and results of vacuum drying calculation of TUK -109 container as compared to experimental data of Leningrad NPP.

Systematic calculations of RCC vacuum drying time, loaded spent fuel were performed in the wide range of operating conditions both on technology with tilting and without tilting of the lower bundle of fuel elements (LBFE). It was proved that absence of LBFE tilting increases the time vacuum drying of reinforced container several times because increasing the initial mass of water in the container (generally because of cooling water ingression in LBFE ampoules during SFA dressing).

In the most typical case (20 gamma camera SFAs in the container, emission into LBFE ampoules during additional dressing by 500 gr. of water) vacuum drying time increases from 18 to 62 hours, i.e. by 3,4 times, which makes it impossible to achieve the required rate of SFA transition from the wet to dry storage on the existing equipment.

Heat-resistant neutron protection material

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Considering existing plans of large-scale development of nuclear power engineering, one of the actual issues is development of shipping packaging set (SPS) of new generation for spent fuel assemblies (SFA) of powerful reactors with increased fuel burnup (GV $\geqslant$ 50 mW·day/kg·U). The important role of the engineering depends on the selection of neutron protection material.

According to modern trends in SPS engineering and conditions of their long-tern usage (more than 50 years), neutron protection material should be solid possess the following properties:

- High coefficient of neutron dose reduction on SPS surface;
- Long-term heat resistance (T ⩾ 200 ºC) and radiation resistance (D ~ 2 Mrad);
- Increased thermal conductivity, which will simplify heat removal from SFA to SPS surface;
- Acceptable price and processability.

Current neutron protection materials do not completely satisfy these requirements, which was served as the basis for this work performed by specialists of RFYATS-VNIITF (Snezhinsk) in cooperation with specialists of Zao "Termoksid" (Zarechnyi) and OAO "IRM" (Zarechnyi).

As a result of performed works:

- Composition ad technology of getting composite heat-resistant, neutron protection material were developed on the basis of magnesium-phosphate ceramics with powder filler from titanium hydride (a decision on issuance of patent No. 2012132730 was received on 31.07.2012);
- Samples from this material were produced and experimental and calculation investigations performed considering their mechanical, thermophysical and neuron protection properties;
- It was proved that the developed composite material is one of the most advanced by its properties for using in heat resistant (at the temperature of normal operation t ³ 250 °C) neutron protection of SPS intended for transportation of SFA of VVER reactors with the fuel burnup ³ 50 mW·day/kg·U.

Calculation experimental works ensuring safety when spent fuel handling of uranium-graphite reactors

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Important issues related to handling with spent fuel of uranium-graphite reactors are ensuring nuclear and radiation safety of works, and observance of requirements on registration and control of nuclear materials. Comprehensive facilities here are installations for measuring nuclear fuel burnup of MKS series, installation for measuring subcriticality of UIP series, installations for RBMK SFA tightness control.

According to NP-061-05, the measured values of burnup can be considered as a parameter of nuclear safety, which gives the possibility to reject the conservative approach while ensuring nuclear safety, where nuclear fuel is considered as "fresh" or insignificantly burnt. MKS installation also allow determining SFA isotope composition and residual heat щт the basis of the measured burnup value.

RBMK MKS-01 installation has been implemented at NPP working with RBMK-1000 reactors (Leningrad NPP). In the nearest future such installations are planned to be installed at Kursk and Smolensk NPP. Installation AMB MKS-01 has been developed for AMB SFA (Beloyarsk NPP), intended to be used on software “Mayak”. Installation DAB MKS -01 is designated for measuring spent fuel burnup of DAB-90 slugs, used in ADE reactors in FSUE “GKHK”. Installation model for measuring of EGP-6 SFA burnup has been produced and the installation model has been tested at Bilibinsk NPP.

Installation of measuring subcriticality UIP-006, used in SFA storage at Leningrad NPP has been developed for measuring the effective neutron multiplication coefficient in SFA storages and containers. The installation allows defining the effective neutron multiplication coefficient of SFA storage area, containing 30-40 SFA.

RBMK SFA tightness control installations, designed for Kursk and Smolensk NPP are intended for measuring the specific water activity in SFA storage tubes in at-reactor cooling pools and SF stages to separate SFA into tight and non-tight.

Installations of MKS series and installation UIP-006 are certified and registered in the state registry of measuring instruments and admitted for use in the Russian Federation. Measuring methods using installations have been certified and recorded in the Federal registry of measuring methods.

Calculation works were performed in at-reactor cooling pools and SF storages to ensure and radiation safety of SFA compact storage using measured values of nuclear fuel burnup as a parameter of nuclear safety. It was proved that in case of experimental validation of RBMK SFA fuel burnup, nuclear and radiation safety are ensured during the normal operation and emergency situations for any schemes of compact SFA storage in at-reactor cooling pools and SF storages.

Problems of experimental industrial operation of spent fuel container storage complex and ways to solve them

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Spent fuel storage and enrichment complex with the spent nuclear fuel is currently put into commercial operation at Leningrad NPP.

For the time of experimental industrial operation of the complex, which began in December of 2011, personnel has to face the problem of equipment breakdown and spent fuel assembly fragmentation cell reaching the maximum design output.

The first main direction of activities was improving of safety and reliability of SFPS CC equipment. Due to the fact that every breakdown of SFPS CC equipment that led to shutdown of SFA CE, was supplemented with the report on deviation investigation.

SFPS CC of Leningrad NPP has accumulated and submitted information to colleagues of Kursk NPP, Smolensk NPP and FSUE "GKHK", which allowed avoiding the occurrence of deviations. Also due to investigation reports, every failure was thoroughly studied with subsequent revealing of causes and developing a range of measures to prevent repeated failures. The whole complex of corrective measures, requiring considerable financing is included in reliability improvement measures that will be implemented by the end of 2014.

The second main direction of activities was increasing the performance of SFPS CC. Considering that in the beginning of 2012 the complex performance could not assure putting of the complex into commercial operation, a working group with involved specialists of OAO "PSR" was created in SFPS CC. Together they implemented a set of activities covering the whole cycle of works related to handling with spent fuel, beginning with at-reactor cooling ponds for spent fuel assemblies and ending with forming and sending of echelons with SPS to FSUE "GKHK". By results of performed works, operability SFA control elements improved from 5,8 to 12 SFA/day, time for receiving and supplying of UkKH-109 to OKH MBK unit reduced from 8 to 4,5 hours, meanwhile the completion of echelon with spent fuel SPS takes 4 days instead of 7 days. Measures that will stabilize and improve fixed results have been developed and they will implemented in the end of 2014.

Thus the set of activities intended to improve reliability and operability allowed to reach SFPC CC the design operability and begin the procedure of putting the complex into commercial operation.

Subcriticality monitoring of Leningrad NPP spent fuel pool during the transition of spent fuel assemblies to dry storage

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Since 2005 hardware-software complex SAPFIR_95&RC_KHOYAT, certified by Rostechnadzor has been used to maintain control of multiplying properties of Leningrad NPP spent fuel pool. This complex allows calculating subcriticality of spent fuel pool for the normal mode of operation and emergency situations, and prognosticating changes of spent fuel pool properties when placing high-enrichment spent fuel assemblies.

Beginning from December of 2011 spent fuel assemblies have been transited for long-term dry storage, which actualizes issues related to prognostication of spent fuel pool multiplying properties.

Using schemes of laying out spent fuel assemblies (approved by FEI NSD) in the spent fuel pool, while placing high-enrichment spent fuel assemblies (2,6 %, 2,8 %) and transition to dry storage of low-enrichment spent fuel assemblies (1,8 %, 2,0 %, 2,4 %) it is possible to maintain multiplying properties of spent fuel pool on the previous level.

On the basis of existing calculations, we can conclude that nowadays spent fuel pools have a sufficiently large reserve of subcriticality.

Using of mobile robotic complex in first generation spent fuel pool of Beloyarsk NPP

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Considering the intensive development of nuclear industry in our country, and industry world trends, a special attention should be paid to the issue of environmental safety and NPP safe operation by operating personnel. To improve safety of personnel in harmful labor conditions it will efficient to use mobile robotic complexes, such as MRK-27 that are successfully used in conditions of increased radiation exposure (for example, the first generation spent fuel pool). This robotic complex, considering the required modifications and using respective design of the instrument, has a large scale of possibilities for independent work, when a person is working only as an operator of the mobile complex without exposure or significant reduction of harmful factors, including also radiation exposure of the personnel. This, in its turn provides the possibility of long-term operation and more detailed development of situations accompanied by negative effect factors.

MRC is designed to develop methods of performing remote operation using attachments in compliance with MRC design.

MRC provides the performance of the following operations:

- Transportation of facilities and integral auxiliary equipment attached on the body;
- Opening and closing of devices (facilities) e.g. doors, gates, locks etc;
- Operation (readjustment on different surfaces) and installation of facilities weighing up to 100 kg on the process equipment;
- Video identification of objects with dimensions from 0,1 m at the distance of 1 - 20 m;
- External examination of objects with dimensions of 1×10 - 500×500 mm at the distance up to 1 m from MRC using video camera installed on the manipulator;
- Operation of MRC with luminosity of 5 of 10000 lx.

Weight of the complex in the wholly packaged condition makes 300 kg, meanwhile its maximum traverse speed reaches 0,7 m/s. The complex is capable of dealing with obstacles reaching the height of 100 mm, flights – up to 20 degrees. The basic design of MRC has the manipulator with the maximum bearing capacity of 100 kg, manipulator boom length from the terminal point of MRC body - 0,9 m. Manipulator arm is equipped with electromechanical drive with 5 degrees of freedom (rotation of rotary mechanism, arm lifting, forearm swinging, hand swinging, hand rotation). Maximum opening of gripper fingers is 250 mm. Control is carried out from the remote control room (RCR) along the cable at the distance of 60 meters, and 50 meters via radio channel. The mobile platform is track-mounted working with electromechanical drive. Overall dimensions of MRC in the stowed position are 1,12×0,71×1,4 m.

Water purification technology of SFA dismantling pool of AMB reactors at Beloyarsk NPP

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Preparation of spent fuel assemblies (SFA) of AMB reactors to radiochemical processing suggests SFA dismantling and separation into fuel (SFA) and structural (SRW) composition. Dismantling of SFA is performed under the sheet of water in a special pool and includes the operation of band sawing, which produces rasping of SFA and SRW. Rasping and spills, including radioactive materials from cutting, being dissolved in the pool water, distribute in the pool and go to AWT through submerged mesh filters with the mesh width of 100 microns. Specific activity of the pool water is defined by radionuclides Cs-137 and Sr-90, and cobalt, manganese, europium radionuclides.

The required amount of water in the pool of dismantling SFA is maintained by continuous pumping of water through the active water treatment system (AWT) with the flowrate of 20 m^3/h. Pool water parameters are maintained at the following level:

- Transparency (optical transmission) of water – min. 95 % concerning distilled water;
- Specific activity of γ-radiating radionuclides – min. $7{,}4 \cdot 10^6$ Bq/l;
- Salt content – min. 5 g/l.

Water treatment in AWT is performed successively in several stages. Separation of hard phase coarse fractions (more than 30 mcm) is performed on pressure hydraulic cyclones, from whence sludge flow with the rate reaching 0,5 m^3/h goes to continuous setting tank, where the hard phase is accumulated, generally consisting of the fuel composition.

To purify fine slurry (generally graphite), colloid particles a method of tangential ultrafiltration through tubular multichannel ceramic membranes with the pore size of 0,2 mcm is applied. Ceramic membranes are located in membrane filters in the form of portable cartridges. Membrane filters and circulation pumps form the circulation circuit of ultrafiltration modules. The mode of tangential ultrafiltration allows monitoring slurries in the circulation circuit to the concentration of 50 g/l.

Removal of cesium radionuclides from water is performed by filters with ferrocyanide sorbent with replaceable cartridges. The final afterpurification from the rest of the radionuclides and ions is performed in ion-exchange columns filled with cationite and anionite.

Dehydration of graphite sludge, construction steel and fuel is performed periodically on centrifuges with the capacity of max. 250 l/h. The centrate is returned to AWT receiving tanks, and the sludge with the humidity less than 20 % is unloaded to containers that will later be sent for vacuum drying.

AWT equipment is located in nonoccupied canyons. Sludge treatment equipment is classified as safe equipment of И type according to NP-063-05.

Construction solutions concerning the main AWT equipment, process modes of water treatment and regeneration, including sludge supply have been tested using cobbled units and simulation media.

Development of emission tomographs for spent fuel monitoring

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It provides results of modeling and describes principles of creating emission tomograph for individual monitoring of SF burnup in SF long-term storage tubes. It is proved that accuracy of defining the integral activity of every SF can reach 5%.

The report also specifies the possibility of developing emission tomographs for:

- Individual monitoring of FE burnup in SFA of NPP working with VVER during their sending from NPP and acceptance for processing in ODTS;
- Monitoring geometry of SFA of NPP working with RBMK (before their submission for dismantling), which are in storage tubes of SFAS cooling ponds, without their withdrawal from water.

It is planned that development of tomographs will use the same principles as for creating tomographs for monitoring SF burnup.

Introduction of automated systems of process control for spent fuel handling

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Implementation of automated systems of operating technical accounting and control of NM (ASOTA&C of NM) is performed to improve the level of process safety and update NM A&CS when transferring SF to dry storage at NPP working with RBMK and subsequent handing of spent fuel in FSUE “GKHK”.

ASOTA&C of NM allows maintaining automated control in the shielded box of SFA dismantling on a real time basis both for technological processes of handling, temporary storage, dismantling and loading of spent fuel in MBK reactor, and actions of operating personnel of SF storage while performing process and control operations.

ASOTA&C of NM is wholly based on home-made hardware, developed by RFYATS-VNIITF for NPPs.

Remote-operating system of effective searching, inspection and elimination of leakages in coating of cooling pools of RBMK NPP SF storage

Remote-operating complexes of effective searching, inspection and elimination of leakages in the coating of cooling pools (intended for operation in conditions of practically complete filling of cooling pools with the spent fuel (SF)) have been developed for effective elimination of emergency leakages in coating of spent fuel cooling pools in RFYATS-VNIITF.

Complexes differ in their work area (bottom, walls and corners of pools), and conditions of use (in clearances between the spent fuel or on the clear area of the bottom). Complexes provide for successive automated examination of pool bottom and walls using submersible remote-operating modules.

Complexes ensure effective elimination of the detected leakage in the bottom and side part of pool coating by glue filling immediately after its detection using equipment of the remote-operating module.

1.2.5. FIRE SAFETY

Fire state of NPP and main directions of activity in the area of fire safety enhancement

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Analysis of fire state of all operating NPPs is described.

At the operating power units of JSC «Concern Rosenergoatom» for the 14 years 15 fires occurred (no fires occurred in 2005, 2007-2009, 2012 and 2013).

No fire occurred produced any impact on operation of safety systems and RP main equipment nor they resulted in any significant material losses, safe operation limits and conditions of the power units were not violated at this.

With the aim to reduce the risk of fire outbreak and reduce the possible consequences when it outbreaks JSC «Concern Rosenergoatom» performed a complex of measures for ensuring and enhancement of fire safety of the operating power units of NPPs.

Work on fire safety assurance was carried out in JSC «Concern Rosenergoatom» according to below main directions:

- updating of the branch normative and regulatory legislative base for the fire safety issues;
- enhancement of fireproof stability of NPPs in accordance with requirements of regulatory documentation in force;
- technical upgrade of obsolete automatic systems of fire detection and extinguishing and fire alarm systems for the personnel annunciation;
- training of NPP personnel for actions if fire outbreak;
- implementation of measures on «Plan of measures for fire safety enhancement at operating nuclear power plants for 2013-2017»;
- implementation of organization and technical measures at all NPPs for preparation for spring summer fire hazardous time period;
- conduct of inspections of the fire safety state when NPP power units construction;
- inspection of fire safety state at NPPs when conduct of integrated and target inspections of the operating NPPs;
- development of measures for enhancement of monitoring efficiency of fire safety assurance at NPPs under construction.

Main directions of activity of the fire safety service of JSC «Concern Rosenergoatom» (FSS of concern) in the area of fire safety assurance at the stage of design and construction of NPP are described.

Fire safety departments were set up in the directorates of NPPs under construction and in some structural units of «General designer» control of fire state was strengthened at the facilities being constructed and the tendency of the reduction of the number of fires was traced.

FSS of the concern has conducted the work to address development of normative technical documentation in the area of fire safety.

In 2013, in accordance with the requirements of the effective fire safety related documents more than 80 fire fighting measures were conducted, including:

- at power units 3,4 of Novovoronezh NPP light aeration lamps of the turbine building were equipped with motor drives for smoke removal;
- at power units 1-4 of Bilibino NPP fire retarding valves were installed/replaced;
- at power unit 4 of Kola NPP and power unit 6 of Kola NPP and power unit 1 of Balakovo NPP the document «Analysis of fire impact on safe shutdown and cooldown of reactor plant» was updated and modified;
- at power unit 3 of Kalinin NPP the outside fire water lines were upgraded and replaced.

Level of performance of fire fighting measures for mitigation the comments from the State fire supervisory bodies is 100%.

Based on the approved «Program of equipping and technical upgrading of the object facilities of FPS for NPP protection for 2012- 2016» and through Order of JSC «Concern Rosenergoatom» of 01.02.2012 № 9/81-П 23 fire vehicles were delivered for fire protection of NPPs.

Usage of mobile fire-extinguishing means for fire suppression at NPP at extremely low environment temperatures

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More than 85% of the Russian territory lies in cold climatic regions characteristic of severe meteorological conditions, especially during epy wintertime period. Such a dangerous natural phenomenon as abnormally cold weather that is accompanied by extremely low environment temperatures can facilitate occurrence of emergency situations on the basis of the whole region. Meanwhile it is necessary to taking into consideration that in Russia 30% of nuclear power plants are located in cold climatic regions, and by 2030 50% of nuclear facilities are going to be located on these territories. Irrespectively of meteorological conditions, these facilities should be properly protected.

Negative influence of low air temperatures limits the tactical potential of fire subdivisions; especially it affects the operability of mobile fire-extinguishing means. Performed investigations allowed defining the major part of fire equipment failures related to their exposure to low air temperatures. Consequences of this exposure, as a rule, complicate the operating conditions in case of fire, which also requires additional forces and is accompanied by extension of operating time of subdivisions, and respectively more serious consequences from fires.

Academy of SFS of the Russian Ministry of Emergencies have performed investigations on performance evaluation of mobile fire-extinguishing means used by service subdivisions to protect nuclear power plant. Performance of pump-hose systems of fire vehicles and fire service subdivisions as a whole was evaluated in a broad range of possible most unfavorable meteorological conditions. The received results allow to draw a conclusion on the level of NPP protection from fire development to large fires at extremely low temperatures; perform preliminary planning of required forces and facilities and take decisions on the necessity to use special technical means to ensure operability of pump-hose systems of mobile fire-extinguishing means in extremely low environment temperatures.

Fire risk assessment at NPP in Slovakia

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The report describes approaches to carrying out an analysis and assessment of fire risk at NPP in Slovakia. Considering fire risk assessment, new methods frequently contradict the implemented deterministic order of presenting documents on fire safety of buildings and constructions. They have traditionally been implemented in many counties and are used during development and assessment of fire safety and NPP facilities.

It was the above-mentioned fact that lead to differentiated approach in investigating and assessing fire risks at NPP in Slovakia.

Deterministic analyses and fire risk assessment include presenting of documents on fire safety of buildings and constructions, which is ensured by drawing up and approving of fire safety projects of these buildings and construction in the competent authority of state administration – Ministry of Home Affairs (hereinafter referred to as - MHA) of Slovak republic (hereinafter referred to as SR).

Probability analyses and fire risk assessment at NPP requires a governmental nuclear surveillance – Nuclear Control Department (hereinafter referred to as – NCD) of SR, however they present only some “superstructure” for deterministic design decisions. That is to say the matter of fire risk assessment in case of probabilistic analysis is not the assessment of fire safety of buildings and constructions and their technologies. Virtually all of these analyses were oriented to searching and in-depth analyzing of design weak points. The bigger part of them is dedicated to identification and analysis of joint cable routings of different safety systems and equipment. According to normative requirements of SR, fire protection is confirmed by project documentation for fire protection systems of respective building. The listed types of analysis are required by SR NCD, West-European Regulatory

Association (WENRA), IAEA, Association of insurance companies insuring NPP equipment (NUIP), and WANO, the more so much that these types of analysis are a good supplement din the process of preparation, assessment of the developed design documentation of new and modernized construction facilities of the power unit.

Accident at the Japanese NPP proved that it is necessary to reassess these new methods, which can be a significant progressive impulse for further fire engineering development.

Organization of fire protection at NPPs. Fulfillment of decisions and agreements on cooperation with EM of Russia

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Currently fire protection of all operating NPPs is carried out by the object subdivisions of Federal fire fighting service (FFFS) in accordance with the Resolution of the Russian Federation Government of April 23 ,2005 г. № 477-рс.

To ensure the fire protection of new power units at operating NPPs it is necessary to increase the number of available subdivisions of FFFS (for set up of preventive groups),and the new NPPs shall be included into the List approved by the above Resolution of the Russian Federation Government with assignment of additional staff for FFFS.

Long-term State power program on construction and commissioning of new power units of nuclear power plant of power industrial complex of Russia was developed and is implemented in the State Corporation «Rosatom». The important constituent of the State strategy of nuclear power industry development is ensuring of the relevant level of fire safety of the new power units of the NPPs both being constructed and commissioned.

Within the frame of implementation of the «Program of activity of the State Corporation for nuclear energy «Rosatom» for the long-term period (2009 - 2015)» new power units at operating NPPs are commissioned. The program stipulates construction completion of the power units with the high degree of readiness - 6 power units, construction of new nuclear power plants - 28 new power units at 10 NPPs (before 2020). With this respect the issue of their fire protection assurance is very urgent.

For ensuring effective management of FFFS object subdivisions, taking into account specific nature of fire extinguishing and conduct of emergency rescue activities, including those conducted under the condition of radiation accident, organization of preventive work for fire preventing at the NPPs, propose the alternatives for organization of fire protection at NPPs.:

On April 24-25, 2013 in Moscow the conference «Ensuring and enhancement of fire safety at NPPs» was held with active involvement of managers and specialists of the State Corporation «Rosatom», EM of Russia and JSC «Concern Rosenergoatom» in its organization. Below are the main decisions made at the conference:

1. Ensure the priority-oriented financing and fulfillment of «Plan of measures for fire safety enhancement for operating NPPs for 2013-2017» within the set up deadlines. Purchase of fire equipment under the «Program of equipping and technical upgrading of FFFS object subdivisions for NPPs protection for 2012 - 2016».

2. Ensure fulfillment of the «Plan for fire protection improvement of newly constructed power plants by JSC «Concern Rosenergoatom» and enhancement of fire safety control effectiveness».

3. Conduct study (test) of the most efficient modern systems, means and methods of fire extinguishing on electric installations under voltage up to 10 kV with account of safe conditions of the personnel and manning staff of FFFS object subdivisions for NPP protection being ensured.

4. Organization of fire protection on the newly commissioned power plants and determination of the number of the manning staff for existing FFFS object subdivisions for NPP protection, as well as set up of fire protection units for protection of newly constructed NPPs.

One of the most important and challenging issues under consideration at the conference was the procedure and time of organization of fire protection at newly constructed and commissioned NPPs. In the fist turn this is the possibility to set up FFFS object subdivisions of EM similar to the operating NPPs included into the «List of organizations wherein the object and special subdivisions of Federal fire fighting service are included» approved by the Resolution of the Russian Federation Government of 23.04.2005 № 477-pc.

In view of the above, within the frame of implementation of the «Plan of fire protection improvement of the NPPs constructed by JSC «Concern Rosenrgoatom» approved by Mr. Kirienko and enhancement of effective control of fire safety state the issues related to organization of fire protection at NPPs were elaborated.

Instrumentation and control system of NPP fire protection «Global-NPP»

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I&C-FP system is designated for fire control of NPP units, fire detection and suppression, control of water and gas fire-extinguishing installations, generation of commands related to controlling of smoke removal and evacuation management systems.

I&C-FP "Global-AES", developed by the group of companies "Rubezh", wholly manages and performs information functions according to SP 13.13130.2009 "NUCLEAR POWER PLANTS. Fire safety regulations".

"Global-AES" includes a hardware complex, designated for automatic fire detection, automatic, manual and remote startup of fire-extinguishing systems, transfer of information on fire to ventilation I&C hardware, actuation of fire annunciation, generation of information on fire and process of its extinguishing in control rooms and in the fire service subdivision, control and diagnostics of the fire alarm condition and control modules.

The system consists of the three main levels:

- Top – AWS and servers with the specially installed software FireSec;
- Average – system kernel on the basis of high integrity group controller (GC) and address facility controllers (AFC);
- Bottom – fire detectors (smoke, thermal, combined, manual), no-address actuation modules, engineering system control modules, fire-extinguishing control cabinets.

Peculiar characteristics of "Global-AES" system:

- Information capacity of the group controller for 50 000 address devices;
- Maximum response time to the event (fire, failure) - not more than 1 sec;
- Ring topology of the address train with the branching feature;
- Automatic addressing of address devices;
- Length of address communication line 1 km from device to device;
- Logbook for 16 mln of event with the black box function;
- Increased interface protection from electromagnetic interferences;
- Multi-stage protection from false activations;
- Possibility of integrating with the video surveillance system for displaying a picture to operator's AWS from the camera located nearest to the fire place;
- System components stay operable and autonomously perform algorithms in case there occurs a communication failure with the group controller.

Design of I&C-FP "Global-AES" considered a positive experience of using the address system "Rubezh-10A" since 2007 at Balakovo, Beloyarsk NPP, nuclear objects and commercial objects.

Usage of fire-extinguishing installations with fine water spray system for NPP objects

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Currently, usage of automatic fire-extinguishing installations with fine water spray system (hereinafter referred to as AFEI FWSS) becomes more frequent for protecting different kinds of facilities. It is primarily related to the fact that the general specific water flow required for fire extinguishing using finely sprayed water (hereinafter referred to as FSW) reduces significantly in comparison with traditional systems of water fire-extinguishing, which in its turn is related to the higher fire extinguishing factor of FSW [1 – 4].

Usage of this technology for NPP facilities is actual, since the usage of traditional systems with existing standard water supply indicators will create a number of problems, both in designing and NPP operation, which is due to the necessity to ensure removal of large amount of water to prevent flooding and protection of expensive digital equipment.

At present the main documents of the Russian Federation (which require their observance in designing of AFEI FWSS at NPP objects) are the Federal Law No.123-FZ "Technical regulations on requirements to fire safety" dated 22.07.2008, Set of rules SP 5.13130.2019 "Fire Protection Systems. Fire alarm system and automatic fire-extinguishing installations. Design norms and rules", Set of rules SP 13.13130.2019 "Nuclear plant. Fire Safety Regulations", and OPB-88/97 NP-001-97 (PNAE G -01-011-97) "General provisions of ensuring fire safety of nuclear power plants".

The main problem for designing of AFEI FWSS is that the specified documents do not contain normative indicators of FSW supply (intensity and time) for protection of premises and constructions with transformation of different fire hazardous substances and materials. This circumstance is the basis for developing special regulations (hereinafter referred to as SR) with regard to ensuring fire safety of the object, which requires additional time and labor costs for designing. Meanwhile it should be noted that SR developing and approving according to the established procedure there arises a necessity to perform appropriate field tests to confirm the standard supply indicator specified in them.

In its turn item 1.2.5 of OPB -88/97 NP-001-97 (PNAE G -01-011-97) "General provisions of ensuring NPP fire safety" also reads that technical and organizational decisions taken to ensure NPP safety should be evaluated by the previous experience or tests, investigations, operating experience of prototypes and meet requirements of normative documents.

Considering the above stated, it should be noted that implementation of AFEI FWSS at NPP requires carrying out a set of field tests, accounting the specificity of the most fire hazardous protection objects, which are storage rooms for combustible fluids (diesel fuel and oil), SDPP, PDPP, oil-filled equipment and cable constructions. After completion of tests a document should be developed for designing these installations, including requirements on the necessity to cooldown heated cable strands to prevent their repeated inflammation, FSW supply before arrival of fire teams or until the complete burning up of the load in "dead zones" etc.[5,6].

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Fire alarm systems of Smolensk NPP (performed modernization and key problems of operation)

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Three power units, 1000 MW of electric power each, are under operation at Smolensk NPP.

Specified lifetime of each power unit is 30 years. With account for performed works on modernization, service time unit №1 was extended for 10 years.

Process diagram of the power units - single-circuit.

Power units №1 and №2 belong to second generation, power unit №3 - to the third generation of power units with reactors of RBMK-1000 type.

Each power unit includes RBMK-1000 reactor with forced circulation circuit and auxiliary systems, steam and condensate-feed paths, two K-500-65/3000 turbines with TBB-500 generators of 500 MW power each.

Smolensk NPP power units were commissioned over the period from 1982 to 1990. Designs of fire alarm units were developed with account for then effective regulatory documents, analogue equipment, produced by industrial facilities of the Soviet Union, was used (fire detector of types: DTL, IDF-1m, DIP-1, etc.., receiving stations TOL-10/100, FDS-1).

Service life of the used equipment was almost exhausted in the 90s, number of failures per year at one unit started to exceed several hundreds. Consequently, reliability of fire alarm units was dropping, which began to cause certain uneasiness for plant management.

In 1996 decision on modernization of fire alarm units was made. Siemens company equipment was used for main production, for auxiliary facilities -

equipment of Russian manufacturing company “Bolit”, which was caused by specific character of production and economic estimates.

Why was Siemens Company chosen? Siemens was at that time and still is the world leader in the area of manufacturing of equipment of fire detection. Also, we believe that this equipment is the most adapted for use at Russian NPPs with reactors of RBMK type in terms of its technical specifications.

In 2002 the first stage of modernization at the 1st power unit was completed, in 2012 - modernization of AFS at all 3 power units was completed.

Operating life of the main equipment of AFS - 60 years.

There has been no technical problems during operation of AFS units based on Siemens Co. equipment.

At the same time there are unresolved issues related to different understanding and interpretation by management documents by operating organization as per work of AFS units at NPP.

At present the scheme of fire alarm units actuation information transmission as per «Fire signal» from all objects to fire brigade of the objects in automatic mode has been implemented at Smolensk NPP. Altogether, about 20 000 fire detectors of different types, mainly smoke, operation principle of which is based on the fact of change of optical thickness of environment, are under operation.

Consequently, change of optical thickness of environment (fog, dust, and aerosol) is not always related to fire/combustion and is not false actuation of fire alarm units.

Besides, detection of damages or failures of elements of automatic firefighting during equipment walk-down or during inspection of operability are normal procedures of inspection/production control and maintenance and repair. The level of these deviations – events of low significance (low level events).

At the same time RD EO 1.1.2.01.0163-2013 «Provision on organization of investigation of events at nuclear power plants important for safety and reliability by JSC «Concern Rosenergoatom» requires to classify each event of fire alarm actuation as per categories of events at NPPs - Ц11.

Other regulatory document RD ЭО 1.1.2.01.0331-2010 «Provision on the order of on-line information about the operation of nuclear power plant transfer to JSC «Concern Rosenergoatom» and concerned organizations». In accordance with appendix E (i. E5.3) it requires «...submit the online message at any event (fire, ignition, explosion, fire alarm actuation including spurious) on the territory or on the facilities of NPP induced by anthropogenic or natural reasons or by the personnel actions as well as events causing the fire unit arrival at the territory or facilities of NPP».

The specified requirements, in our opinion, are redundant and are not justified by the requirements of the effective fire safety rules and are not understood both by the NPP management and by the engineering personnel.

Fire safety assurance when lifetime extension of power unit 4 of kursk NPP

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Power unit 4 of Kursk NPP was commissioned in 1985.

Designated design lifetime of the power unit is 30 calendar years and expires in December 2015.

In 2008 the investment project «Extension of operation lifetime of Kursk NPP power unit №4» was developed that defined the measures implemented within the time period 2008 through 2015 and ensured the safe operation of power unit №4 in accordance with the regulatory documents in force for the period of additional (extended) lifetime of power №4 to not less than 15 years. With account of the specified project the «Program of Kursk NPP power unit №4 preparation for additional operation time» №КуАЭС 4ПРГ-92К(1.34)2009 was developed and approved.

The «Program...», as well as the «Plan of measures for fire safety enhancement and fire safety systems upgrading at NPP for 2008-2012» approved by the operating organization stipulated performance of the below measures at the power unit 4:

- replacement of combustible plastic on the evacuation routes in the controlled access areas;
- upgrade or replace the elements of automatic fire alarms system that have been operating beyond the set up service life;
- bring the fire-resistance rating of carrying metal structures of turbine building to 0,75 hour;
- provide non- constantly attended rooms with electronic appliances of automatic control of NPP technological process with the automatic installations of gas fire fighting;
- upgrade the systems of smoke removal and space air overpressure on the personnel evacuation routes.

In December 2013 JSC «VNIIAES» completed the improvement and amendment of «Analysis of fires impact and their consequences for safe shutdown and cooldown of reactor plant of power unit 4 of Kursk NPP». The purpose of this study was determination and justification of sufficiency of the fire protection level of the power unit taking into account the executed modifications and measures, as well as evaluation of possibility to ensure conditions for reactor plant safe shutdown and cooldown in case of fire outbreak.

Based on the results obtained when «Analysis...» updating, it was noted that at power unit №4 of Kursk NPP the safety principles were implemented that enabled ensuring the acceptable level of its fire protection.

In the course of analysis no emergencies initiated by fire were detected that would directly result in damage of the reactor core due to damage of the ways of safe shutdown.

Updated analysis of the fire consequences in the rooms with equipment involved in reactor safe shutdown and cooldown demonstrated that at power unit №4 the possibility of reactor plant transfer to subcritical state and its cooldown by the technical means available at the power unit is ensured.

Complex of innovation technologies for fire safety assurance of NPP

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Vector of the enterprise development– implementation of the complex of innovation technologies for fire safety assurance of NPP and increase of economic efficiency of these technologies implementation.

Innovations for fire safety assurance of the facilities consist, fist of all, in systematic approach based on comprehensive study of interactions of practically all engineering systems of NPP life support.

A simple task is to be solved – based on the technical specification set up by the Customer establish an economically effective system of fire protection capable to mitigate fire outbreak in its initial stage in automatic and remote modes including on the remote facilities.

- Set up of scientific test complex. We suppose that this is the most advanced non commercial project addressing the implementation of practical inspection of separate non standard technical approaches based on simulation of fire evolution in specific environment.
- Address system of fire alarm and control of fire automatic equipment «Turmalin» was developed (ASFAC «Turmalin»).
- ASFAC «Turmalin» operates under control of OC in real time QNX. Main elements of system equipment have 100% of hardware duplication. SW was developed for upper level SCADA on the base of QNX.
- We mastered the serial production of a ruler of modules for fire extinguishing of gases in the volume of 60, 80 or 100 liters.
- MPG modules are used with the following chemical gas extinguishing compositions: freon HP 125, freon HP 318ts, freon HP 227ea.

The following innovation element of the fire protection system – system of automatic water fire-fighting on the basis of the principle of "finely sprayed water".

- Our specialists, considering domestic and foreign experience developed and technically implemented the automatic fire-fighting installation using finely sprayed water "Votum" for different premises of protected objects.
- Physical and chemical properties of supplying finely sprayed water allow effectively suppress the dangerous fire factors and create necessary

conditions for a more effective usage of forces and facilities applied by fire-fighting and emergency rescue services.

Implementation of innovative technologies is still actual, concerning the solution of issues of improving efficiency of the control over ensuring NPP fire safety. Solution of these issues certainly requires professional skills and a sufficient preparation period before testing. In case there is a multilevel fire safety system, then the issue of confirming the observation of regulations is quite labor-intensive.

- We introduced QR-coding to reduce labor intensity of the control over our equipment considering modules of gas fire-fighting and control-and-indicating devices.
- From the economical point of view, costs for introducing automated accounting will be insignificant.
- There is no doubt that introduction of multiple databases and their manual correction is quite labor-intensive and admits a large probability of error.
- Extension of QR-coding for all of the NPP operated equipment

Subsection 1.3
RADIATION SAFETY, NPP ECOLOGY, EMERGENCY PREPAREDNESS

Complex assessment of radiation ecological impact of NPP based on data of integrated radiation situation monitoring

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Evolution of nuclear engineering requires that highly efficient system of safety assurance of public and environment be constructed. An important tool of safety assurance when nuclear energy use is radiation monitoring of environment, this means the system of regular observations over the content of radionuclides in the components of environment and other parameters of radiation situation with the purpose of prompt detection and forecast of consequences unwilling for humans and ecosystems. At today's stage of nuclear engineering development as one of the most strategic branches of Russia's economy in one line with the growing need of capacity increase, the studies addressing the assessment of the nature of radiation impact on humans and environment are of particular urgent.

Based on data of many years radiation monitoring, assessment of radio ecological impact of reference power units was performed. The number of assessed reference power units covered such NPPs as Novovoronezh NPP, Leningrad NPP, Balakovo NPP and Kola NPP.

It was demonstrated that under the routine operation conditions the planned exposure rates of public from NPP releases and discharges are considerably lower than the permissible values. According to design estimates, annual doses of planned radiation exposure of public from the routine releases at Novovoronezh NPP make up $2{,}0 \cdot 10^{-7}$ Sv, Leningrad NPP — $1{,}5 \cdot 10^{-7}$ Sv, Balakovo NPP — $1{,}1 \cdot 10^{-9}$ Sv, Kola NPP — $6{,}4 \cdot 10^{-10}$ Sv. In practice, these doses are in consistency with the level of negligible radiation risk, which is as small that it does not require special measures for its reduction. Compared to radiation induced by NPP releases, higher doses are specific for existing radiation, which is stipulated by regional man-induced radiation background caused by the past activity of the NPP or global radioactive releases. In the number of cases the main contributor to anthropogenic background is given by ^{137}Cs of «Chernobyl» origin. It should be noted that the doses of existing radiation are considerably lower than the permissible dose limits.

For the first time in accordance with new international basic safety regulations ONB ОНБ-2011, in line with radiological assessments for the public, the radiation impact estimates for biota were calculated. It was demonstrated that radiation doses of objects of biota in the vicinity of NPPs location are considerably lower than reference safe levels of radiation.

Analysis of data of many-year radiation monitoring and model estimates show that radio ecological situation in the vicinity of NPPs under consideration are within the norms with respect to both public and biota objects, i.e. is in consistency with criteria of favorable environment.

EDF nuclear rapid response force: status at may 2014

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EDF, France

Right after the Fukushima event, EDF decided to reinforce its crisis organization, by implementing a nuclear rapid response force (FARN), able to provide quickly assistance to a plant facing big difficulties, with human and equipment support.

The following aspects will be presented:

- Creation context and design hypothesis of the FARN,
- Regulator's requirements
- Organization of the FARN. Human resources
- Intervention of the FARN. Equipments and logistic mean.

Environmental protection of Russian NPPs — current state and persepctives

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Environmental protection and rational use of natural resources are the most important tasks for JSC «Concern Rosenergoatom» (hereinafter — Concern). To observe the requirements of environmental legislation the ecological services of nuclear power plants perform production ecological monitoring and assess the state of ecological safety to elaborate the prompt and effective solutions for minimization of nuclear power plant impact on environment. The main tasks of NPP ecological services consist in supervision over environmental quality norms observance.

The principles of environmental activity and obligations of Concern in the area of ecological safety ensuring are stated in the Ecological Policy of Concern updated in 2013, the goal of which is ensuring of such a level of NPP safety when impact on environment, personnel and public for the nearest future and in the long run will ensure safety of natural systems, maintain their integrity and survival functions.

In order to achieve the goal and implement the main principles of ecological Policy the Concern committed to implement and support the best methods of ecological management. For practical implementation of ecological Policy the measures are annually developed and implemented to address the reduction of negative impact on the environment when NPP operation.

Since 2011 the ecological management systems of all operating NPPs and central headquarter of the Concern have been certified for compliance with the requirements of international standard ISO 14001:2004 and national standard GOST Р ИСО14001-2007.

Efforts taken for implementation of ecological Policy of Concern practically confirm the adherence of Concern to implicit observance of environmental legislation requirements and ecological safety of NPP assurance.

Thus, for the recent 10 years the volumes of releases and discharges of contaminating substances into the environment almost two times reduced, processes of formation and interim storage of production wastes, their disposal on the own test ground and transfer to the dedicated organizations are constantly updated.

The main directions of environmental protection activity for the nearest future are: retaining of the achieved level of responsibility of the Concern branches with respect to observance of regulations and validity conditions of approving and licensing documents in the area of environmental protection; implementation of the best existing technologies of contaminated production release treatment, contaminated substances discharge to water pools, production and consumption wastes handling methods, resource saving; updating of the monitoring programs and graphics of radioactive substance release to atmosphere, ensuing that technical and methodological base of production ecological monitoring of radioactive substance release to atmosphere are in conformity with the requirements of international and national standards in the area of calibration.

Three dimensional simulation models of radioactive substances distribution in surface layers of air for NPP observation zone

P.A. Bunto

The report presents the results of development and use of special three dimensional simulator for distribution of gas-aerosol releases with radioactive substances in the areas of nuclear facilities and actions of dedicated departments during emergency evolution.

During emergency evolution at nuclear facility radioactive substances can be released to environment. For simulating the distribution of radioactive substances (RS) in the environment the software based on the models of gas aerosol additives distribution in the atmosphere are used on site, protected zone and observation zone. Majority of software can submit graphic display of design characteristics (exposure fields, volumetric concentration or surface density of radionuclides drop) to the users in the form of two-dimensional projection on the maps of the regions of distribution. The alike presentation of emergency evolution requires simultaneous analysis of some additional data in order to get a full picture of events evolution, which complicates delivery of information to the wide public.

Apart from simulation and forecast with the help of software of RS concentration fields and exposures based on models of gas aerosol additives distribution in atmosphere, software simulating operation of ARSMS detectors when radioactive release at nuclear facility is used in the system of emergency response. The system of ARSMS operation simulation enables to get the notion of the operation of real networks of detectors when radioactive cloud with RS moves through the region and learn about exposure characteristics of radiation fields at the specific measuring points. However, the process of visualization of RS containing cloud distribution is not connected to the operation of ARSMS simulator system in the mode of real time. Therefore, the purpose of the study is to set up software «Special-purpose three dimensional simulator for distribution of gas-aerosol releases with radioactive substances in the areas of nuclear facilities and actions of dedicated departments during emergency evolution», which is synchronized with the operation of ARSMS simulator system.

This simulator contains a three dimensional model of NPP OZ land form, on which the satellite high resolution photos are laid, as well as models of NPP site, dwellings covered by observation zone (OZ), ARSMS detectors, roads, water recourses and wood lands.

Three dimensional observation models of Kursk NPP OZ, Leningrad NPP OZ, Kalinin NPP OZ and Novovoronezh NPP OZ have been developed for this simulator.

For three dimensional models of observation Kursk NPP and Leningrad NPP have been simulated in a full scope. Observation zones of Kalinin NPP and Novovoronezh NPP have been simulated in a simplified form. For them the model of NPP site, model of NPP satellite town, as well as models of RSMS detectors have been setup. These models are arranged on the map of NPP OZ without the land form taken into account.

For simulating the evolution of radioactive contamination software «NOSTRADAMUS», developed by RASEB is used.

The simulator enables to visualize the dynamic of volumetric concentration of radiation contamination in the NPP OZ caused by emergency, response of ARSMS detectors, dynamic of getting the messages about emergency by the ministries and agencies, ways of the plant personnel evacuation and information about the subdivisions involved in mitigation of emergency consequences at NPP site and places of their deployment.

The simulator also visualizes the picture of concentration of surface radioactive contamination, total exposures received by public as a result of emergency, as well as needed measures for public protection against radioactive contamination on the three dimensional model of land.

In the course of simulator development the simulator functions were integrated into it related to organization of evacuation management to enable drills conduct for the specialists responsible for organization of evacuation in case of emergency at NPP.

Use of the proposed visualizer will enable a better informing of State Corporation «Rosatom», Federal and local authorities, civil defense and emergency centers, mass media and public about the emergency related to disturbances at nuclear facility. Special purpose three dimensional visualizer will enable the specialists, based on virtual reality technology, to prepare the objective and illustrative picture of hypothetical accident evolvement for the authorities and mass media.

Moreover, use of visualizer will enable to demonstrate to the wide public that even though МПА occurs at nuclear facility the exposure rate for public will not differ much from the exposure rates under normal conditions. This will strengthen the confidence of the public in high reliability and safety of Russian NPPs.

Sources, accumulation and migration of tritium and carbon-14 at VVERs

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Operation of NPP with VVER results in formation of a large number of isotopes ^{3}H and ^{14}C. In spite of a large number of studies issued with respect to this subject, till now there are no calculation methods for ensuring the obtaining of data on volumetric activity of these isotopes in the process media of equipment of the support systems under various periods of power unit operation, as well as regarding their migration and formation of the sources of release and discharge at VVERs. These data are needed for creating the protective barriers hindering delivery of ^{3}H and ^{14}C to the environment and justification of radiation safety both for new projects and for power increase at operating VVER-1000.

For calculation of isotopes ^{3}H and ^{14}C accumulation the software MUHTAR was used in the design of VVER-TOI, this program simulates the changes of water chemistry parameters of the primary circuit and process media of WC support systems depending on the parameters of reactor operation mode. For this purpose the possibility of calculations of formation, accumulation and migration of isotopes ^{7}Li, ^{14}C and ^{3}H in the process media was implemented in the algorithms of program MUHTAR. The structural characteristics of equipment of the circuit and WC support systems of the power unit are used in this program as input data.

From the analysis of the obtained results it follows that the volumetric activity of ^3H in process media of power unit with VVER-TOI reaches maximum to the 20-th fuel cycle and makes up:

- in primary circuit coolant about $2,0\times10^8$ Bq/dm^3;
- in check tanks of unbalance water of KPF system about $9,0\times10^7$ Bq/dm^3;
- in water of spent fuel and refueling pools during refueling about $4,0\times10^7$ Bq/dm^3.

Total release of tritium for the reactor cycle from one power unit after balance reach amounts to 33 TBq. Total release of tritium with unbalance water of one power unit after balance reach for the power cycle amounts to 20 TBq.

Volumetric activity of ^{14}C in coolant and support systems reaches balance after the first cycle of reactor operation. In the circuit coolant activity of ^{14}C makes up $5,5\times10^4$ Bq/dm^3. In blowdown water at the output from blowdown deaerator - $4,7\times10^4$ Bq/dm^3. Total release of ^{14}C for the power operation cycle form one power unit makes up 440 GBq. Discharge of ^{14}C with unbalance water is possible on account of formation of difficulty soluble forms of carbonates in KPF system and amounts to less than 0,1 % of the total volume of formed ^{14}C .

For the purpose of verification additionally the calculated results of volumetric activity of tritium in the process media of VVER-1000 support systems were compared with the data of tritium control at Balakovo NPP.

Calculation results under the program will be used for justification of radiation safety of new projects and for power increase of operating VVERs.

Peculiarities of nano and microfiber materials use for radioactive aerosols anaalysis

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For the first time the microfiber filtering materials FP were used for analysis of alpha active aerosols concentration when conduct of nuclear weapon test in 1949. While nuclear industry development the need appears for monitoring the radioactive aerosols referring to class of extra hazardous substances and formed when all nuclear facilities operation. The requirement is to measure qualitative and quantitative content of radioactive aerosols at all stages of production, including in the premises and vent systems, as well as perform monitoring of atmospheric aerosols around the enterprises.

Currently tens of analytical filtering materials are available; they are used for radioactive aerosols sampling. They are different by the structure of material, polymeric composition, method of production, in-service characteristics and, therefore, are intended for different goals and tasks. Provisionally, analytic nano- and microfiber materials can be divided by their purpose: analysis of beta radioactive particles, analysis of alpha radioactive particles, assessment of effectiveness of ventilation filtering systems and monitoring of atmospheric aerosols for radioactive particles presence. For fulfillment of below tasks the requirements imposed on the fiber materials are different in principle

Filtering materials on which sampling of aerosols are made for analysis of beta-radioactive particles were developed in 50-years of the past century. These are mostly materials with the average fiber diameter of 1,5 μ, intended for volume filtration with efficiency about 90 % at high speed of sampling. For analysis of alpha radioactive particles it is necessary to conduct surface filtration. For this purpose the modern nanofiber materials were developed with the average fiber diameter 80-150 nm and filtering efficiency about 99 % at high speed of sampling. Peculiar feature of such materials is that the whole aerosol sediment remains in the surface layer not more than 3 μ, which provides a high quality result when spectrometer analysis is conducted.

Analytical nano and microfiber materials are also used for assessment of filtering ventilation systems effectiveness. The main requirement for these materials is the effectiveness of aerosols filtration 99,95 % by most penetrating particles. Structure of such material is composed from composition of nano and electrostatically charged microfinber arranged in a definite sequence.

Materials used for the purposes of low level radionuclide monitoring of atmosphere and evaluation of characteristics of radioactive aerosol sources shall meet a number of requirements, whereto the following refer in the first turn: low temperature of polymer ashing in the form of ultrathin fiber and minimum carbon residue; high efficiency of aerosol particles sampling in the wide range of their sizes; high dust holding capacity; minor resistance to air flow.

Thus, it is not possible to run with only one universal fiber filtering material for analysis of radioactive aerosols. Specific features of application impose different requirements on the structure of analytical filtering materials and, consequently, on the production technology.

Development of hardware methodological complex for radioactive aerosol inhalation delivery monitoring

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Hardware methodological complex for monitoring of radioactive aerosol inhalation delivery, physical and chemical characteristics and assessment of internal exposures of personnel are presented in this report. The complex represents two-level system. Sampling systems are arranged on the first level containing the flowrate actuators and devices selective to radioactive substances dispersion.

As a result of a number of scientific research studies the ruler of impactors was developed in radiation hygienic research laboratory FMBC named by A.I. Burnazyan, which includes both stationary and moveable devices for radiation monitoring of air environment of industrial premises and sources of aerosol contamination and individual impactors, which are used in one set with personal flowrate actuators for aerosol particles sampling from the breathing zone of personnel.

Ruler of stationary impactors is headed by model АИП-2. The impactor is protected by RF patent, entered into State Register of measuring instruments and is included into the content of secondary reference standard of artificial radioactive aerosols ВЭТ-39 on the branch calibration base of JSC «SNIIP» of SC Rosatom.

Model ИРАМ-3-6К, reproducing specific features of impactor АИП-2 was supplemented by a set of filters АФА-3ДА, which enabled to expand the dimensional range of sampled aerosol particles up to 0,4 μ.

For prompt analysis of distribution of anticipated dose of internal radiation through organs and tissues of respiratory route an impactor-phantom was developed. This device simulates fractional deposition of aerosol particles in each of compartments of respiratory channel of human in accordance with dosimetry model of breathing system in Publication 66 МКРЗ.

Ruler of individual impactors includes its own licensed developments of laboratory: ИРАМ-4И, ИРАМ-2-4И, as well as foreign models: SKC Sioutas, Impact sampler. These impactors are used in one set with individual actuators of flowrate Leland Pump, Buck LP-20, УХ-100П (JSC «SNIIP» SC Rosatom).

Complex of calibration and analytical support is arranged on the second level including the equipment and certified techniques for conduct of lab analysis of sampled aerosols (radiometry, spectrometry, determination of type of chemical compound during inhalation), verification and calibration equipment for measuring instruments (digital flow meters, secondary reference standard of radioactive aerosols, special aerosol sources (SAS), as well as special software for automatic processing of obtained experimental data (AMAD Calculator 1.0, Star-CCM+).

Results of work presented in the report demonstrate a complex approach to the solution of one of the most complicated tasks of internal exposure dosimetry when inhalation delivery of radioactive aerosols inside the body.

Updating of rosgidromet radiation monitoring network within the frames of USCASRSM

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Since 2008 Federal Target program «Ensuring of nuclear and radiation safety for 2008 and for the period till 2015» (FTP «ENRS») has been implemented in Russian Federation. Within the frames of FTP «ENRS» the unified State computer aided system of radiation situation monitoring (USCASRSM) is set up on the territory of RF. RMS of Rosgidromet is the main base territorial subsystem of radiation monitoring as part of USCASRSM (BTSSRM) and shall ensure radwastes monitoring on the territory of the country.

To achieve the goals of USCASRSM it is necessary to settle a number of tasks, one of which – upgrading of already existing radiometric laboratories of Rosgidromet (as main structural elements of BTSSRM) and set up of new laboratories in the areas of large ROO location.

For solving the set tasks at the first stage (2009) the design approaches of upgrade program were developed that included: content of the laboratories to be upgraded; requirements for the list of measurements for radiation situation monitoring in the interest of USCASRSM; list of the content of typical equipment for radiometrical labs as part of BTSSRM (the following were developed: typical table of devices and equipment for radiometric groups (RMG) fitting, radiometrical labs (RML), regional radiometrical labs (RRML); list of equipment for upgrading of every lab based on the analysis of existing equipment of laboratories and requirements fro the equipment to be purchased.

Totally 40 radiometric labs work in the system of Rosgidromet.

It is supposed to fully upgrade (equipment was partially bought in 2009-201) 25 radiometric labs: in 19 UGMS (in 4 UGMS by 2 labs), Kaliningrad CGMS and NPO «Taifun», as well organize three new laboratories: in Bashkir and Far East UGMS, where till now sampling has only been made and specimens are sent to respective regional laboratories (Ural and Primorsky UGMS; and in Privolzhsky UGMS (in the area of Balakovo NPP).

The first stage of Rosgidromet radiometric labs upgrading has been completed (2009-2011), now the second stage is implemented (2012-2015).

As of beginning of 2014 the new equipment was supplied to 19 radiometric labs out of 16 UGMS and to laboratory of IPM NPO «Taifun»;

- one new laboratory was set up (RG LMZA in Balakovo in the vicinity of Balakovo NPP);
- 7 air-filtering plants of new generation were supplied as new ones (Balakovo, Kazan, Tomsk) so for replacement of old plants operating more than 40 years within 100-km zone of ROO;
- 2 vehicle laboratories of radiation survey were bought.

All equipment supplied within the frames of FTP «ENRS» in 2009-2013 was put into service (ALRR, VFU, gamma spectrometers).

In 2011-2013 the experts of NPO «Taifun» conducted inspections of the part of upgraded laboratories in Upper Volzhsky, Primorsky, Sakhalin, Kamchatka and Central Siberian UGMS.

The plan is to continue upgrade of radiometric laboratories as part of BTSSRM in 2016-2020 within the frames of new FTP « Ensuring of nuclear and radiation safety for 2016 - 2020 and for the period till 2025».

Possibilities to use updated regional weather forecast cosmo-ru system for ensuring prompt calculations of air contamination dissipation in atmosphere

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When forecasting the shift and dissipation of air contamination in the atmosphere ensuring of calculations with meteorological information becomes significant. It is a common practice to use numerical weather forecasting that are prepared by national weather centers.

Nowadays Gidrometcenter of Russia (RF GMC) issues prompt data of global forecasting with a rather gross spatiotemporal resolution (on the level ~ 140 km, vertically from~ 750 m and more, in time – 6 h). Improvement of spatiotemporal resolution of forecasting meteorological data produces a great influence on improvement of accuracy of contamination dissipation calculation.

For implementation of more detailed simulation of meteo- elements RF GMC joined the European Consortium for regional modeling of weather forecast COSMO. Complex COSMO-Ru provides spatial resolution in level ~7 km, vertically from ~20 m at ground and up to 400 m on level 3,5 km, in time – 3 h in ETP and Western Siberia. Trial operation of complex COSMO-Ru is carried out in RF GMC.

FIAC of Rosgidromet conducted the work on ensuring of the fast emergency response system ECASS NT of improved forecasting meteo-information received by system COSMO-Ru. To enable this the following was performed:

- development of procedures for drawing parameters of COSMO-Ru meteo forecast from system COSMO-Ru, coding and transfer from RF GMC to FIAC Rosgidromet;
- development of new calculation algorithms for atmosphere boundary layer parameters and upgrade of block «Meteoprocessor» of system RECASS NT on their basis;

- comparison of calculations of contamination fields when test releases of contaminating substances to atmosphere using forecasting data of system COSMO-Ru and global forecasting of RF GMC.

Calculations made by using forecasting data of system COSMO-Ru showed more accurate accounting of local peculiarities (impact of water surfaces, orography and temporal changes of synoptic situation) than by the global model.

Block «Meteoprocessor» using forecasting meteo data of complex COSMO-Ru is subject to trial run in system RECASS NT.

Problems and tritium and carbon-14 monitoring equipment.

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Tritium (H-3), by a number of reasons, takes an exclusive place in the issues of radiation safety assurance of NPP. Tritium generated at NPP is delivered, in contrast to other radiounuclides, to the environment escaping the cleaning barriers with the liquid discharges and with gas releases.

Moreover, the final regulation of C-14 monitoring has not yet been set up at the Russian NPPs, and, based on UNO NKDAR data, C-14 contributes to a half of the entire regional dose for critical group of public from radioactive releases at NPP. In nuclear reactors C-14 is mainly generated when interaction of heat neutrons with coolant materials, moderator, fuel, process media and admixtures.

Monitoring of tritium content in water media is rather easily implemented (because tritium spectrum is located in the area of low energies from 0 to 18,6 keV and stands apart from spectra of other beta- emitters) by way of water samples measurement using lab liquid-scintillation spectrometers.

The problem of tritium volumetric activity monitoring in the air however is only at the beginning of settlement at the Russian NPPs through method of portable sampling devices and lab analysis.

If monitoring of tritium and carbon -14 content in NPP releases is absolutely compulsory, the set of statistical data for determination of activity levels and their variations are needed in addition to the theoretical grounds for specifying the regulation of monitoring of these radionuclides in the rooms of constant attending by the personnel depending on the technological processes.

The need of continuous automatic monitoring of tritium activity in air in the mode of real time shall be stated as well with indication of the measured values of alarms if levels are exceeded.

Overview of updated world methods for tritium monitoring is also made in the report and the approaches to carbon-14 monitoring are reflected in the report. A ruler of equipment of the company Overhoff (USA) is presented, which has no analogues, from the viewpoint of organization of continuous automatic monitoring of tritium volumetric activity in air. The report also reflects separate models of equipment Premium Analyse (France), wherein the similar methods of tritium extraction and measurement are implemented.

Ecological monitroing of natural reservoirs used for service water supply to NPP before commissioning

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One of the most important types of activity arranged at NPP to ensure ecological safety and environmental protection is a complex ecological monitoring. Each NPP adhering to the regulatory documentation shall develop its own individual system of ecological monitoring, taking into account its location, local climatic, hydro geological and other conditions. Need of such system development is particularly important for Baltic NPP under construction with regard of its border location.

River Neman is stipulated by the design as a source of service water supply – water reservoir of the highest fish industrial territory, therefore, the goal of this study is establishment of ecological monitoring system of natural reservoir (river Neman) and conduct of observations before NPP commissioning.

It is proposed to use the ecological monitoring program of river Neman that includes the following aspects:

1. Refining of natural-climatic nature (topographic, hydro geological, geological, hydrological, aero-meteorological;
2. Determination of anthropogenic load of water reservoir at water collection (inhabited localities, production facilities, agricultural facilities, recreational loads);
3. Detection of factors that are stipulated by the global phenomena (radioactive and toxic releases, greenhouse effect);
4. Hydro biologic monitoring (phytoplankton, zooplankton, zoobenthos, periphyton, plant association);
5. Ichthyological monitoring (determination of species composition, migration of migrating and semimigrating species, spawning migration, reproduction characteristics; skate of young fish, investigations of spatial-temporal dynamic of ichthyocide);
6. Hydro chemical and bacteriologic studies (water and bottom sediments);
7. Saprobiotic study and determination of eutrophication of water reservoir.

With account that the main factor of NPP impact on natural reservoirs is the discharge of heated water into it, the program shall contain the forecast of thermal impact of NPP discharge water on natural reservoir, for example by method of hydrodynamic simulation.

Study of background parameters of river Neman before Baltic NPP commissioning will allow for:

- verify its natural characteristics and promptly apply the engineering and technical approaches and/or compensatory measures capable to minimize negative impact on water biologic resourses;
- determine the types of indicators for observations during operation.

Urgent issues of forecast and assessment of radiation impact of NPP on water rerervours

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The main task of ecological support of NPP investment projects is a justification of ecological safety of the plants being designed or upgraded. Based on Federal law "Environmental protection" in order for any NPP to be designed, placed, constructed and operated it would be necessary to prove its ecological and radiation safety at all stages of its "life" cycle. Unsolved issues regarding normative base and mathematical modeling of radioactive substances in water reservoirs are considered in the report. Solution of these issues will enable more efficient justification of ecological safety of designed NPPs.

After the Decree of Russian Federation №1069 issuance regarding criteria of solid, liquid and gaseous wastes to be referred to radioactive wastes, the need appeared of updating the existing normative base, which currently falls behind the contemporary requirements. When designing the complicated, from the technical view point, facilities and plants to which NPPs refer, the designers shall use the approved normative documentation. This is the reason why updating the normative documentation of agencies based on the Decree №1069 must be accelerated.

For calculation of permissible radioactive substances (RS) discharges to water reservoirs the existing methodology MUK 2.6.1.29-2000 became obsolete and the draft new methodology was not put into force. Moreover both these methodological documents also require modifications taking into account the Decree of Russian Federation Government №1069.

Justification of radio ecological safety of NPP regarding protection of surface water of land and sea water is based on assessments of comparison of actual and anticipated radio ecological situations, as under the normal operation conditions so in case of design basis or beyond design basis accidents. In some cases, when the NPP being designed or upgraded is located close enough to the border of neighboring countries, the design documentation shall be supplemented with a section, wherein assessment shall be given to the possible trans-boundary delivery of radioactive substances under normal operation and in emergencies. Different from water reservoirs, which are covered by the methodology of Rosgidromet «Methodology of forecast of the contamination state of water reservoirs when disturbance of normal operation of NPP», no alike methodologies exist for rivers an seas. Absence of necessary methodological base makes the designers search for simplified and too conservative calculation methods, which in some cases fail to reasonably prove the radio ecological safety to a needed degree of NPP being designed.

In a number of countries that use or plan to use nuclear power the regulations of radiation impact on separate groups of living organisms have already been put into force on the level of national norms. No ecological regulations exist in Russia so far. There is no consolidated position regarding the issue of such regulations development in the professional environment of scientists or specialists either. In the future the absence of ecological regulations of ionizing radiation impact on living organisms can have a negative effect on Russian nuclear energy technologies promotion to the foreign markets.

Soonest solution of the issues considered in the report will enable to enhance the quality of ecological follow up of NPP construction projects both in Russia and abroad.

Challenges and perspectives of arsms development in the areas of NPP location

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Based on 6.3 SP NPP-03 automatic radiation situation monitoring system (ARSMS) shall be provided off site by the design of NPP. The alike requirement is stated in i. 5.4.4 (OPB-88/97): the NPP design shall stipulate the continuous changes of ionizing radiation doses in protected zone and observation zone, wind velocity and other meteorological parameters, as well as periodical changes of radioactive release density for evaluation and forecasting of radiation situation in environment under normal operation of NPP and in case of disturbance in the NPP operation, including design-basis and beyond-design basis accidents. Information from NPP ARSMS shall be transferred to the unified State computer aided radiation situation monitoring system (i. 6.6.5 SP NPP-03).

For implementation of these requirements ARSMS runs in every region wherein the NPP of JSC «Concern Rosenergoatom» is located. The main task of ARSMS is a continuous monitoring of radiation situation for earlier detection of changes in radiation situation in the area of NPP location to render the information support of the organizations that make decisions about the measures of the public protection.

Experience of existing in JSC «Concern Rosenergoatom» ARSMSs use, as well as experience of establishment and operation of ARSMSs at NPPs abroad including operation of automatic radiation situation monitoring system after accident at Fukushima Daiichi (11.03.2011) enabled to detect a number challenge, as well the directions of further development of ARSMS including:

- justification of ARSMS configuration in the areas of NPP location;
- technical equipping;
- software and methodological support;
- information support.

Issue of NPP operation safety and issues of its equipment maintenance

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1. At the border of XX-XXI centuries when scientific technical ideas reached the highest point of their development, man-induced catastrophes and accidents with human victims and large materials loss periodically occur.

2. But the examples of safe and reliable equipment do exist. It sounds paradoxial but this is a nuclear weapon at the stage of its in-service handling. Therefore, it is possible to invent and operate the equipment with practically zero accident risk!

3. For solving the problem on NPP safety enhancement of vital importance are safety culture, team interactions in the course of inventing and operation of contemporary engineering systems, acknowledgement of safety problems as the top-priority matter in the human activity.

4. With the help of respective organization of equipment a very high safety degree can be reached thanks to goal-directed activity of human beings. «Philosophy» of equipment helps to enhance the responsibility of a human being in today's system «Man-machine».

5. Human factor shall address NPP safety enhancement and not conversely.

6. SRAs shall be conducted with material and intelligent expenses for safety improvement at NPPs.

7. When non-destructive test and diagnostics of NPP equipment metal one shall adhere the principle of INFORMATION REDUNDANCY and its state.

8. Examples of remote technical means are given:

- MDR-5 – manipulator for leaktightness monitoring even at the level of micro leaks for heat exchanging tubes of steam generators;
- MDR-3СЛ – manipulator for sealing of failed heat exchanging tubes of steam generators without use of polar crane in reactor hall.

Approaches to the solution of radwastes certification problem at russian NPPs

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Radiological danger of radwastes and, consequently, the methods of handling and disposal are defined by radionuclide composition and specific activities of separate radionuclides. Specific activities of radionuclides and category of radwastes by radiation factor shall be stated during their certification according to Russian and international practice in force (international term – characterization) – id est, statement of package or radwastes batch characteristics and their recording into passport.

In Russia like in majority of countries the responsibility for measurements and documenting of wastes characteristics is laid upon manufacturing plant, particularly on nuclear power plant (NPP). In accordance with Federal law № 190-ФЗ adopted on 11.07.2011 «Radwastes handling...» (hereinafter - FL) and Decree of RF government dated 19.10.2012 № 1069 «Criteria of solid, liquid and gaseous wastes to be referred to radioactive wastes ...», officially the characterization of radwastes shall include control of specific activities of 275 radionuclides. Bigger part of these radionuclides refer to so called «complicated for detection» radionuclides, direct measurement of which activity is related to cost consuming procedures of sampling, preparation and analysis of the sample (^{3}H, ^{14}C, ^{55}Fe, ^{59}Ni, ^{63}Ni, ^{90}Sr, ^{99}Tc, ^{129}I, row of trans-uranium radionuclides and others). Currently there are no methods or measuring instruments at the NPPs of concern Rosenergoatom required for performance of such comprehensive control of radionuclide composition and specific activities of radionuclides in radwastes. Procedure of radwastes certification at NPP is not unified, specific activities of only those radionuclides are recorded in the passports that are considered by this or that NPP to be worth of recording or can be recorded. Mostly these are easily detected gamma irradiating radionuclides with high percent of decay output and energy of gamma- quanta (^{60}Co, ^{137}Cs, ^{134}Cs). The situation evolved, according to FL statements, will with a high level of probability result in refusal of the national operator for radwastes handling from accepting radwastes from JSC «Concern Rosenergoatom» for burial.

To fulfill the requirements of regulatory and legislative documents and prevent refusal of national operator from accepting radwsates from NPP for burial it is advisable to justify and comprehensively agree upon the necessary and sufficient list of radionuclides under control in radwastesw of JSC «Concern Rosenergoatom». Availability of such list will enable to bring the process of radwastes characterization to measurement of specific activities of necessary and sufficient number of radionuclides. For comparison, the list of radiologically significant radionuclides recommended by IAEA for characterization of NPP radwastes (IAEA Nuclear Energy Series NW-T-1.18) contains 30 radionuclides.

Increase of efficiency and effectiveness of control of radionuclides complicated for detection from the content of the stated list can be achieved by way of stating the stable or conservative relations between specific activities of radionuclides that are commonly called radionuclide vector for various types of radwastes. Stating of radionuclide vector for every type of radwaste in accordance with international standard ISO 21238-2007 will enable to bring control of radionuclides in radwastes to measurement of specific activities for only separate easily detected radionuclides.

SECTION 2

NUCLEAR POWER ECONOMICS

Participation of jsc «Concern Rosenergoatom» in advanced models of wholesale market of electric energy and power

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The working model of the wholesale market of electric energy and power (WMEEP) with respect to purchase and sale of electric energy has been working since 2006 already and with respect to purchase and sale of power — since 2011 without significant changes. With respect to purchase and sale of electric energy the model enables formation the adequate price signals for suppliers and consumers. At the same time the acting market of power does not possess such properties guaranteeing return of investments to construction of generating facilities not on account of market mechanisms but on account of agreements on power submission, wherein the prices are stipulated by the State regulatory bodies.

Moreover, recently the forecast of electric energy consumption growth rates in RF is being revised and implementation of the mechanism of contracts on power submission eliminated the midrange threat of electric generation deficit.

Under these conditions the active discussion of the ways for reformation of the market of power needed for involvement of long-term investments into electric generating branch and progressive advance of generation is going on. Currently, the market society discusses three variants of reformation of the market of power.

Various alternatives of changes of the acting model of the power market and expert assessment of their possible consequences for both WMEEP on the whole and from the view point of NPP operation at the market and continuation of the program of the nuclear industry development will be presented in the report.

Estimated liabilities of raw and snf

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Grounds for acknowledgement of obligations

Economical nature and main issues at execution of works on assessment of the scope of obligations.

Industrial regulatory documents

Principles of formation of regulatory framework and its peculiarities at different levels of obligations life cycle management.

Stages of RAW and SNF handling

Principles of economic classification of stages and peculiarities of formation of the scope of obligations depending on the current stage.

Industrial reserves and estimated liabilities

Differences, specific features, creation and use principles.

Economical principles of acknowledgement of the scope of obligations

Change of value upon different grounds, adjustment, and principle of classifying as costs/profits.

Responsibility centers as per the stages of life cycle of obligations

Engineering calculation of physical values, analytical models, reflection in accounting and impact on financial and economic indicators of organization activities.

Criteria of effectiveness of investment to nuclear power engineering

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The effort is made to word and estimate the conditions under which nuclear power engineering can become «cost-efficient self-developing business». In other words, to identify conditions of stable growth of the set up capacity of NPP (from operating reactors) for the purpose to provide the national economy with the set up rate of electric energy generation without involvement of other financial recourses except for the incomes from the activity of nuclear industry itself. In other words, development on account of the own profit of the nuclear branch.

The study consists of two parts. Conditions of nuclear industry investment projects profitability are considered in the first part. For this purpose four criteria are used that take into account financial flow through the entire life cycle of NPP. [1, 2]. Analytical expressions of widely used UNIDO criteria of investment projects efficiency are given in the report on example of power industry (NPV, IRR and period of cost recovery), including efficiency of mountain projects, as well as IAEA criteria — capital value of electric energy defined as a tariff of break-even condition of the project. The obtained analytical expressions for criteria of power industry projects investment efficiency enable to solve not only «direct task» (calculation of criteria by the anticipated incomes and costs), but also «reverse task» — define the restrictions for incomes and costs, capital and running expenses, deadlines for construction and operation of the facility at desirable values of efficiency criteria.

The second part describes a so called «relaxation approach» of nuclear power engineering progress dynamics for assessment of possible rates of its development due to the own income and for detection of restrictions for capital and running expenses and incomes under the conditions of self-developing of the nuclear industry [1, 2]. Condition of self-developing of the nuclear industry is met when ratio of annual income to capital expenses for the new power unit exceeds the set up (directive preferably) rate of development (relative annual increment of growth) related to the period of the number of operating power units being twice increased. The more higher rate of nuclear industry development is needed the greater the ratio of annual income assigned to development (SREDA including) of the industry to capital costs for construction of new power units shall be.

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Cost estimation procedure for construciton of npp abroad at pre-implementation stage

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At present there are more than 20 VVER NPP power units in the portfolio of international orders, and even more are at the stage of case study. Part of these projects will be implemented or is already underway according to the plan «build, own, operate» (BOO). The issue of systematizing of activities on planning and cost estimation of NPP construction is becoming more relevant, more often it is required to make investment decisions on high risk level projects, to which first and foremost projects of NPP construction abroad are related to, under most tight schedule.

International NPP construction cost estimation procedure is designed to systematize approaches to assessment and optimize the process of making investment decisions at pre-implementation stage. The method shall be applied for assessment of NPP projects, in which Corporation or its affiliates are investors, at the stage of pre-investment and pre-design, when main project expenses have not been effected and the main decision has not been taken, and it can also be applied at the stage of implementation of construction projects for updating and optimizing of cost estimation and ways to reduce NPP construction period. Obtained pre-investment assessments can also be used with the purpose of monitoring of competitive strength of NPP projects of Russian design.

When assessing the cost it is suggested to use the following approaches to cost estimation of the project (cost engineering) with account for corresponding terms of estimation execution:

№	Name of approach	Character of approach	Terms of assessment
1	Express assessment	Based on analysis and extrapolation of specific parameters of earlier implemented projects.	Shortest (up to 1 day)
2	Title decomposition	Based on analysis of the cost of individual objects inside the title	
3	Decomposition of elements	Based on analysis of cost of individual elements of the objects inside the title	
4	Decomposition of resources	Based on analysis resource cost of the project/OCP	
5	Decomposition of resource estimate	Based on analysis resource cost LCE	Longest (up to half a year)

Example of using the method of «economic cross» in calculations on innovation nuclear fuel for nuclear power engineering

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The methodology for identification of economically effective distribution of value added received as a result of innovation technologies implementation among the participants of nuclear energy generation industry within the frames of the model «economic cross» is presented in the report. The procedure of use the calculation methods and discounting of financial flow by the parties is proposed on the example of economic relations of the suppliers and consumers of nuclear fuel with respect to problems of purchase of innovation types of nuclear fuel. Under discounting of financial flow the money price adjustment is meant at any time moment by way of dividing the sum of financial flow by the related coefficient of discounting. Discount parameters of financial indications have been defined at time diversity of costs on investments at various stages of energy resources life cycle. The formulae for calculation of efficiency of investment to nuclear fuel cycle haven been derived. Under the economic efficiency of investment the ratio of discounted final result to discounted expenses is meant. For determination of total efficiency all financial flows are summed up both on horizontality and on verticality of economic cross with amendment made for discount coefficient. At the same time the technology actual for every stage of market technology is defined according to the subsequent economic effect from the use of this technology and capital value of raw stock and materials. Estimates of effectiveness of the use of innovation types of nuclear fuel when delivery to traditional nuclear power facilities are demonstrated taking into account possible profit when their operation. Approach to estimates of innovations is described when intersection of life cycles of nuclear energy resources and nuclear power units. For this purpose the good economic effect received by all participants will be decreased by the discounted value of costs of all participants of the investment process. Here the portion of every participant is determined as the product of economic effect for portion of this participant in total discounted costs. While economic estimates of discount making, it is possible to use the rate recommended by Central Bank of Russian Federation. The alike indications are the most difficult for forecasting, however, this problem can be solved by means of use the hedging elements in the methodology. For the purpose to reduce the volatility of prices for nuclear raw stock and equipment, which must be bought and then used throughout the entire life time of the facility operation the proposal is made to deal with not specific prices for energy related to the present time and being discounted with the time but with the future prices of assets, which prices are strongly correlated with the investor interest. At this, to avoid the possible losses from the jumpiness of market the investor is proposed to hedge the investments by investing the goods, which price dynamic is reversely correlated with the indications that present interest for the investor.

One of the most important problems of the system of economic relations development as inside State Corporation «Rosatom», so on the example of assessment of one activity of foreign nuclear utility is the problem of objective assessment of fuel cost, particularly when the new type (design etc.) presents the economic advantages to the electric energy producer at NPP. It is a common practice that when price of fuel assembly is determined for VVER the method of «defining the price by the incomes» is used, in other words, based on the potential value of the sold product in the form of nuclear energy resource for counterpart. At the same time, in accordance with the concept of «economic cross», not only the real producer (particularly at the world market) but also the interim buyer of the fuel (claiming for its own portion of added value after the fuel treatment) can be considered to be specific «vendor» of nuclear fuel in a full production chain, and the «buyer» — a final consumer of the product (electric energy). Based on this approach two economic issues arise:

1. What is the portion the nuclear fuel occupies in the final cost of production (electric energy) and is it possible to change this portion after refusal from traditional type of fuel in favor of innovation type of nuclear power resources.

2. How to distribute the final added value of new FA type among the producers and consumers of the nuclear fuel taking into account time diversity of costs for payment of nuclear power resources and getting the profit on account of additional energy generation

Initially, these proportions were identified based on the economical practice of the entities of the economic cooperation even during the Soviet time, and some of their elements were additionally defined within the 90-th years of the past century. However, a certain compromise relation once stated did not have any ground in the form of elaborated calculation technique. The direct consequence of this becomes impossibility to define, based on the common practice, economically justified price for more efficient innovation types of nuclear fuel, which increases the risks both for their potential developers and for the producers and, on the whole, hinders the innovation development of atomic energy and competitive potential of the national economy on the whole. The set problem can be solved by the way of creating the model of «transparent» and economically justified distribution of added value received from the implementation of innovation process of advanced types of nuclear fuel implementation.

The proposed approach to formation of the model of conflict of interests and its solution can be used for preliminary evaluations, however, it sets a number of additional issues, which can be solved in future. In particular, when use of nuclear power units for the purpose of development of natural complexes the part of the natural rent can be assigned to reduction of the cost of generated energy for their cultivation.

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Tasks of ensuring human capital quality in the directions of nuclear power engineering activity: safety, efficiency and economic

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Tasks for human capital formation in nuclear power engineering activity are solved within the concept of culture safety and organizational production culture. It goes without saying, that a human factor is of the greatest significance in the sociotechnical system of the industry efficiency ensuring and in the selection of the optimal way of its economy development.

Human factor is determined by characteristics of human capital of the organization and represents the most valuable thing in the nuclear industry. Studies and many year experience of the nuclear industry psychological service prove: reliability, responsibility, intelligent potential, competency, motivation, systems of values and satisfaction by the work define the successfulness of the personnel and managers and, therefore, safety of production, its efficiency and economic reasonability. The important aspect is safety management in the real environment of plants and enterprises operation, including social, social-economic and social-political safety.

Under the environment of unstable world the ensuring of human capital quality to a great extent depends on the progress of external conditions, evaluation system of real situation and accurate forecast of forthcoming changes of social economic, military political and other outside conditions, threats and provocations and industrial risks. At the same time the work on formation of the manning reserves for ensuring development of nuclear engineering is as important as the breakthrough in the technical decisions. Very often the tasks of technical, technical process and manning levels and social policy are solved by various people and organizations without a due coordination.

Systematic approach to the work on scientific-methodological and organization practical safety ensuring and production efficiency of the industry supposes a planned, for 5–20 years, continuous and intensive work on formation of human resources, not occasionally done, «from grant to grant». This work on human resources management is to be done in the regions, in the areas of science and education, in related branches of the country economy. Such work requires interface with the work on formation, development and support of human capital at the plants and in the organizations for its management by the system of horizontal and vertical communications at all levels of the industry activity.

By setting and solving the tasks on ensuring the quality of human capital the policy is built for perfection of the system of work for personnel management, which implies a complex approach while ensuring the organizations of the industry with the qualified and psychologically trained personnel at all stages of the plant life cycle.

Main areas of works on enhancement of energy efficiency of npp

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At implementation of federal law №261 as of 23.11.09. «On energy saving and enhancement of energy efficiency...» fundamental difference of NPP from TPP, which use organic fuel, shall be accounted for in part of criteria of energy saving and energy efficiency.

Characteristics of NPP energy efficiency are: yearly supply of electric and thermal power. Economic criterion of increase of energy efficiency of operating NPP is the increase of yearly income.

Areas of enhancing of NPP energy efficiency are: increase of electric power production, decrease of auxiliary electric power consumption, increase of thermal power production, decrease of auxiliary thermal power consumption.

On the basis of investment pay-off period of 5 years and modern tariff of energy, the following expenses are economically sound:

- 33,8 mln.rub. and less at the increase to 1 MW of energy supply;
- 6,2 mln.rub. and less at the increase to 1 Gcal/h of thermal power supply on account of reduction of auxiliary expenses.

The most energy and economical efficiency can be gained at increase of electric energy production. Measures aimed at reduction of auxiliary power consumption can provide next lower grade of efficiency, and two grades lower and less when it comes to implementation of measures on gain in production and reduction of auxiliary thermal power consumption. Within the national experience, works on increase of electric power production were performed on the initiative of JSC «Concern Rosenergoatom»: installation of ball cleaning unit of condensers, modernization of LPC of turbine K-500-65/3000 and HPC of turbine K-1000-60/1500, modernization of MSR separators of almost all turbine units of NPPs.

When implementing measures on enhancement of NPP energy efficiency, it should be noted that efficiency of these measures shall be reflected in NPP reporting documentation.

Reporting documentation on technical-and-economic indexes of NPP is developed on the basis of readings of regular I&C, which have certain measurement error.

During development and implementation of «Program of energy saving and increase of energy efficiency» it should be noted that 90% of power and economic impact of NPP consists in the system of generation of not more than 9% in the system of auxiliary electric power consumption.

In order to design energy and economic efficiency of measures to receive confirmation, in reporting documentation of NPP it is required to specify the following measures:

- additionally mount measuring instruments of a higher accuracy class at NPP;
- develop and create automated system of TEI calculation at each NPP on the basis of application of mathematical models of turbine units and power units adapted to specific working conditions and readings of regular I&C.

Cost effectiveness analysis of works on long-term performance recovery of graphite stack of RBMK power units

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As a consequence of performance of set of research and development works during 2012–2013 technology of long-term performance recovery (LTPR) of RBMK reactor plant was developed and implemented. Implementation of the technology of maintenance works on the first power unit of Leningrad NPP enabled to perform unit connection to the grid and approach nominal power level.

Economically making decision on the set of works related to long-term performance recovery (LTPR) of graphite stack of RBMK power units shall be done based on criteria of cost effectiveness.

At that, it is practical to consider the following two extreme cases (scenarios):

- Scenario 1. Execution of LTPR works on all RBMK power units and continued operation of the whole park of NPPs with RBMK until decommissioning with account for life extension.
- Scenario 2. Refusal to perform LTPR works on all RBMK power units and continued operation of the whole park of NPPs with RBMK until decommissioning at the time of scheduled PC reaching marginal state.

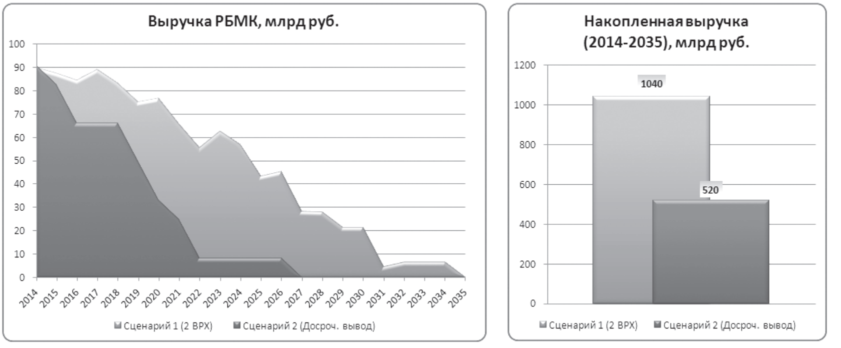
The case of single-time shutdown of all RBMK units has not been considered due banality of obtained technical-and-economic indices.

Estimation horizon of technical-and-economic indices of NPPs with RBMK covers the period of 2014–2035.

Estimation results reveal that for the considered time period of 2014-2035, refusal to perform LTPR works will result in non-produced energy to the extent of 401 bln kW·h (50 % from potential amount), which is equivalent to reduction of total accumulated revenue to 520 bln rub. (pic. 1).

Accumulated generated investment resources for 2014–2035, which are advisable to direct to construction of replacement capacities, will be 497 bln rub. according to scenario with LTPR, taking into account financing of LTPR works, and 263 bln. rub. – without LTPR.

All indices were calculated under condition of self-financing of LTPR works from depreciation payments of RBMK units.



Pic. 1. Yearly revenue of NPP with RBMK and cumulative result for 2014-2035.

Obtained results prove that performance of LTPR works for all power units with RBMK-1000 is economically efficient and is distinguished by significant (twice) excess of revenue and investment resources accumulated over the period of 2014-2035 in comparison with the scenario without LTPR and premature decommissioning.

Among negative consequences of early termination of power generation and decommissioning of RBMK reactors the following can be singled out:

- significant decrease of investment resources for financing construction of new NPPs, since almost half of electric energy sales revenue is generated by RBMK power units;
- necessity to find finances for conservation and decommissioning of almost all park of RBMK power units within next 5–10 years;
- decrease of economic security, increase of credit load (Debt/EBITDA) of «Concern Rosenergoatom» which will entail reputation risks and complicate attraction of additional credit funds;
- decrease of workload level of productive capacities of companies of nuclear fuel cycle, drop of revenue in the part of fabrication and enrichment.
- destabilization in electric energy price formation in the regions where NPPs are located;
- growth of social strain in the regions, because NPP operation provides high population employment;
- «electric power hunger» before commissioning of replacement capacities in the regions receiving power from NPPs.

NPP as part of russian model of electric power market

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This analytical research covers modern aspects of development of Russian model of electric power market, risks of electric power companies related to rules of market organization, mechanisms of market regulation, and rules of marketing of products. Brief analysis of the current market concept, possible trends and tendencies in development, which are to be taken into account by electric power companies, is given.

Systemic and market factors influencing economic indices of NPP functioning at electric power market have been noted (design aspects, security — economical situations in the grids with NPPs, commercial factors — complication of the model of the set of documents on offtake of electricity and power in the market).

Tasks requiring analysis from the point of view of economical substantiation of feasibility and economic effectiveness of new NPP projects operation under maneuverable modes with application of marginal analysis of break-even condition of NPP project (volumes, prices, sales patterns) were identified. Adjoint problems requiring accounting and substantiation at construction and operation at different operating modes of NPP power units of different power in RF grids were considered.

Application of simulation modeling for assessment of costs of decommissioning at designing of new npp power units

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Modern level of development of nuclear power engineering industry requires estimate of costs of decommissioning of designed NPP.

At present time costs are assessed in accordance with instructional guidelines, which are based on calculations as per lumped indices. Instructional guidelines are by nature phenomenological model with the simplest hierarchy level. Such models are over-aggregated, which does not allow to assess correlation of ultimate result (overall cost of DC) and the change of quality of execution of production processes (equipment demounting, decontamination of engineering constructions, etc.). Complication of hierarchic structure allows to specificate modeled processes and, thus, to find critical elements which allows to optimize financial expenses.

Concept of the model of simulation model for assessment of costs for decommissioning of modern NPP with VVER RP as per options, established RF regulations is presented in the report.

Suggested model, as well as many others, is phenomenological. But due to switchover to a higher hierarchical level, it allows to take into account changes of NPP decommissioning costs to the maximum at changes of technical means and technologies of works execution when modeling financial costs.

Suggested model is the new step in determination of amount of financing at the stage of new NPP designing. With a help of this model adjustment of accepted design and engineering decisions for new facility is possible.

Calculation methodology of production costs and sale prices for power engineering products from the point of power balance

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The proposal is made to perform evaluation of competitiveness and efficiency of innovation and investment projects of power facilities using the equivalents of production costs and sale prices combining production costs and sale prices for separate products of energy: electric energy, electric capacity and thermal energy for industrial heating.

Further break up of equivalents of production costs and sale prices for production costs and sale prices of separate products is proposed to perform on the basis of correlation of power balance of power generating facilities. Description of the above correlations is given.

Calculation algorithm of production costs and sale prices for the products of power industry is also presented.

As a conclusion, the example of calculation of production costs and sale prices for the products of power industry with the random selected parameters is represented below.

Method of calculation of cost prices and wholesale tariffs for power plant products based on balance of energy

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It is offered to execute estimations of competitiveness and efficiency of innovative and investment projects of power plants with use equivalent the cost price and the wholesale tariff, which are uniting cost prices and wholesale tariffs for separate products of power plant: the electricity, electric power and thermal energy for central heating.

The subsequent decomposition of equivalent the cost price and the wholesale tariff on cost prices and wholesale tariffs of separate products offer to execute on the basis of formulae of balance of energy for power plants. The description of the mentioned formulae is presented.

The algorithm of calculation of cost prices and wholesale tariffs for of power plant products is presented also.

There is presented also the example of calculation of cost prices and wholesale tariffs for products of power plant with arbitrary parameters.

Report «Optimization of logistics system of nuclear power plants»

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1. Optimization of storage facilities

- Organizational decisions and process-oriented approach
- Key performance indices
- Warehouse based on principles of «Thrifty production»
- Main results and risks of optimization of storage facilities at its implementation

2. Optimization of inventory reserves

- Cost
- Key performance indices
- Reserves management models
- Main results and risks of optimization of inventory reserves at its implementation

3. Optimization of inventories delivery to departments

- Distribution of responsibility areas
- Key performance indices
- Models of delivery management
- Main results and risks of optimization of delivery at its implementation

Impact of npp emergency indications on the price of electric power under the conditions of worldwide insurance pool

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For the whole time of nuclear power industry existence the operation experience of power reactors was accumulated, which amounts to about 15000 «reactor-year» (this is the sum of operation time of all reactors in the world at this specific time period). The large accidents happened during this time resulted in full loss of six power units of NPP: 2-nd power unit of NPP «Three Mile Island» (USA, March 28, 197., losses about 1 bln USD); 4-th power unit of Chernobyl NPP (USSR, April 26,1986, losses about 25 bln. USD); four power units at Fukushima-Daiichi (Japan, March 11, 2011, losses about 200 bln USD). In other words, the average frequency of severe accident is about $6/15000=4\cdot10^{-4}$ 1/yera [1, 2].

Accidents occurred at the NPPs in three countries, but their consequences, anyhow, produced the impact on the development of nuclear power industry in all countries, on formation of internal safety standards and on upgrading the structures of reactors in all countries – manufactures of NPPs. Therefore, it is advisable to implement a worldwide pool of NPP insurance. Exactly of «worldwide pool», since any accident wherever it occurs produces an effect on the technological progress in all countries generating electrical power.

Economic losses from NPP accident can be compensated if the respective insurance funds are established beforehand by way of including insurance payments to the electric energy cost; the value of this payment shall be such as the sum of money equal to the expected loss would be saved during the supposed time between two possible accidents. Our estimates show that the relative increase of the capital value of electric energy with account of insurance payments would be directly proportional to possible frequency of accidents and resulting loss (loss factor). Since the ratio of running expenses to capital expenses at NPP hardly depends on the reactor capacity, relative increase of electric energy cost when insurance allocations is approximately equal for all power reactors.

If we assume that average frequency of severe accident is $\leqslant 4\cdot10^{-4}$ (1/year) and effective rate of depreciation is 10 %/year, a relative increase of the capital value of electric energy for every NPP worldwide for insurance of accidents with various damages will make up:

⩽0.6 % for accidents of Three Mile Island type;

⩽4.5 % for Chernobyl NPP accident type;

⩽ 12 % for Fukushima-Daiichi accident type.

In other words, economic efficiency of the NPP is saved to a great extent, if all NPPs worldwide pay insurance contributions to a worldwide insurance pool. In such a case even such accidents as Fukushima will not drastically deteriorate the competitiveness of NNPs.

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Simulation model for assessment of costs of decommissioning of npp pwr-toi power unit

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Simulation model, which estimates combined costs for decommissioning of NPP VVER-TOI power unit according to the option «Abandonment with immediate dismantlement» based on determination and summing up of discrete cost estimates related to execution of required works and process operations was developed.

In the structural diagram of developed model the number of facilities, types of works and process operations performed at these facilities is determined with account for layout and design peculiarities of the power unit.

Model functioning is carried out in accordance with prescribed event schedule, which reflects the list and sequence of execution of works and process operations at each facility.

For the cost estimate of each type of works or process operations performed at facilities of the model special formulaic mechanism was developed.

Initial data for the model are:

- mass and dimensions parameters of equipment, systems and engineering structures of power unit under VVER-TOI technology;
- radiation properties of equipment, systems and facilities of power unit of VVER-TOI design;
- specific technical-and-economic indices of execution of process operations including RAW handling and other required data.

The model allows to assess costs of execution of works both at each object of the model and the power unit on the whole, including the possibility to assess required number of containers with radioactive wastes of different activity categories, which are handed over to national operator.

This model can also be used for performance of comparative cost analysis of different layout, design and technical decisions on decommissioning taken during designing of NPP power units of VVER technology.

impact of natural uranium mining progress on nuclear power engineering development strategy

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Analysis of today's state of providing the world nuclear power industry with the stock of natural uranium is presented. Results of this analysis state the deficit of this type of non-renewable resource for long-term development of the industry under the existing types of nuclear reactors. Economic value of uranium resources under IAEA classification is identified by a probable production cost of the products. Until 2009 three price categories of Uranium were used: <40, <80 and <130 USD./kg U. Since 2009 a new category was introduced <260 USD/kg [1]. Resources of uranium of price category <40 USD/kg can be considered exhausted. Based on the information of World-wide nuclear association, the mined resources of uranium make up 5.3 MT at price less than 130 USD./kg and 7.1 MT at price less than 260 USD/kg.

Analytical «relaxation model» is presented for forecast of trends and peaks of natural uranium mining on the basis of its available stocks and actual rates, levels and mining technology [2]. Calculation results of uranium mining progress till the end of century XXI are given according to the proposed model in Canada, Australia, Kazakhstan, Russia, USA and all over the world. Calculations show that the peak of uranium mining in these countries can be achieved within the period of 2015–2040. In the second half of century XXI the mining of uranium from the operating minefields in the world under the existing technologies will be considerably decreased, and natural uranium stocks by production cost less than 130 USD per a kilo will be practically exhausted. In other words, economically acceptable stocks of natural uranium when its inefficient use in existing thermal reactors are small. They do not exceed the stocks of oil in an energy equivalent and are not equally accessible for different countries.

Fast breeder reactors (breeders) shall become the basis for the long-term development of nuclear power engineering, which enable 100 times increase of the resource base of nuclear power engineering and more on account of including of uranium-238 and thorium-232 into fuel cycle of isotopes. Commercially acceptable and safe fast breeder reactors shall be widely involved in power industry not later than the middle of the century, insomuch as natural uranium is sufficient for their commissioning.

The proposed model is convenient for annual monitoring and improvement of uranium mining progress (and other non renewable resources) taking into account new data on input parameters (recent values of rates of change and level of annual mining, specification of data on uranium reserves at the beginning of forecast period).

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2. V.V. Kharitonov, A.V. Kryanev, U.N. Kurelchuck, N.Yu. Dudin. Analytical forecast of uranium mining progress. Economic strategies, 2013, № 3, с. 58–72.

Section 3

NUCLEAR POWER DEVELOPMENT

Subsection 3.1
DEVELOPMENT OF NEW NUCLEAR POWER UNITS

Optimization of approaches for NPP unit commissioning

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The report will contain description of optimization of approaches for NPP unit commissioning based on target-oriented planning. The complex task of NPP unit commissioning is fragmented into finite set of targets. Individual schedule is developed for each target. Schedules as above are interlinked to make up the unified integrated network schedule of pre-commissioning, Level 4. The report will describe methods of network scheduling.

Schedule as planning and control tool. Schedule-assisted control of implementation of target tasks. Civil-and-erection priority assignment for timely implementation of target tasks. Timely detection of would-be problems, affecting NPP commissioning schedule observance. Prompt development of compensating measures for successful timely settling of arising troubles. Target attainment (NPP commissioning in line with time schedule) by successful systematic settling of target tasks.

Optimization of RP BN-800 equipment installation processes

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The following basic approaches for work arrangement are traditionally applied in course of construction of NPP units, so that to shorten construction period and optimize cost recovery; such approaches are also applied for equipment design and power unit construction planning:

A. Ready-to-use equipment (Reactor pressure vessel, particular equipment items) manufacture and supply to the site.

Construction of NPP unit buildings concurrently with equipment manufacture.

B. To optimize civil-and-erection period, concurrent installation procedure is applied, with the following operations envisaged:

- installation of shop-made ready-to-use equipment in design-envisaged position with the use of construction crane, prior to completion of premise floors, or via temporary erection openings, left in course of particular building construction;
- equipment modular installation in design-envisaged position, upon completion of all the civil, lining and finishing works in particular rooms, and preparation thereof for "clean" installation of equipment.

Concurrent installation

Concurrent installation ensures the following possibilities:

- equipment installation period inclusion into premise construction period;
- equipment installation period shortening by assembly and installation amount decreasing;
- equipment quality improvement by reduction of number of field welded joints.

Eventually, possibilities as above enable to decrease power unit construction cost.

Modular installation

Construction of premises shall be preceded with the erection of Reactor assembly building (RAB) at the site; building as above is intended for field assembly (re-working) of equipment components (assemblies), supplied from manufacturing plant, to get mounting blocks. In the RAB, supplied modules are assembled to get mounting blocks, with subsequent transportation to installation area. The mounting blocks are installed in premise (well) with the use of heavy-duty cranes (of "DEMAG" type). Final assembly of mounting blocks is taken up in power unit premises up to the unit construction completion.

Used as an example is BN-800 Reactor pressure vessel installation sequence.

Reactor pressure vessel re-working and installation, concurrent with the building construction, made it possible to shorten power unit construction period to 1.5 years.

Approximate saving rate made 420 mln. roubles.

Designing of new-generation units for medium-power NPPs

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According to IAEA classification, medium-power NPPs (MP NPP) shall be understood as NPPs with power units of installed capacity 300 to 700 MW.

Possible applications for MP NPP are listed below:

- in areas, where the use of conventional fossil fuel is impossible or impeded, with no power lines or output capacity of power-generating facilities restricted due to design features of available power lines;

- as secure non-interruptible power source for state-significance objects;
- for seawater desalination, which looks a currently important issue, in view of potentially critical situation concerning fresh water availability worldwide.

The MP NPPs are advantageous for lower project costs and risk cushioning for investors.

Despite a good amount of preliminary studies in the field, such issues as elaboration of MP NPP unit commercial prototypes and timely development of international market thereof, have not received due attention in Russia so far.

The results of MP NPP market research and outlook infer the possibility of Russia's participation in construction of such NPPs in Azerbaijan, Armenia, Kazakhstan, Malaysia, Qatar, Mongolia and other countries.

Currently in progress is design-preceding engineering study of two options for MP NPPs, taken up by JSC "NIAEP", viz.: on the basis of reactor plant (RP) VVER-600 and on the basis of RP VBER-600.

Particular features of NPPs with VVER-600 are listed below:

- Use of serial equipment with high performance ratings, corroborated by operation experience, for the main systems.
- Proven practices of manufacture, construction, installation, pre-commissioning, operation, repair and maintenance.
- No necessity of basic R&D.
- Prompt development of Working and Design documents.

Particular features of NPPs with VBER-600:

- Use of matured shipboard technologies.
- Use of custom equipment with presumably high performance ratings.
- Use of equipment modular manufacture technique.
- Development of Working and Design documents upon completion of key R&D efforts, design-basis justification and development of RP Technical design.

MP NPP options, presented above, are rather close in terms of technical-and-economic indices. Each option implies the use of innovative technologies, in combination with time-proven conventional approaches.

Development of turbine plants for VVER-TOI NPPs with the use of equipment supplied by JSC "Power machines"

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The report contains detailed description of the design of steam turbines and other equipment, proposed by "Power Machines" Co. for Russian and international NPP designs, including those on the basis of VVER. Available in the report is brief review of 1000 MW turbine plants (TPs), successfully operated at Russian and foreign NPPs, as well as 1200 MW turbine plants, already manufactured and supplied to "Novovoronezh" (units 1, 2), "Leningrad" (units 1, 2) NPPs, and turbine plants under manufacture for "Belarus" NPP. In addition to detailed review of up-to-date fast-speed turbine of the company's make, the report presents TP auxiliary equipment supplied by the company.

The report contains a section, dwelling on the latest developments of turbine plants for VVER TOI NPP turbine halls. Given here are layout solutions and comparison of particular performance ratings of various modifications of TP.

The report addresses the company's experience and successive development, allowing for market trends and the highest demands for Turbine hall equipment reliability and efficiency, presented in the recent NPP designs.

Technical diagnosis as control element of NPP life cycle

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This report describes the concept suggested by author – "Technical diagnosis as control element of NPP life cycle". Creation and implementation of this concept meets the requirements of "Federal law about nuclear energy use" of the Russian Federation [1], meets goals and tasks of diagnosis development, requirements of other normative documents, rules, including order documents and decisions of SC "Rosatom" and operating documentation of OAO "Concern Rosenergoatom" on ensuring safety, quality and efficiency of operating nuclear energy objects.

Extended design operating life of new NPP energy units (60 and more years) requires more representative supporting base of ensuring conditions of safe operation. There is a complex of IAEA recommendations on ensuring a long term operation [2, 3], which can effectively be taken into account while commissioning of new NPP energy units.

The effect of applying diagnosis facilities for new NPPs at all stages of LCM is ensuring continuous operation, transition to maintenance judging by the technical condition and finally to cost saving of holding basic funds of operating organization and receiving maximum profit.

The basis of the concept is founded on requirement of operating organization in receiving filled out Digital Dossier at every stage of LCM from design, construction, installation and alignment organizations, specifically:

1. Digital Dossier "As Designed"
2. Digital Dossier "As Built";
3. Digital Dossier "As Commissioned";

4. Digital Dossier “As Operated”;
5. Digital Dossier “As Is”;
6. Digital Dossier “As Decommissioned”.

Highlights of diagnostic support for valves and rotating machinery at NPP lifecycle operation-preceding stages

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Formation of advanced trend of science, like Product digital data-based model (PDDM) [1], imposes certain requirements for diagnostic model information contents and data exchange format. First of all, it implies standardization and unification, with all the agreements on formats and rules of data exchange and storage. Moreover, information entered into the PDDM of NPP technical arrangement, shall contain all the required data from each stage of the lifecycle, from design to de-commissioning. As early as now, such format determination and development of document, specifying or regulating the PDDM information content, can be treated as “hot” issue.

The PDDM development will entail the creation of power unit digital data-based model, unified information space, development of rules of the data-based model handling, standards, general principles and schedules of work. Diagnostic constituent of both PDDM and power unit digital data-based model shall be treated as background information. For NPP unit lifecycle management, digital models of products, containing diagnostic constituent, can and shall be used as key tool intended to control elimination of non-conformities, moving on from stage to stage.

According to the results of analysis of statistic data on non-conformities in lifecycle segment “Motor-operated valves adjustment and operation” for “Rostov” NPP (units 2 and 3) pre-commissioning, positive experience in combination of electric drive adjustment with operability diagnostics for electric parameters of motors in online mode makes it possible to prevent 80 % of faults, provoked by defects unrevealed heretofore. It is noteworthy that the use of mobile diagnostic tools is conductive for pre-commissioning period shortening and quality improvement, as diagnostication confirms operable state in case there are no defects, and enables the detection of hidden (shop) defects, installation defects and design errors. And, vice versa, late diagnostication of motor-operated valves is fraught with the process protraction owing to time gaps between adjustments of valves, whereas proper conditions for re-idling of valves for diagnostic purpose is observed rather scarcely.

Shaft adjustment in course of rotor equipment installation, with subsequent control with the use of advanced diagnostic tools, control of electric motor foot-to-frame contact spot, foundation vibration are conductive for pre-commissioning period shortening and quality improvement.

JSC "Atometekhenergo" efforts, jointly taken with JSC "Concern Rosenergoatom" branch offices, partly facilitate PCW diagnostic support, but they are insufficient to completely settle the issue. Owing to the assigned limit for pre-commissioning costs, it is impossible to go beyond the consolidated cost estimate for power unit construction, expense item "Start-up costs". As of the date, no special-purpose funding for pre-commissioning diagnostic support is envisaged.

All the facts, stated above, offer special-purpose funding for NPP equipment diagnostic support. Such funding shall embrace, apart from PCW and NPP commissioning, even earlier stages of equipment lifecycle, i.e. design, manufacture (including acceptance and handover testing), incoming installation-preceding inspection at NPP Site.

Implementation of enacted document "Branch measures for improvement of safety, quality of construction of commissioning of new units of NPPs...", calls the discussion of issues pertaining to JSC "Atometekhenergo" participation in the following activities:

- equipment incoming inspection;
- shop acceptance testing of equipment prototypes.

Support of design service life of NPP equipment and pipelines in long-term operation conditions

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Support of long-term operation of new power units of NPPs throughout the assigned service life calls for integrated approach and implementation thereof as early as at "pre-operation" stage: technical and detailed designing, equipment manufacture, civil-and-erection works, pre-commissioning and commissioning.

Currently applicable Lifecycle management (LCM) concept implies the use of systems, engineering technologies and tools for status diagnostics of components (heat mechanical equipment and pipelines) at all the stages of lifecycle. That is why the design "point-wise" updating, intended to incorporate the LCM aspects, is required for NPP units of "AES-2006" series; meant here the units under design and even under construction.

Prompt updating is required for currently applicable Regulatory documents of the level of Federal standards and rules, applied in nuclear power engineering for about 25 years.

State Corporation "Rosatom" concept in technical regulation field, voiced at ATOMEX-2013 Conference, harmoniously incorporates requirements, set forth in applicable Federal Laws, and envisages the development of technical regulation system with participation of the "Rostekhnadzor", in four key areas as follows: formulation of mandatory requirements, standardization, assessment of non-conformances, and accrediting in the field of nuclear power use.

The system for metal state field monitoring is one of the key constituents of long-term operation support. It implies the use of base metal and welded joints advanced NDT practices, generation of items (equipment) data-based models upon incoming and pre-operation inspection results, starting from manufacture in shop conditions, and power unit digital dossier, proceeding from the results of field, pre-commissioning, and commissioning tests.

Pre-operation inspection accuracy, application of diagnostic tools, and safety analyses results (including strength and structural integrity assessment) will make it possible to differentiate field non-destructive testing (FNDT) amount and regularity; the same will enable to take up critical equipment repair and maintenance depending of actual state thereof.

Ensuring safe operation of pipelines and NPP unit equipment, potentially approved by ECW

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Damages of pipelines and equipment, produced from low-carbon steel under the influence of coolant flow on the mechanism of erosion and corrosion wear (ECW) are typical for many nuclear power plants both in Russia and abroad.

Considering the multifactor degree of damages in the mechanism of ECW base metal and welded joints of heat and mechanical equipment (HME) elements and plant pipelines require coordinated efforts in the following directions in order to ensure their safe operation:

- specification of actual load specter (and detection of possible hydraulic impacts, beyond design basis installation efforts etc.);
- implementation of effective means and methods of nondestructive testing in areas with nonequidistant external and internal surfaces,

characteristic of local damages in the mechanism of ECW, and welded joint areas;
- monitoring epy content of such chemical elements as chromium, copper, aluminum in the base metal and root zone of welded joints effecting on ECW rate;
- registration of actual history of maintaining water chemistry for considered HME units and pipelines;
- applying means of conservative forecasting of possible ECW rates in elements under investigation considering the above mentioned operation specifications.

Positive experience of Lovisa NPP deserves a detailed analyzing and considering during the development and implementation of activity package on control and monitoring of situation with damages in ECW mechanism on Russian NPPs.

For new nuclear power units it is necessary to perform a detailed control of initial technical condition of HME and pipelines potentially confirmed on damages in ECW mechanism already at the pre-operational stage with recording all results of control in diagnostic passports.

Differentiated schedule of inspections at the operation stage both for existing and new NPP units can be developed on the basis of normative methods of ensuring construction integrity.

Selection of NPP equipment seismic stability verification strategy

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NPP equipment seismic stability assessment (qualification) can be performed in different ways: calculation, experiment-calculated, or indirect methods, based on earlier obtained results.

All the methods as above include procedures of acquisition and review of equipment Design documents, installation drawings, datasheets, earlier performed seismic stability analyses and tests in lab conditions. Moreover, all the methods of seismic qualification include the procedure of equipment visual inspection for compliance with design requirements for fastening and routing of adjacent pipelines, with concurrent assessment of actual state of fastening members, and possibility of non-incorporated interaction with other equipment items or civil structures.

For seismic-resistant equipment with test reports available, installed on rigid foundations and free from external mechanical connections, conclusion on seismic stability in the field is drawn up provided that design-envisaged requirements for fastening are met. For equipment with external mechanical connections and piping, installed on intermediate metal structures, additional experiment-calculated seismic stability assessment shall be done, as it is next to impossible to credibly assess the effect of interconnections on dynamic parameters thereof by calculation only. Design models are clarified by experiment-based determination of intrinsic dynamic parameters of equipment in actual installation conditions thereof.

As alternative to such a laborious procedure of seismic resistance determination, the method of equipment screening-out as per GIP procedure has been developed in the USA. The GIP procedure includes comparison of boundary spectra with actual spectra of response to seismic impacts. Boundary acceleration spectra, under which equipment retains seismic stability, are obtained proceeding from equipment bench test results generalization, or results of actual earthquake at power facilities. The indispensable condition for equipment seismic qualification with the use of such method is enveloping (excess) of required spectrum, plotted for 5% damping, by the boundary spectrum. However, as shown in the figure, the boundary spectra are rather low, and, upon comparison thereof with, for instance, with floor response spectra for NV NPP-2, it is clearly seen that the GIP boundary spectra are appreciably lower than required spectra within entire frequency range. Hence, the GIP procedure is inacceptable and, in fact, there is no alternative to experiment-calculated assessment of equipment seismic stability in actual conditions observed at NPP units.

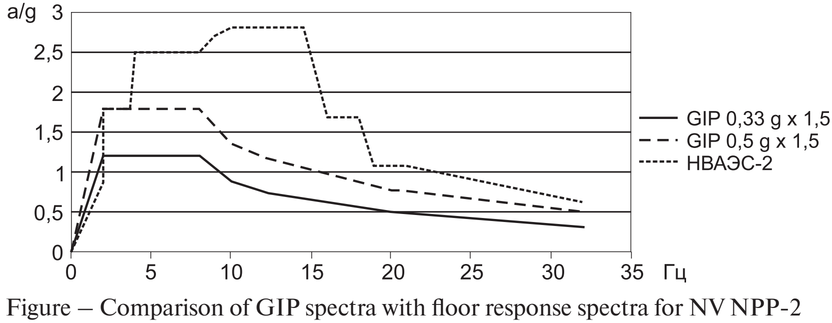

Figure – Comparison of GIP spectra with floor response spectra for NV NPP-2

Core prestress analysis of Rostov NPP unit 3

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Active core prestress of Rostov NPP Unit 3 was performed using the active core prestress system.

During works related to the core prestress of Rostov NPP Unit 3 measures on improving methods of work performance were taken considering results of similar works performed at Kalinin NPP Unit 4. Taking into consideration reasonably large spread of effort losses while performing wire cable anchorage at Kalinin NPP, this process was paid a special attention to. Sufficient number of PSI-01 sensors installed on the containment of Rostov NPP Unit 3 allowed monitoring not only tension of straining end of wire cable anchorage and evaluate tension stabilization time along the length of the wire cable after its anchorage.

Design value of loads in cables during prestressing (of the jack) makes 9,0 MN (918 tf). Considering anchorage prestress losses, design loads of armored cables equal to: 8,7 MN (887,15 tf) – in the cylindrical part of the containment; 8,58 MN (874,92 tf) – in the dome part of the containment.

After prestressing of the first nine armored cables in the containment cylinder anchorage loads approximately amounted to 849,27 tf, which is 4,3 % lower than design values.

Accordingly, in order to achieve the design value of armored cable tension it was suggested to increase the jack load hold time before the load transfer from the jack to anchoring from 5 to 15 minutes. Received results show that before the load transfer to the anchor block, jack loads varied, and anchorage loads greatly increased.

After that it was suggested to increase pressure value up to the design value during the hold time (15 minutes) every 5 minutes. Prestress analysis, which was performed considering the above mentioned, showed that before the load transfer from jack to anchorage loads were higher and were more stable, which is very important to define force distribution along the whole length of the armored cable. Anchorage loads were close to the design value.

In this context IBRAE RAN proposed to change the technology of armored cable tensioning related to the active core prestress system of Rostov NPP Unit 3.

As a result, after the load transfer from jack to anchor average tensions of the containment cylinder cables reached 878,67 tf. Average loads in armored cables of the containment dome after the load transfer from jack to anchor reached 879,59 tf.

Thus, given results show that the armored cable tensioning method proposed by IBRAE RAN allowed achieving the design level of Rostov NPP Unit 3 containment prestress.

Construction materials for power unit with BN-1200 reactor plant

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The principal role in the structure of the future large-scale nuclear power industry is assigned to fast breeder reactors. They allow improving the efficiency of using the natural uranium and thus lift nuclear industry development restrictions considering natural resources of nuclear fuel. Since the beginning of this trend in Russia in the middle of the previous century, FSUE TSNII KM "Prometey" has been developing basic and welding materials, including technologies required for production of fast sodium reactors.

Currently a new industrial power unit with BN-1200 reactor plant is being developed with high demands to increasing safety level, decreasing of specific metal content (as compared to BN-800 reactor plant), extending the life cycle of RP up to 60 years, and that one of steam generator up to 30 years. Performance of these requirements is impossible without using improved construction materials, works on creating which are performed by FSUE TSNII KM "Prometey" beginning form 2007 in two directions.

Works on adjusting chemical composition of 10Х18Н9 (09Х18Н9) and 08Х16Н11М3 steel grades related to changing the carbon content and additional introduction of nitrogen into steel are performed for the reactor vessel and vessel internals, sodium pipelines. Preliminary data of experimental researches show enhancement of steel creep resistance of tested compositions, and corrosion resistance required to decrease the probability of corrosion damaging of materials during the pre-operational (installation) period.

A new creep and corrosion-resisting 07Х12НМФБ steel grade of martensite-ferrite class was developed for the steam generator pipe system and internals. This steel grade has a sufficiently improved resistance to pit corrosion in the tertiary circuit, than currently used 10Х2М steel grade. Preliminary experimental data bear evidence of that the developed material completely satisfies requirements of the steam generator designer on short-term and long-term mechanical properties.

07Х12НМФБ steel grade has passed industrial implementation in the wide range of metal semi-finished part sizes. Technology of producing seamless cold-worked pipes from 07Х12НМФБ steel grade, diam. 16 mm with the wall thickness 2 mm and length reaching 25 m, has been developed and implemented.

Considering 07H12NMFB steel grade, welding wire of Sv-10H12NMFT grade and rods of EM-99 grade have been developed on its basis and implemented in the industry; technology of automatic welding under the flux layer for manual and arc welding using welding rods 40 mm wide, including technology of welding heat exchange pipes to tube sheets for models using multiple-pass automatic argon-arc welding with filler metal wire has been developed and industrially tested.

Materials engineering aspects of VVER-type power installations operation safety enhancement based on reactor pressure vessel steel quality improvement

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One of materials engineering aspects of safety enhancement of VVER-type advanced reactors is improvement of Reactor pressure vessel (RPV) resistance to brittle failure to the level of impossibility of catastrophic failure in whatever nominal and emergency conditions. In this case, potential failure of materials in course of operation as part of RPV is only possible as per ductile fracture pattern, requiring a very high energy for crack propagation in RPV. Condition as above shall be ensured as a result of both RPV design improvement and due to improvement of performance ratings of RPV steel, allowing for embrittlement under effects of neutron irradiation and operation temperature.

The presented report specifies requirements, related to VVER-1200 and VVER-TOI RPV operation safety enhancement, with two safety levels presented.

Safe operation management condition, Level 1, implies RPV material state in which its failure as per ductile fracture pattern is only possible, i.e. RPV brittle failure and, consequently, catastrophic failure thereof, is ruled out in principle.

Safe operation management condition, Level 2, as well implies the possibility of material failure as per ductile pattern only in case of presence of steam-water mixture in RPV during emergency cooldown thereof. Crack propagation as per ductile pattern is only possible with reactor pressure, and crack may propagate only under effect of self-balanced thermal stresses. In such conditions, crack propagation cannot entail the RPV perforation damage. Yet, extended cracking is not ruled, which fact is not conductive for RPV subsequent operation.

The applicability of 15H2NMFA Cl. 1 and 15H2MFA-A (mod. A and Mod. B) steel grades were analyzed for applicability for enhanced safety RPV manufacture. 15H2NMFA-A Cl. 1 steel ability to ensure only the 2nd level of RPV safety has been demonstrated.

RPV safe operation of Level 1 can be ensured by the use of new-generation steel grades (15H2MFA-A mod. A and 15H2MFA-a mod. B) on the basis of alloying composition of 15H2MFA steel.

Risk management in course of transfer to NPP shop-free structure

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1. Preconditions for transfer of Russian NPPs to shop-free structure. Analysis of best practices and foreign experience.

Brief information on the analysis of positive aspects and challenges of transfer to shop-free structure proceeding from Russian and foreign experience in implementation of shop-free structures at power engineering facilities.

2. Organizational changes risk management procedure allowing for IAEA recommendations.

List and purpose of documents pertaining to methodological support of organizational changes risk management:

- IAEA – TECDOC-1226 International Atomic Energy Agency, "Management of changes in NPP power generating companies";
- TP 1.2.6.1.0098-2012 "Standard regulation on the analysis of organization changes and assessment of their after-effects for NPP safety proceeding from IAEA recommendations";
- MR 1.3.3.99.0159-2013 "Guidelines on the analysis of organization changes after-effects for safety".

Brief description of basic analysis stages and assessment of the risk of organizational changes from the viewpoint of NPP safety:

- classification of changes from safety significance viewpoint
- identification of risk types (inherent and individual risks)
- risk calculation and assessment, development of preventive actions intended to decrease the planned organizational changes risk after-effects for NPP safe operation.

3. Experience in design and implementation of shop-free structures at Russian NPPs and analysis of risks, related to organizational, social and human factors.

Brief information on experience in design and implementation of shop-free structures, with "Novovoronezh" and "Leningrad" NPPs taken as examples.

4. Measures intended to minimize after-effects of planned organization changes risk for NPP safe operation; risk management in course of transfer to shop-free structure (with "Baltiysk" NPP taken as an example):

- description of risks and preventive measures intended to minimize their after-effects for "Baltiysk" NPP safe operation;
- analysis of risk value diagrams of the "Shop-free structure" project before and after implementation of corrective measures intended to minimize their values

- risk management in course of transfer to shop-free structure based on coordination of "Shop-free structure" project time schedule with time schedule of "Baltiysk" NPP construction and commissioning, and preventive action implementation monitoring.

Training of personnel in charge of pre-commissioning of NPP units

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Requirements for the training of personnel in charge of pre-commissioning of NPP units are specified in line with documents, listed below:

- "Basic rules of NPP operation safety assurance" (STO 1.1.1.01.0678–2007);
- "Arrangement of NPP personnel development" (ORP-2006);
- "Pre-commissioning preparation at nuclear facilities. Requirements for personnel" STO SRO-S-60542960 00004-2010;
- "Regulations on procedure of training, examination and work permit draw-up for JSC "Atomtekhenergo" employees" PL ATE.108.0157-2013.

Personnel in charge of pre-commissioning shall be trained in line with programs, developed for each job position, based on MU 1.3.3.99.0026-2010 "Systematic approach to NPP personnel training. Instructional guidelines for application".

Systematic approach to training is implemented at five interconnected stages as follows:

1. Analysis – determination of necessity of training and authorities so that to determine required knowledge, skills and expertise for particular job;
2. Planning – development of training programs and topical plans;
3. Development – development of teaching and learning aids;
4. Implementation – training performance;
5. Assessment – implementation of feedback in form of corrective actions.

Individual training program is developed for each employee (newly hired or reassigned), based on standard program and allowing for incoming examination results.

Training for job holding includes the following:

- theoretical training;
- practical training;
- on-the-job probation training;
- primary examination;
- clearance to work unsupervised.

Programs of training for job holding by PNR JSC "Atomtekhenergo" specialists, as well as advanced training programs for managers and specialists of SRO NP "Soyuzatomstroy" member-companies in charge of pre-commissioning works at nuclear facilities, have been developed in NV Training center under JSC "Atomtekhexport":

- Pre-commissioning of NPP Reactor building systems and equipment
- Pre-commissioning of NPP Turbine building systems and equipment
- Pre-commissioning of NPP Chilling systems and equipment
- Pre-commissioning of NPP Electric systems and equipment
- Pre-commissioning of NPP I&C
- NPP operation fundamentals (AES-2006 design)

Advanced training course is taken up in two stages as follows:

- Stage 1 – remote on-the-job training (32 h)
- Stage 2 этап – full-time training (40 h)

Information system "Pre-commissioning support portal"

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JSC "Atomtekhenergo" is JSC "Atomenergoprom" power engineering division business unit in charge of all the types of Pre-commissioning works (PCW) for entire range of NPP systems and equipment. The PCW include development of coordination plan and Pre-commissioning documents, development and filing of Estimate financial documents, supervision of test performance, Reporting and handover documentation submission, and other activities. To automate certain functions, pertaining to the Company key activities, improve PCW supervision and control efficiency, motivate personnel for proactive participation in innovation efforts the Company takes, and shorten lead time, information system "Pre-commissioning support portal development" was initiated in JSC "Atomtekhenergo".

The portal is arranged as a package of soft- and hardware (SHW) tools, intended to ensure the availability of unified access point (portal solution) for users to utilize embedded functions allowing for access right assigned.

The portal ensures the settling of the following tasks:

- set-up of unified project management system;
- Company key activities information support;
- work performance follow-up;
- support of common information area for spatially dispersed divisions of the Company;
- automation of some routine processes, so that to shorten lead time;
- innovation efforts intensity and efficiency improvement;
- personnel prompt access to information on object concerned;
- availability of unified communication tool for users.

The Portal SHW package is comprised of the central server, including a cluster of two data processing servers, central file depot and depot for backup copies, and, as well, slave servers for the Company divisions and client workstations. Entire SHW package of the Portal will be certified for compliance with requirements, made for Class 1D of system intruder protection.

The Portal is comprised of 12 sub-systems, 7 being directly related to PCW performance (e.g., sub-systems for PCW support, documentation storage, planning, control, etc.), with remaining 5 sub-systems intended to implement support and auxiliary functions (administration, communication, informing, etc.

March to July 2014, one of the modules the Portal is comprised of, namely, “Pre-commissioning documentation management”, underwent commercial test operation (CTE). The same was taken up in one of JSC “Atomtekhenergo” branch offices, within the framework of “Rosatom” production system “Optimization of Pre-commissioning documentation issuance for NV NPP-2 power unit 1”. The module CTE results have demonstrated an appreciable improvement and higher efficiency of such documentation issuance. Actually underway is SHW intensive development for other sub-systems and modules the Portal is comprised of.

Personnel training with the use of “Professional” remote training system

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“Professional” remote training system (RTS) is a comprehensive functional solution for remote training management. The system is advantageous for the use of network and multimedia technologies for submission of teaching and learning aids, availability of unified portal access point in internet/intranet network, and use of materials, meeting requirements of International Center SCORM.

“Professional” RTS is developed on the basis of portal solution construction kit Microsoft SharePoint with the use of add-ins SharePoint Learning Kit and Microsoft Learning Gateway, enabling the RTS upgrade with the use of built-in updateable libraries.

In 2007, the set-up of extra possibilities of continuous timely advanced training of employees was decided by JSC ”Atomtekhenergo” Top Management. Considered as a tool for such training was implementation of “Professional” remote training system, conductive for prompt transfer of knowledge on advanced technologies to appreciable number of Pre-commissioning staffers, dispersed in the Company branch offices. From 2010 on, “Professional” RTS has been used and is now highly important for the Company's training process. In late 2012, comprehensive personnel training program for main directions of BN-800- based BAES power unit 4 pre-commissioning was completed; system as above was the key training tool within the mentioned program.

The remote training program is essentially used for personnel on-the-job training. For instance, for JSC "Atomtekhenergo", theoretical training target is acquisition of expert knowledge by personnel, for primary and collateral skills, related to structure, design, particular features of systems and equipment, operation modes (including emergency and transient modes), parameters of physical processes, functions, circuits, parameters of process flow control. The list and subjects of required programs are determined depending on personnel specialization, equipment and work types.

"Professional" RTS-based training process is built up with the use of "double administering" principle: the process is controlled by the branch office ordering particular training course and by the company rendering training services. Such approach is intended to motivate the trainees and ensure adequate level of knowledge acquired upon training course graduation.

For the training course period, "Professional" RTS central server is available round-the-clock, related software is monthly updated to fit training process needs, weekly supplemented with teaching aids, and daily used by trainees.

Information system for operation experience incorporation "ORNIS"

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This report dwells on the creation of Information system for operation experience incorporation "ORNIS" (ISIOE) and such system implementation at "Smolensk" NPP (SNPP). The ISIOE implementation is intended to automate production process, related to operation experience incorporation. The system is being developed on the basis of Smolensk Branch office of JSC "Atomtekhenergo" Training Center.

Operation experience (OE) accumulation and analysis, particularly, experience in NPP operation, is source of information for decisions, intended to ensure NPP safe, reliable and efficient operation. OE incorporation and analysis are conductive for the decrease in the number of accidents and emergencies at NPPs, prediction of possible failures and timely performance of preventive maintenance of process equipment and systems, reduction of power output loss, and, consequently, improvement of NPP efficiency and operations staff skillfulness.

The Information system for operation experience incorporation "ORNIS" is intended to improve OE application efficiency in the context of distributed use of related information by specialists of NPP divisions.

The ISIOE is primarily intended to ensure registration of events. Events are registered both by the system intrinsic features and by adjacent systems of running NPP; registration process depends on particular event source:

- information message – target message or anonymous posting by NPP employee, containing data on possible malfunctions or faults in NPP operation;
- event – detected at operation and maintenance stage, entered into the "Events logbook" by the Shift supervisor;
- information message on external OE – document, containing description of malfunction and corrective actions proposed;
- event, registered by NPP external system – abnormal event, containing series of descriptive attributes. Automatically entered into the "Events logbook" for possible investigation as one of pattern-forming event.

Upon analysis of entries in the "Events logbook", event investigation is decided; here, event concerned can be treated as one of pattern-forming events as follows: malfunction, deviation, or event of minor significance.

Corrective measures (CMs) shall be developed upon investigation of events and filling the OE information message card.

Corrective measure shall be understood as actions taken to eliminate abnormal event root cause and prevent similar event occurrence in the future (STO 1.1.1.01.002.0646).

Each stage of CM development and implementation is worked out by NPP division in charge of OE incorporation and malfunction investigation.

Reporting documents on OE incorporation are compiled upon registration and investigation of events, and implementation of corrective measures.

As of the date, the Information system for operation experience incorporation "ORNIS" has successfully undergone trial operation; in December, 2012, the system was handed over for commercial operation and SNPP.

Experimental substantiation of operability of passive safety systems in new designs of VVER-based NPPs

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Nuclear power plant with new-generation water-moderated water-cooled reactor "3+", with improved performance ratings (AES-2006), has been designed in the Russian Federation. The key feature of the design is the use of additional passive safety systems, in combination with conventional active systems. Used as construction site for "AES-2006" prototype units is the second stage of NV NPP. Moreover, underlying solutions for safety assurance are applied at "Kudankulam" NPP running in India.

Passive Core cooling systems for VVER-based designs as above include the Core flooding system from Stage 1 hydro-accumulator (HA-1), additional system for Core passive flooding from Stage 2 hydro-accumulators (HA-2), and, as well, Passive heat removal system (PHRS).

Implemented in the HA-2 system is four-stage profiling of flow characteristic, ensured through the use of header, enabling the flow passive correction, based on consecutive cessation of outflow via discharge line, turning out to be located above water level in tank. Additional function of the HA-2 system is automatic passive removal (withdrawal) of primary circuit non-condensable gases from SG tubing, which is conductive for heat exchange improvement and provides SG longer operation in condensation mode.

The Passive heat removal system is intended for Reactor core residual heat long-term removal in case of accident with total loss of a.c. sources, both with tight primary circuit and in case of primary or secondary leak. In case of primary leak, the system operates jointly with Stage 2 hydro-accumulators. The system is comprised of four independent trains (one for each SG of Reactor plant).

In case of accident with Reactor coolant pipeline (RCPL) break, the PHRS ensures the transfer of horizontal SGs to the mode of condensation of primary steam, supplied to the tubing from Reactor, thus providing the Core makeup. As a result of condensation, secondary water is heated up to saturation temperature, with steam generation. Due to natural circulation in the PHRS steam-condensate path, steam is supplied to steam heat exchanger, installed on the containment outer surface. Steam is condensed, giving up heat to ambient air, whereas generated condensate is supplied back to the SG inter-tubular space.

To substantiate design functions and operability of new-generation VVER RP Core passive cooling systems, large-scale experimental research program was initiated in SSC RF-IPPE. Key data on performed work results are presented in the report.

Mathematical model of reactor core thermohydraulics at supercritical paramaters

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One of the ways to improve efficiency of power plants with nuclear reactors is transition to supercritical parameters. However transition to supercritical parameters leads to necessity to solve problems related to heat exchange degradation. At present, judging by experimental tests it was detected that sharp falling of heat transfer coefficient was due to the effect of flow heat acceleration.

Calculation of coolant flow with supercritical parameters in the reactor plant flow tubes (active heating area both rising and falling) is characterized by the fact that changing of flow enthalpy is much higher than the characteristic value of kinetic energy. It leads to the necessity to solve transfer equations considering kinetic effects – Burnett summands comprising stress tensor and thermal flows. Coolant flow is realized in conditions of Kn ® 0, $Re >> O(1)$. With these values of operating parameters Burnett summands have the same order as those of Navier-Stokes in the impulse equation, however in case of turbulent flows – the same order as Reynolds shear stresses.

Mathematical model of coolant flow was constructed on the basis of formal application of Chapman-Enskog procedure and constraining of the first three summands in expanding by Knudsen numbers. Such an approach allowed receiving an expression for the stress tensor and thermal flow considering Burnett corrections.

Coolant flow in reactor plants with modest and large Reynolds numbers is mainly determined by geometry: for the majority of internal streams the flow is limited by the wall (walls), and geometry of walls determines the selected (prevailing) direction. This consideration allows transferring from Reynolds conservation equation considering Burnett corrections for the stress tensor and thermal flow to simplified Reynolds equations that are parabolic.

Reynolds parabolic equations were received applying expansion by Reynolds numbers to the initial equation system and cutting of summands of smallness of higher order as compared to $O(1)$ and $O(1/Re^{-1/2})$. The received simplified model of coolant flow includes summands considering Burnett corrections.

Model of coolant flow in reactor cores

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Models of flow developed on the basis of parabolic Navier-Stokes or O. Reynolds equations (depending on the type of low) are most complete out of all parabolic models. Currently the model of parabolic equations is used for calculating flows with supersonic parameters in case there are chemically reacting gases or without those without chemical reactions. However, its main advantage – possibility to calculate lateral velocity by solving a parabolic equation is very attractive, especially during the development of real time training systems or concept simulators for nuclear engineering.

Out of many tasks of modern reactor thermal hydraulics the task of accurate transverse momentum and heat transfer in reactor cores is the most important.

Construction of the model is performed on the basis of cutting sums containing second differential coefficients, both repeated and mixed, containing a marching coordinate differentiation in Navier-Stokes equations. Thus the resulting model is not completely consequential for the solved equation contain only a part of complete Navier-Stokes summands with the order of $O(1/Re^{-1/2})$, but at the same time they preserve a part of summands of smallness of higher order.

The proposed model was constructed for two-dimensional coolant flow in the active core and annular downhole space. Momentum equation contain summands determined by gravity action, lift and friction and pressure resistance force. The thermal conductivity includes a summand, considering energy release determined by nuclear reactors. The connection between the energy equation for the coolant flow and thermal conductivity equation is performed on the basis of thermal current formula. Heat transfer coefficient is calculated using known formulas for wire or mesh wrapped rod bundles.

Numerical solution of the received equation system is performed by double-sweep method.

VVER-1000 core characteristics during multiple recycling of remix fuel

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This paper pays a special attention to justification of fuel cycle characteristics in existing VVER-1000 reactor plants and in prospective VVER RP designs while using fuel produced in compliance with REMIX-technologies. It considers basic peculiarities of VVER active core loading with this fuel performing multiple recycling. It includes comparison with other variants of fuel cycles (manly with MOE-fuel); considering the possibility of feeding VVER with uranium-233 from thorium screens BN-1200 and exchange of unloaded plutonium between VVER and BN.

Relevance of paper: Solving of problems related to spent fuel accumulation and providing with fuel resources for the whole life cycle of power units are vexed questions today both for domestic nuclear power industry, and for its place on the world market of nuclear industry services. This is the reason why seeking and analyzing optimal variants of closing fuel cycle of nuclear energy system with thermal and fast neutron reactors is the critical task. It is caused by the necessity to reduce amounts of stored spent nuclear fuel, increasing the efficiency of using uranium-238 and reduction of natural uranium consumption. It is proved that one of the prospective variants of closing nuclear fuel cycle for VVER is the concept of using REMIX-fuel developed by NPO "Radiyevyi Institut" in partnership with SRC "Kurchatov Institute". It also considers proposals on Work program on justifying the role and position of REMIX-fuel in the closed nuclear fuel and experimental justification of REMIX-fuel technology elements and radiation safety aspects while handling with REMIX-fuel.

Optimization of concrete composition of new generation NPP radiation protection

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Modern approach to designing and construction of nuclear power plants of new generation is required already at the stage of designing a package of activities oriented to decrease amounts of radioactive wastes in subsequent decommissioning of NPP power units.

Previous investigations proved that amounts of radioactive wastes can be substantially decreased on the account of object-oriented selection of low-level concrete compositions and construction steels.

Taking into consideration that NPP designs of new generation, specifically NPP with VVER-TOI, are well optimized. Such optimization shall be performed with concrete compositions that are used in constructions of reactor radiation protection.

To achieve that goal some calculation investigations of transferring neutron radiation have been performed in one-dimensional geometry and activation of radiation protection construction performed from different concrete compositions.

Data previously obtained by authors were generally used to define the content "impurity" and "trace" elements, such as cobalt, europium, cesium etc, isotopes of which form long-lived radionuclides.

Tests showed that **substantial decreasing** of induced activity levels, and subsequently amounts of radioactive wastes is possible by object-oriented selection of low activity compositions of VVER type reactor vessel protective concretes.

Considering the perspective mass construction of NPP units of new generation and their optimization it is necessary to perform the following in the nearest future:

- development, certification and implementation of technical specifications and passports for process and chemical composition of protective materials for the construction of most vital constructions and elements of radiation protection;
- development of respective normative technical documents, specifying practical use of prospective materials for NPPs of new generation.

Practical implementation of approaches suggested in the report will allow performing the future decommissioning stage of new generation NPP safely and economically by reducing amounts of radioactive wastes, reducing of radiation exposure on the personnel, population and environment, increasing amounts of re-using materials of stripped down NPPs.

Operability assurance of NPP power unit with VVER in modes of power cycling and load following (100-50-100)% from P_{NOM} preserving main parameters of energy efficiency on the basis of combined application of high voltage variable-frequency electric drive on energy-intensive equipment (RCPS, FP, CEP and CN) in the automatic mode

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All new designs, including VVER-TOI still do not provide for in-depth power maneuvering - (100-50-100)% from N_{nom} and load following.

The trend of the next decade for Russian generation, especially considering its European part is inability to maintain NPPs in the basic mode and, consequently, their operation in modes of insufficient power with multiple transitional and power cycling modes, which suggests another approach to quality of reactor plant (RP) power control systems and energy-intensive process equipment of NPP Unit.

Requirements of potential international customers to maneuvering – (100-30-100)% from N_{NOM}. All designs of key international engineering companies provide for maneuvering – (100-25-100)% from N_{NOM}.

Inability of power maneuvering in new NPP designs with VVER is the loss of international market, financial losses of SC "Rosatom" in the domestic market considering sale of electric energy – one day of downtime of NPP unit makes the loss of 28 mln. rubles.

Currently there appeared a necessity of complex task decision considering NPP unit with VVER operation maintenance in modes of power cycling and load following (100-50-100)% from P_{NOM} preserving basic parameters of energy efficiency, which is subdivided into two directions:

1. Reactor plant (RP) power adjustment and control within (100-50-100)% from N_{NOM} (resp. "OKB Gidropress", SRC "Kurchatov Institute" and OAO "TVEL").

2. Adjustment of power-intensive process equipment (RCPS, FP, CEP and CP) and turbine generator in modes of power cycling and load following (100-50-100)% from P_{NOM} preserving basic parameters of energy efficiency in the automatic mode (resp. OAO "Atomenergoproekt").

With current control methods of energy-intensive process equipment and turbine generator the operation power unit in the mode of daily maneuvering and load following (100-50-100)% from N_{NOM} is very ineffective both from the point of view of the quality of process equipment control systems of I, II circuit and cooling water loop, and from the point of view of consuming electric energy auxiliary process loads, and the level of operation reliability and safety of the whole power unit according to requirements specified in new designs of NPP with VVER.

It is possible to implement the optimum mode of NPP unit operation in transitional and power cycling modes using modern systems of variable-frequency electric drive (VFED) in RCPS, FP, CEP and CP pumps.

Variable-frequency electric drive can perform a smooth and concurrent performance transition of 4 RCPS units according to the specified or configurable process parameter or any other regularity, maintaining the equalized coolant flow in all RP loops.

Before the inclusion of technical decisions on RCPS into operating documentation it is necessary to perform the complex of research and development work considering justification of using high-voltage variable-frequency electric drive on RCPS with complex in-situ testing on stand equipment according to the developed, agreed upon and approved program and research and development work on simulating of power unit operation in transitional and power cycling modes using high-voltage variable-frequency electric drive on power-intensive process equipment of I, II circuit and power unit cooling water loop.

Designing the construction of fast breeder reactor vessel working with lead coolant BREST-OD-300

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Safety principles of reactor plants, diagram of closed fuel cycle, concept of natural safety "BREST-OD-300" reactor plant have been considered, including design features and properties.

Proposals on setting of requirements to structure and content of operational limits and conditions, including limits and conditions of safe operation

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Novovoronezh nuclear power plant

According to requirements of NP-006-98 Chapter 16 of the report on validation of safety of the power unit includes limiting values of controlled parameters and actuation setpoints of safety systems as safe operation limits. Safe operation limits are being replaced with operational limits.

It is suggested to systematize and structure design limits and conditions for all conditions and modes of the power unit on the basis of protection in depth.

Operational limits and conditions are set to prevent violation of normal operation, including testing.

It is suggested to set limits and conditions during the operation with minor deviations to prevent development of violations of normal operation in emergency situation.

Limits and conditions of safe operation are set to prevent severe accident consequences.

In conditions of severe beyond design basis accidents it is suggested to set limits and conditions of risk reduction for population and environment.

Limits are set to identify initial and final limits of various conditions and modes of the power unit. Limits defining initial limits of conditions are used to perform various activities by automatic systems or operating shift personnel in order to prevent achievement of limits, defining final limits of conditions.

The possibility of performing such activities is the condition of preventing the transition to a more condition judging the radiation exposure aftermath.

Technical requirements for Safety parameters representation system to be used in the designs of new-generation NPPs

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The Safety parameters representation system (SPRS), Level 1, is intended for status monitoring of critical safety functions (CSF). In case of CSF accident-induced disturbance, video frames of SPRS, Level 2, containing interactive electronic simulators of the CSF recovery procedures, are activated.

We propose, in addition to information support of human actions, related to power unit status monitoring and decision making with an eye towards control action performance, to use the SPRS for operator support in course of performance of actions as above.

Power unit safety analysis involves the analysis of accident scenarios with certain IEs and initial conditions. This is the basis for event-orientated accident management. The Design-basis accident mitigation instruction (DBAMI) and Beyond design-basis management manual (BDBAML) are supplemented with control action procedures for particular accident scenarios, with certain IEs and equipment operation conditions, addressed in the design or additional (non-conservative) design-basis justification. It is noteworthy that a certain control action procedure is used for each accident scenario, and the number thereof is not limited. Yet, all the major accidents, registered at NPPs, evolved in line with non-incorporated scenarios, for which control action procedures were neither developed nor justified.

Starting from the 1990th, DBAMIs and BDBAMLs, elaborated on the basis of symptom-oriented accident management, have been developed and implemented at NPPs running in Russia. Here, a set of interrelated control action procedures is used for any accident scenario.

The use of video frames, linked to particular process systems, for control action performance in all the modes (including emergency), is a time-intensive task, implying a high intellectual performance of operator. This is a coercive measure to be resorted to every time unlimited number of event-oriented procedures is used. While using a standard set of symptom-oriented accident mitigation procedures, control actions shall be performed from formalized video frames of interactive electronic simulators of symptom-oriented emergency procedures.

Subsection 3.2
ADVANCED NUCLEAR POWER PROJECTS

Nucler power engineering in context of use of reactors with water coolant

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As of the date, the basic type of Reactor plant (RP), used in our country and abroad, is RP with VVER (PWR) power unit, operating as per two-circuit pattern. Of about 500 power units, running worldwide, some 300 are of this very type. Despite the "post-perestroika" stagnation, Russian nuclear power engineering has made a good progress of late. As far as VVER power units safety and reliability scientific substantiation is concerned, "Rosatom" is among the world leaders. The current state of Russian VVER-based RPs does not completely reflect potentiality thereof. Really expectable in near future are their service life extension to 60 years, nuclear fuel burn up fraction increase, appreciable improvement of breeding ratio thereof with adoption of Reactor spectral regulation technique, implementation of newest arrangement solutions, chain power extension from 300 to 1500 MW, etc. Real application is found for technical solutions for ship-based RPs for floating NPPs (KLT-40 can be given as an example).

Water-coolant RPs can be developed based on both up-to-date circuitry and structural solutions and "extraction" of previously unused possibilities from existing RPs. In particular, cooperation between "academic science" and key nuclear operators enabled substantiation of the "Concept of extended application of coolant natural circulation (VVER-1000/1200)" to ensure self-actuation and support of auxiliary loads in case of NPP long-term blackout.

The report contains some doubts as to "Rosatom" complete rejection of further development of pressure-tube pulsed uranium-graphite reactors – MKER design, developed after Chernobyl event, was free from safety deficiencies, inherent in high-power pressure-tube reactors.

Beyond consideration are good foreign practices, related to the use of vessel-type boiling-water reactors (about 100 power units worldwide), corroborated with vast domestic experience (VK-50 reactor, NIIAAR; VAU-6C, "A.P. Alexandrov Research Institute of Technology"). In particular, these features might have been used as displacing capacities for "Bilibino" Nuclear heat and power plant, with practically overage EGP-6 power units.

"Fukushima" accident is totally unrelated to reactor type and does not compromise vessel-type water-moderated power reactors (BWR).

The most significant problem, related to the development of water coolant with over-critical parameters (supercritical pressure - SCP) to avoid existing "low temperature limit" in RP is not supported with adequate extent of scientific study and R&D; here, the key functions shall be assigned to physicists, with an eye towards the development of safe reactor, ruling out prompt criticality and reactivity minimally sensitive to coolant density variations. The problem was taken up as early as in the 1960th by a prominent physicist S.M. Feinberg in "Kurchatov Institute"; under study was ship reactor option with intermediate neutrons (VPN-705), but in fact nothing has changed for fifty years.

The problem can be successfully settled only with high-quality competent centralized management of all the related activities, with adequate scientific and design engineering support.

Support of VVER NPP unit operation in maneuver and load-follow modes (100-50-100)% of P_{NOM} with main EE parameters retained by extensive use of variable-frequency HV electric drive in energy-intensive process equipment (RCPS, FWP, CEP and RFP) in automatic mode

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By now, all the new-generation designs, including VVER-TOI, have no possibility of wide power maneuvering - (100-50-100)% of $N_{nom.}$ and load following.

The near-future trend for nuclear facilities in Russia, especially in its European part, is impossibility of NPP basic mode maintaining, and, respectively, operation thereof in partial-power modes, with a wide variety of transient maneuvering modes, which fact implies a diverse approach for the quality of systems for Reactor plant (RP) power regulation and power-intensive process equipment of NPP unit.

Maneuvering requirement, presented by potential customers, is (100-30-100) % of N_{nom}. Maneuvering range, embedded in all the designs of key engineering companies worldwide, makes (100-25-100)% of N_{nom}.

The absence of power maneuvering possibility in new-generation VVER-based NPP designs implies the loss of market positions and financial loss of SC "Rosatom" on domestic energy market (the cost of one idle day of NPP unit amounts to 28 mln. roubles).

Actually observed is the necessity of integrated solution for the problem of VVER NPP unit operation support in maneuvering and load-follow modes

(100-50-100)% of P_{nom}, with basic energy efficiency parameters retained; here, two directions can be discriminated:

1. RP power variation and regulation within the range (100-50-100)% of N_{nom} (entities in charge: OKB "Gidropress", RCC "Kurchatov Institute", JSC "TVEL").

2. Regulation of power-intensive process equipment (RCPS, FWP, CEP and RFP) and Turbine generator in maneuvering and load-follow modes (100-50-100)% of P_{nom} , with basic energy efficiency parameters retained in automatic mode (entity in charge: JSC "Atomenergoproekt").

With actually available techniques of power-intensive process equipment and TG regulation, power unit operation in daily maneuvering and load-follow modes (100-50-100)% of $N_{nom.}$ is extremely inefficient – both from the viewpoint of quality of regulation systems for process equipment of primary and secondary circuits and cooling water circuit, and from the viewpoint of power consumption for process auxiliaries, and, as well, entire power unit reliability and safety levels, in line with requirements presented in new designs of VVER NPPs.

The implementation of NPP unit optimal operation mode in transient and maneuvering modes is possible with the use of up-to-date systems of variable-frequency electric drives (VFED) in RCPS, FWP, CEP and RFP.

The variable-frequency electric drive enables smooth synchronized variation of capacity of four RCPSs in line with pre-set or variable process parameter or any other consistent pattern, retaining the same coolant flow in all the loops of RP.

Incorporation of RCPS-related technical solutions into WD shall be preceded with R&D efforts, intended to substantiate the use of variable-frequency HV electric drive for RCPS, with integrated field testing at bench equipment, in line with duly developed, agreed and approved program and R&D for power unit operation simulation in transient and maneuver modes, with the use of variable-frequency HV electric drive in energy-intensive process equipment of primary and secondary circuits, and power unit cooling water circuit.

Optimization of concrete composition for new-generation NPP radiation protection

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Currently applicable approach for new-generation NPP designing and construction calls for development of measures, intended to reduce RW amount in course of future de-commissioning of NPP units; measures as above shall be taken as early as at the stage of designing.

According to previous investigation results, RW amount appreciable decrease is possible due to purposeful selection of low-activated concrete mixes and construction steel grades.

As new-generation NPP designs, including VVER-TOI, are optimized ones, such design optimization shall as well extend on concrete mixes to be used in structures, providing Reactor radiation protection.

To this effect, unidimensional design investigations of neutron irradiation transfer and radiation protection structure activation for different concrete mixes, used for such structure, were taken up.

The content of impurity and tracer elements (Co, Eu, Cs, etc.), on which surface long-lived radionuclides are generated, was determined with predominant use of data the authors obtained heretofore.

According to investigation results, induced activity level **appreciable decrease** (and, respectively, the decrease in RW amount) is possible due to purposeful selection of low-activated protective concrete mixes for VVER-type Reactor cavity.

Allowing for prospects for large-scale construction of new-generation NPP units and optimization thereof, the following measures shall be implemented in the near future:

- development, qualification and implementation of specifications and datasheets for technology and composition of protective materials for construction of the most critical structures and components of radiation protection;
- development of Regulatory technical documents to govern practical use of future-oriented protective materials for new-generation NPPs.

Implementation of approaches, brought forward in the report, will be conductive for safe and cost-efficient de-commissioning of new-generation NPPs, through the decrease of RW amount, personnel and population exposure, environmental impact, and higher amount of material re-use after NPP de-commissioning.

Technical parameters of VVER-1250 with spectral regulation (VVER-S-1250)

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In the recent years, certain works were performed with an eye towards the improvement of investment appeal of future-oriented vessel-type power reactor technologies with water coolant (Super-VVER). Works as above were performed by SRC "Kurchatov Institute", OKB "Gidropress" and JSC

"Atomenergoproekt" under commission by JSC "Concern Energoatom". The "Super-VVER" has two independent and properly structured aspects: evolutionary and innovative.

The VVER evolutionary development with spectrally regulated Core reactivity variation in course of the Core burnup was identified as VVER-S. The VVER-S type reactor is a modification of evolutionarily-modernized thermal reactor, meeting requirements of Nuclear power engineering system in the mid-term. The VVER-S development has the following objectives:

- improvement of efficiency and competitiveness of future-oriented VVER units;
- natural uranium saving at Nuclear fuel cycle facilities NFCF (less than 130 t of natural uranium per 1 GW*year of energy);
- development of flexible nuclear cycles, embracing both NFCF and CNFC (Closed Nuclear fuel cycle);
- loading of different types of fuel (UOX, REMIX, MOX and combination thereof);
- ensuring adequate safety level of Reactor control and protection with no use of system for reactivity compensation with boron acid in all the modes of operation, including "idle" modes.

Core reactivity variation spectral regulation in course of fuel burnup is performed due to the change of water-uranium proportion in FA by the use of movable water displacers.

The document presents the results of design study of VVER Core parameters with spectral regulation of power 1250 MW, in stationary 6-year fuel cycle with uranium fuel and no use of boron regulation system in course of power operation and in "idle" (refueling) modes.

Autonomous passive systems of reactor plant emergency cooling on the basis of two-phase thermal siphons

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Under consideration are operation principle, circuits and design parameters of thermal siphon-type RP PHRS, PRZ passive cooldown systems (PCS), Hermetic volume PHRS and PHRS for SNF FP, and, as well, of thermal siphon device for secondary circuit piping tight penetrations (TP) passive thermal protection. The use of heat-exchange equipment on the basis of closed-loop type evaporators and condensers (two-phase thermal siphons – TTS) is one of efficient methods of RP residual heat removal safety and reliability improvement in case of long-term total power loss. The TTSs, as autonomous closed-loop type heat removal features, create the system of extra barriers between energy release source and ultimate sink.

RP PHRS is intended for heat removal from the primary circuit and comprised of four autonomous cooldown loops, connected to ECCS pipelines. The PRZ PCS is comprised of two autonomous loops and provides PRZ cooldown to ensure primary pressure fast decrease for earlier connection of ECCS HA, which prevents Reactor re-criticality. Each loop of the RP PHRS and PRZ PCS includes intermediate heat exchanger on the basis of ordinary TTSs, closed circular two-phase intermediate circuit outside the Containment, and heat exchanger-condenser for heat removal to ultimate heat sink.

Particular feature of systems under consideration is serial installation of two intermediate closed heat-transfer circuits: cooldown heat exchanger TTS assembly and intermediate circuit, operating as per circular TTS principle. Circuits as above are installed in between the primary circuit and ultimate sink. Performance parameters of RP PHRS and PRZ PCS are indicated, along with comparative efficiency parameters of proposed systems, with the use of water in emergency heat removal tanks, and atmospheric air, as ultimate sinks.

SNF FP PHRS and Containment PHRS are comprised of circular TTSs, with evaporators installed in the pool and in the Containment, respectively, whereas the condensers are installed outside the Containment. Available here are design parameters of FP and Containment PHRS, along with comparison of thermal efficiency of system options proposed.

The piping TP passive thermal protection device is arranged as L-type TTS with horizontal coaxial evaporator with hot secondary pipeline inside, and vertical tubular condenser, installed in vertical air duct outside the Containment. Heat is passively removed from hot pipeline by the transfer of latent heat of steam generation of TTS intermediate coolant to atmospheric air, circulating due to natural draught in air duct. Given here are the results of design simulation of piping TP passive thermal protection process, with assessment of TP thermal protection efficiency dependence on atmospheric air temperature.

Development in full spectrum ACP nuclear power plant series in NPIC

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To meet the variable requirement of utilities and markets, Nuclear Power Institute of China (NPIC) has been continuously developing advanced nuclear power plants for achieving enhanced safety, better performance and more economic. The full spectrum of ACP serial Nuclear Power Plants provide three distinguished products: the standard ACP1000 for nuclear power, the medium-sized ACP600 and the modular multi-purpose SMR (Small Module Reactor) ACP100. All of the above designs take the philosophy of passive safety and the experience of Fukushima feedback. The core damage frequency (CDF) and large radioactive release frequency (LRF) has been further reduced by the optimized passive & active safety systems and dedicated severe accident mitigation measures to the magnitude of 10^{-7} and 10^{-8}. The cost of ACP series have been further brought down by longer refueling cycle to 24 months, advanced core management, 60 years design life, Leak-Before-Break (LBB) and localized manufacture. While the ACP100 are targeting the specific market such as floating reactors for coastal deployment, district heating and independent grids. Based on the fundamental research of Science and Technology on Reactor System Design Technology Laboratory (LRSDT), advanced design technologies like on-line reactor core monitoring, Best Estimate Plus Uncertainty (BEPU) analysis, computational fluid dynamics (CFD) analysis and etc. has been elaborately implemented for reliable and flexible operation.

Section 4

INTERNATIONAL COOPERATION FOCUSED ON ENSURING NPP SAFETY

INTERNATIONAL SCIENTIFIC AND TECHNICAL COOPERATION AS AN EFFECTIVE WAY OF OBTAINING NEW KNOWLEDGE AND EXPERIENCE

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Over all more-than-20-year life span of Rosenergoatom Concern OJSC, international scientific and technical cooperation (ISTC) has been an integral part of its activities, which is focused on:

- strengthening Rosenergoatom reputation, and establishing favourable external conditions for its functioning and development in the environment of international business community;
- organization of fulfillment of international obligations of the Russian Federation in the part related to the operating organization activities;
- making use of international best practices in the technology, engineering, organizational and human performance areas by means of sharing operational experience, knowledge and technological achievements;
- making use of R&D results in the safety field, their reviewing, analysis and lesson learning with implementation of appropriate measures, in order to avoid duplication of efforts and to reduce costs;
- safety culture enhancement;
- making use of international best practices as necessary for implementation of overseas business projects on nuclear plant operation issues;
- utilization of foreign business solutions with positive references.

ISTC of Rosenergoatom is implemented on the basis of international conventions and agreements in the field of peaceful use of atomic energy endorsed by the RF government, the State Corporation «Rosatom», agreements and protocols concluded between Rosenergoatom and foreign organizations and enterprises, which activities are affinitive to that of Rosenergoatom or may present a practical interest for it.

Rosenergoatom co-operates with French generating company Electricite de France, the world largest one, for more than 20 years, with Ukrainian NNEGC «Energoatom» – for 15 years, and with Spanish company Iberdrola – since 2005.

Historically, Rosenergoatom takes an active part in the work of major international organizations such as World Association of Nuclear Operators (WANO), International Atomic Energy Agency (IAEA), European operating organizations club (EUR).

Activities under the IAEA aegis are performed in connection with governmental obligations of the Russian Federation, under co-ordination of the State Corporation «Rosatom».

Membership in WANO and European club EUR reflects a voluntary intention of Rosenergoatom to participate in the work of these major international nuclear organizations that bring together nuclear operators functioning in the field of the use of atomic energy.

Since 2013, Rosenergoatom, under the auspices of the State Corporation «Rosatom», is involved in activities of Nuclear Energy Agency of the Organization for Economic Co-operation and Development (NEA/OECD).

Major events like accidents at Three Mile Island NPP in the USA, Chernobyl NPP in Ukraine and Fukushima-Daiichi in Japan change the situation in the world, and the international co-operation changes correspondingly.

While the process of understanding and learning from lessons of accidents is going on, new challenges appear which increasingly assume an international character in the environment of globalized industry.

Taking the external context into consideration, Rosenergoatom continuously follows the global development trends, while solving in parallel its own internal tasks by means of effective use and introduction of the best international practices into the company's production activities.

ACTIVITIES OF WANO MOSCOW CENTER AIMED AT SAFETY IMPROVEMENT

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After the Fukushima accident in Japan 11 March 2011, the World Association of Nuclear Operators (WANO) embarked upon a course towards an increased influence on the nuclear society, and the dedicated Mitchell's commission had developed five main areas of WANO reformation.

In line with the course chosen towards WANO reformation, the WANO Moscow Center activities in 2013 were oriented on implementation of the Mitchell's commission recommendations, improvement of the programme quality, augmentation of the resources needed for this.

WANO had taken obligations to reach a significant progress of implementation of the Mitchell's commission recommendations. In course of implementation of the recommendations the following has been accomplished:

- The WANO-MC Board of Governors' initiative on establishment of WANO representations at each NPP site has been implemented. Relevant contracts with WANO-MC member organizations concerning the representations have been drafted and approved. Personnel of the representations have been selected on a competitive basis, received training in WANO programmes and English language, and have been certified.

- In 2013, there has been launched a WANO Moscow Center pilot project on NPP monitoring, identification of an appropriate level of support and development of the support provision plans. This has become feasible due to establishment of the representations and development of the WANO-MC documents defining a procedure for monitoring and support rendering. As per the monitoring results, the WANO-MC plant interaction plans for 2014 have been developed for each NPP and have laid down the basis for development and approval of the WANO-MC plan for 2014, which is oriented on provision of support to WANO-MC members.
- One of the most important initiatives of WANO-MC, namely, establishment of Regional Crisis Center (RCC) has been implemented on the basis of Rosenergoatom Crisis center. The RCC membership encompasses all companies members of WANO-MC and their supporting organizations. Necessary procedural documents have been drafted and approved, and 2 drills have been conducted with NPPs participation. The RCC development activities will continue in 2014.
- One of the Mitchell's commission recommendations on WANO reformation relates to an expansion of WANO programme scope including *inter alia* Severe Accident Management (SAM) area. General responsibility for the SAM project has been imposed to WANO Moscow Center. By the end of 2012 the reporting documents for the said project have been drafted, and currently self-assessment activities of operating organizations and NPPs in the Severe Accident Management area are under way.

Regarding the cooperation with NEA/OECD

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Since 1 January 2013 the Russian Federation became a full member of Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD).

NEA/OECD is one of a few authoritative international organizations providing for the environment for international technological, scientific and technical cooperation. The NEA/OECD scope of activity encompasses studies of long-term aspects of global peaceful utilization of nuclear energy in the context of sustainable development, technical aspects of ensuring and regulation of nuclear and radiation safety (NPP safety, decommissioning of nuclear installations, radioactive waste management, nuclear fuel cycle), performance of applied studies, calculation and experimental works as part of international programmes and projects, support to enhancement and harmonization of international legislation, experience and information exchange on the issues of nuclear and radiation safety regulation in course of utilization of atomic energy, carrying out the researches helping to enhance efficiency of regulations.

The full membership format of the Russian Federation involvement in NEA/OECD provides some additional advantageous conditions for promotion of Russian nuclear power technologies to the global market. In particular, this may help in our work on harmonization of the existing Russian requirements to reactor technology certification with the international practice, would allow us to make full use of NEA Data Bank containing a vast and useful information, inter alia, on calculation codes, integral experiment results, properties of reactor materials, and capacity of experimental facilities available at member countries. Involvement of Russian organizations in implementation of NEA/OECD programmes and projects would be beneficial for enhancement of calculation and experimental justifications of safety of Russian nuclear power technologies.

From its side, Russia being a strong actor at the global nuclear power market and one of a few countries that are continuing to perform an extended R&D programme will be sharing its experience and knowledge with other NEA/OECD member countries.

EDF outline & Post-Fukushima Safety Improvements – Practical implementation and challenges

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The presentation will provide an updating of the information already exchanged with REA and at the last MNTK meeting in May 2012. After an overall view of main EDF activities, results and nuclear experience, the presentation will focus on the main challenges for nuclear generation and development. The results of «Stress tests» in EDF reactors and Main steps of deployment will be detailed, following the last updated studies and exchanges with the French Safety Authority: assessment of existing margins, improvements for existing NPPs following the stress tests. Key additional measures will be presented: implementation of an «hardened safety core» and a «rapid action force» (which will be detailed in another EDF presentation). The main steps to implement modifications along time, their content, technical issues and challenges will be detailed and discussed, especially in regard of the safety objective to prevent and mitigate any large release.

Lessons learned from severe accident at Fukushima-Daiichi NPP

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11 March 2011 at 14:46, the Great Eastern Japan Earthquake occurred that caused an inconceivable damage to the region. At the moment, the earthquake magnitude exceeded 9.0 points, and that was the heaviest earthquake over the history of Japan. The subsequent tsunami brought a series of waves (7) that covered 561 km^2 of the coastal area and took more than 25000 human lives. Meanwhile, the design wave height for NPP Fukushima Dai-ichi was only 3.1 m. In accordance with outcome of an assessment based on the «Tsunami Assessment Methods for Nuclear Power Plants», a methodology provided by the Japan Society of Civil Engineers, the design tsunami level had been raised up to 5.7 m. However, on 11 March, the incident wave height was in the range from 14 to 15.5 m.

11 March 2011, after the earthquake and tsunami strikes, Fukushima Dai-ichi NPP had lost all internal and external sources of electric power supply. This had resulted in the severe accident at the NPP and in a large release of radioactive material to the environment.

The report presents the accident progression and its main consequences. The basic causes that led to loss of critical safety functions at Fukushima Dai-ichi Units 1–3 are considered, including:

- Loss of the reactor core cooling function;
- Loss of the radioactive material confinement function.

Weaknesses in the defence-in-depth levels and design deficiencies are considered as well as shortcomings as regard to NPP siting assessments, safety culture and emergency management at Fukushima Dai-ichi NPP.

The most important lessons that should be learned from the Fukushima Dai-ichi accident are presented as well as positive experience of coping with the tsunami impact that had been gained at Onagawa and Fukushima Dai-ni NPPs suffered from the same earthquake and tsunami of 11 March 2011.

A brief analysis of Japanese nuclear industry perception of the lessons from severe accidents at TMI and Chernobyl NPPs in the context of Fukushima Dai-ichi accident is given.

On the basis of the lessons learned, proposals are made regarding strengthening of nuclear safety globally and in Pacific region particularly.

LIABILITY FOR NUCLEAR DAMAGE: INTERNATIONAL COOPERATION ISSUES

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By now, due to joint efforts of states and international organizations, an international legal regime has been established for regulation of the safe use of atomic energy, with the Russian Federation being a part to this regime. The global safety regime comprises international conventions, the IAEA safety standards accepted internationally, and multilateral and bilateral cooperation. One of the components to the complex nuclear safety system is liability for nuclear damage («nuclear liability»), which being an integral part of both the global nuclear safety regime and the technology development, plays in the same time a role of a connecting link between these two. It is the area where a balance between risks and benefits, between an adequate compensation for damage and investors' protection shall be ensured. Any documents adopted by the international community on measures to strengthen international cooperation in the nuclear safety field emphasize the importance of existence of efficient and coordinated nuclear damage liability mechanisms at both national and global levels. The IAEA has repeatedly noted that Member States shall co-operate towards establishment of a global nuclear damage liability system that would consider interests of all states that may suffer from a nuclear accident, with the goal of assuring provision of an adequate compensation for nuclear damage.

Specificities of nuclear damage have resulted in establishment of a special civil liability regime at supranational level, because a common law regime would be inappropriate for the specific problems encountered in this field. This regime is aimed at an increased responsibility for determination of legal scope of the use of atomic energy, in addition to establishment and continuous improvement of the environment needed for nuclear safety ensuring. The global nuclear damage liability regime is shaped by relevant international conventions and protocols to amend them (one group is under the auspice of OECD, another is under the auspice of IAEA). The main refinements to the conventions are related to expansion of the nuclear damage definition, to extension of the action limitation period for damage to physical persons, and to expansion of geographic boundaries for application of the conventions. The basic principles of liability laid down in the conventions which are absolute and exclusive responsibility of the nuclear installation operator, liability limitation by the amount and time span, availability of a financial guarantee, unity of jurisdiction and absence of discrimination. Development of national legislation in the field of the use of atomic energy is determined by the same factors as the international industry development, but is influenced by various processes that go in individual countries.

Special civil liability regime requires from operating organization, as a mandatory action, to secure funds as necessary for damage compensation to sufferers. Nuclear insurance is the most important financial instrument when liability for nuclear damage is considered, and is essentially the only available form of a financial guarantee of nuclear damage compensation to the operating organization.

Completion of the first round of insurance inspections at Rosenergoatom facilities

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Russian Nuclear Insurance Pool

Organization of International Insurance Inspections (IIIs).

In course of International Insurance Inspections performance thee inspectors found a number of factors that are reflecting certain insurance risks, in the three main inspection areas: nuclear safety, operations, third part liability and fire protection.

The main approaches to assessment of insurance risks for a NPP:

- in-depth review of examinations of design and operational solutions, upgradings, OLE activities at the Unit;
- assessments of NS, RS, operating performance level by results of previously conducted inspections (IIIs, WANO PRs, IAEA missions);
- analysis of OO/NPP steps taken towards interaction with Russian Association of Nuclear Insurers (events, meetings, reports);
- assessment of plant equipment ageing and design life expiration;
- investigation into deviations from normal operation and abnormalities connected with human factor or subcontractor performance deficiencies;
- consideration of the statistics of deviations and abnormalities during the commissioning phase for new units as well as for existing units – after their modernization and OLE and after the overhauls.

New formats of International Insurance Inspections for the next five-year cycle:

- influence on insurance risks stipulated by human factors in NPP operation, ensuring NS, RS and operational reliability of units (there are needed elaboration of guidance documents, involvement of specialists for assessment and analysis of events and abnormalities, psycho-physiological examination of personnel);
- in-depth approach to assessments of risks from NPP property damage insurance;

- insurance of construction and installation works at new facilities (NPP units, icebreakers, Floating NCPP);
- acquisition of additional information, analysis, calculation of the engineering rating factor for nuclear power units;
- engineering assessments of insurable risks of the main nuclear plant equipment (process and electrical one) damage, with its actual residual life taken into consideration;
- challenges of M&R performance at the units by force of external maintenance organizations as potential beneficiaries;
- insurance inspections to new unit construction sites prior to their physical start-up for assessment of the risks arising from performance of construction and installation works, pre-startup adjustments of equipment and systems;
- NPP site visits to review any current operational problems and/or to investigate insured events, etc.;
- an enlarged list of inspectors from French and Japanese national nuclear insurance pools to be engaged to International Insurance Inspections at Russian NPPs.

30 years of OSART programme: Achievements, challenges and future evolution

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The IAEA-run OSART (Operational Safety Review Team) mission programme is the oldest world-wide peer review service. OSART is recognised and recommended for use by the IAEA General Conference, the Review Meetings of the Convention on Nuclear Safety and the Ministerial Conference of 2011.

The paper overviews OSART history starting from the first OSART mission to Kori NPP in 1983. Achievements OSART as a very powerful tool for harmonisation of operational safety based on IAEA Safety Standards are highlighted. Challenges such as development of new OSART guideline and WNO; revision of operational safety standards; improvement of the OSART methodology; implementation of the IAEA Nuclear safety action plan, coordination with WANO Peer Review programme, and human resource issues are discussed.

An overview of OSART future evolution is presented including new modules (corporate, OSART during construction), training of OSART reviewers, new OSART guidelines and a methodology that provides for a more directed and structured self assessment by the plant, use new IAEA Safety Requirements as basic criteria and Safety Guides as criteria to define depth of the review, and gap analyses.

Safety culture. Peculiarities of introduction in various professional communities and countries in Asian-Pacific region

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The safety culture (SC) concept was for the first time presented by International Nuclear Safety Advisory Group (INSAG) in the Summary Report of the Post-Accident Review Meeting on the Chernobyl Accident [1] in 1986. Later on, the concept was further elaborated in the reports on «Basic Safety Principles for Nuclear Power Plants» [2] in 1988, and «Safety Culture» [3] in 1991, as well as in a number of other IAEA documents [4,5]. In Russia the SC concept was introduced in the General Norms and Rules OPB-88/97.

Further development of the SC concept has resulted in an awareness of necessity of different approaches to its shaping for various professional groups related to NPP operation, NPP maintenance, physical protection of nuclear facilities and material. This list is not an exhaustive one and, to authors' opinion, needs to be complemented on the basis of an analysis of life cycle of nuclear facilities.

Shaping of the SC concept as well as establishment of a methodology for its introduction and development were influenced by the experience of European and American operating organizations and therefore were based on the general principles of Western culture and mentality. It seems that specificities of national mentality in the Asian-Pacific region (APR) countries, which formed mostly under a centuries-old influence of Chinese culture, unfortunately, give rise to a low level of safety culture, particularly as concerned to decision making and transparency in investigation of violations. This fact is clearly identified in the Report of the Fukushima Nuclear Accident Independent Investigation Commission [6]: «What must be admitted – very painfully – is that this was a disaster 'Made in Japan.' Its fundamental causes are to be found in the ingrained conventions of Japanese culture: our reflexive obedience; our reluctance to question authority; our devotion to 'sticking with the program'; our groupism; and our insularity.» Situation around the violations in course of equipment supply to NPPs of Republic of Korea (also a high-tech country) is one more example of deficiencies in safety culture.

In the paper presented the authors analyze the influence of professional environment and national peculiarities on the safety culture development process.

THE MATERIALS AGEING INSTITUTE

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The Materials Ageing Institute (MAI) was founded on the belief that sharing research, experimental results, feedback and scientific information on materials degradation can contribute significantly to the long term operability and life extension of nuclear power plants. Material Ageing Management is increasingly considered to be a key challenge worldwide for safe long term operation.

The MAI addresses this crucial issue from an applied research and development (R&D) perspective. By teaming up with utilities and related industries as well as academic partners, the MAI is able to combine operational expertise and theoretical knowledge, and to apply experiments and computer modelling to the understanding of the ageing process in materials and components. Since establishment of the MAI in 2008 by EDF, other nuclear plant operators and organizations have joined the Institute. Today, its eleven members include Kansai Electric Power Company (KEPCO, Japan), Tokyo Electric Power Company (TEPCO, Japan), the Electric Power Research Institute (EPRI, USA), EDF Energy (United Kingdom), CGN (China), Rosenergoatom (REA, Russia), Mitsubishi Heavy Industries (MHI, Japan), Central Research Institute of Electric Power Industry (CRIEPI, Japan), Areva (France) and the french Atomic Energy Commission (CEA, France).

The MAI's mission is to be an international center of excellence coordinating a worldwide collaborative R&D program, driven by utilities' needs, on materials ageing processes. It provides scientists and engineers with access to state of the art experimental facilities to develop and corroborate the most credible and up-to-date scientific knowledge that can be integrated into physical and engineering models providing utilities with predictive capability for degradation management and mitigation measures, required for long term operation of nuclear power plants. An extension of this mission is to share knowledge and offer the necessary training on material degradation issues to the current utility staff as well as the future generations of utility scientists and engineers.

International Cooperation to Increase Severe Accident Safety with Application of the IVMR Strategy for VVER-1000 Units

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After the Fukushima accident there is a strong requirement to increase safety and prevent severe accident consequences not only for new design NPPs, but also for units in operation. Proposed new retrofits are focused on several issues of severe accidents and one of key issues is localization of corium and its coolability. One possible strategy for melt localization is the IVMR (In Vessel Melt Retention). This strategy is already applied for lower output NPPs, up to 600 MW. Our presentation will describe international cooperation initiated with the aim to investigate applicability of this IVMR strategy also for existing VVER 1000 units and at the same time to develop guidelines for this strategy application for new reactors with higher power. At present this international cooperation has strong support not only from the UJV Rez institute, but also from IRSN, CEA, AREVA, and EdF just to name few of the key participants. Participation of Russian institutes like of AEP Moscow, OKB Gidropress, and KI Moscow is also under negotiation. At the beginning the effort started with very close and effective cooperation between UJV Rez and Kurchatov Institute in Moscow on the first analytical simulations of the melt formation and heat flux distributions into RPV wall. Detailed description of already performed and also ongoing or planned activities is a part of the contribution.

Organization of the aftersales service at VVER NPPs abroad

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The paper deals with issues of organization of the aftersales service at foreign nuclear power plants with VVER-type reactors.

At present, more than 70 nuclear power units with Russian-design VVER-type reactors are in operation worldwide. 9 nuclear power units with VVER-1200 are currently under constriction.

According to operational specifications for newly commissioned nuclear power units it is required to provide for carrying out, within the first 10 years of operation, a full complex of activities on maintenance and repair (M&R) of Safety Class 1 and 2 equipment in line with requirements set by Russian Rostechnadzor and by Rosenergoatom Concern OJSC standards.

The paper describes the main stages in organization of planning, preparations and conduction of M&R works at a shutdown power unit. Critical points in the power unit M&R management are identified based on the experience of the first mid-term repair conducted at Bushehr NPP (Iran) in 2014.

At present, a consortium of Russian enterprises has organized performance of integrated examinations and activities on assessment and justification of the plant equipment operating life extension at Metzamor NPP in Armenia and Kozloduy NPP in Bulgaria. There are ongoing works on organization of equipment modernization with the goals to increase rated power, improve safety and reliability of foreign NPPs with VVER reactors.

On the basis of insights from analysis of repair, modernization and operating life extension work performance at VVER NPPs in Russia and abroad, the authors propose a complex of measures aimed at organization of aftersales service at foreign NPPs with VVER reactors with involvement of Russian enterprises of SC «Rosatom».

Organization of supply of protected nuclear fuel of PWR-1000 reactors to the international markets

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Strategic goals of State enterprise «Rosatom» are aimed at achievement of global leadership of the enterprise in the nuclear power engineering field. One of such goals is the advancement of PWR process platform abroad accompanied by increase of provision of products and services of the Enterprise throughout the whole NPP life cycle [1]. However, increase of provision of nuclear services enhances the risk of the spread of nuclear fissionable materials (NFM) and technologies, which can be used for non-peaceful conversion. Non-availability of technologies of enrichment and processing of nuclear materials in the novice-countries results in necessity to transport fresh and spent nuclear fuel, therefore increasing the risk of nuclear material theft [2].

Organization of nuclear fuel supply on the basis of regenerated uranium is a way to increase the protection of PWR reactor fuel cycle from the unauthorized spread of NFM. Presence of even uranium isotopes ^{232}U, ^{236}U in regenerate leads to significant increase of the number of work units of uranium fission required for production of material of an explosive device, the increase of power of ionizing radiation dose, as well as generation of protected plutonium with the increased level of plutonium isotope ^{238}Pu (more than 6%) in the spent nuclear fuel [3].

Analysis of different strategies of PWR-1000 reactor switching to regenerated uranium oxide nuclear fuel is given in this paper. It has been shown that initial presence of 1% and more of ^{236}U isotope in the fresh nuclear fuel is a sufficient condition for accumulation of protected plutonium, and calculations of the time required for generation of necessary amount of ^{236}U isotope were carried out. The calculations have revealed that the formation of fuel loadings with required material for the strategy of recycling of uranium of one PWR-1000 reactor is possible after 20 years. For the case of full-scale involvement of the regenerate of accumulated spent fuel of PWR reactors, generation of protected fuel is possible starting with the first loading of reactor. Additionally, analysis of uranium sources economy for all considered strategies of PWR-1000 reactor conversion to regenerated uranium oxide nuclear fuel was carried out.

Results of assessment of seismic safety of Zaporozhskaya NPP

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Within the framework of execution of program of extension of service life terms of ZNPP power units in 2011-2014 a set of activities on assessment of seismic safety of the plant was performed. The research included the following works: works on general, specific and micro-seismic risk zoning; obtaining response spectrum at the surface of the site simulating seismic event of design basis earthquake (DBE) and safe shutdown earthquake (SSE) levels; calculation of seismic stability and floor-wise response spectrum with account for effects of interaction with earth foundation.

Procedure and results of obtaining of response spectrum at the surface of earth cover and calculation of seismic stability and floor-wise spectra performed with account for the data of additional research of seismicity of ZNPP site are given in the report.

Procedure of obtaining of response spectrum at the surface of earth cover of ZNPP site came down to the following stages:

1) analysis and comparison of response spectra given in regulatory documents (NUREG/CR-0098 and NP-031-01) and calculated on the basis of NGA models for the event of SSE level from remote area (Vranch area) and local area (Konkskiy rupture);

2) generation of accelerograms for the indicated response spectra;

3) accounting of local resonance conditions conditioned by specific features of geologic aspects and physical properties of the cross-section of ZNPP site was carried out by recalculation of accelerograms from the «the rock» to daylight surface by SHAKE program;

4) creating generalized response spectrum in the form of spectral envelope of all spectra with account for local resonance conditions.

Based on obtained data, calculation of seismic stability and floor-wise response spectra with account for foundation soil strain capacity of buildings and facilities of increased seismic stability category was carried out at the surface.

Calculation of seismic stability and floor-wise response spectrum with account for effects of interaction with earth foundation was subdivided into four stages:

1) building finite element scheme in calculation system Robot Structural Analysis Professional 2013;

2) calculation of equivalent dynamic characteristics of the foundation by the procedure for accounting of effects of interaction of foundation and the building suggested by prof. G. Gazetas;

3) development and calculation of simplified analytical finite element dynamic scheme of building for account of interaction in the system «foundation-building»;

4) calculation of three-dimensional finite element scheme subject to seismic impact, the f\data for which were obtained during previous stages of calculation, with account for interaction of the building and the foundation.

This approach allowed to take into account effects of interaction of the building with the foundation.

Calculation of seismic stability and floor-wise response spectrum was carried out by way of direct dynamic analysis of three-dimensional finite element scheme subject to seismic impact of specified intensity in three directions. Initial data were accelerograms at the level of foundation bed with account for interaction of the building and the foundation.

RESULTS OF EXPERIMETAL CALCULATION RESEARCH OF PROCESSES OF LIGHT GAS MIXING FOR THE PURPOSES OF MANAGEMENT OF ACCIDENTS AT NPPS WITHIN THE FRAMEWORK OF ERCOSAM-SAMARA PROJECTS

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At development of severe accident at the NPP with pressurized water coolant, due to water radiolytic decomposition, chemical reactions with reactor core materials, hydrogen degassing from primary circuit, and other processes, hydrogen ingress into containment volume is possible. When there are certain concentrations of hydrogen and its stratification in containment volume, there is danger of emergence of explosive hydrogen-containing mixtures in the systems, elements and rooms of containment.

Within the framework of European Community Program «FP7-Fission-2009» experimental calculation research of possibility of stratification of hydrogen in containment as per scenario reproducing the sequence of events at severe accident development with hydrogen release at light-water reactor and impact of severe accidents management means (sprinklers, coolers, recombiners) on distribution of light gas is being performed. Research is carried out by two consortiums: EAEC and State enterprise «Rosatom» – within the scope of two simultaneously carried out projects ERCOSAM and SAMARA and includes 11 organizations-participants out of 7 countries. Experimental research is being performed at four experimental units: TOSQAN (IRSN, France) with 7 m^3 reservoir capacity, MISTRA (CEA, France) with the volume of containment simulator of 97,6 m^3, PANDA (PSI, Switzerland) with the volume of containment simulator of 183,3 m^3 and SPOT containment (**JSC «OCBM Afrikantov»**, Russia) with the volume of containment simulator of 59 m^3. Calculation-analytical works include planning of experiments, pre-test and after-test calculations, which are carried out with the use of codes in lumped parameters and CFD-codes. Participation of **JSC «OCBM Afrikantov»** in the international project is planned in the form of performance of two experiments at SPOT ZO installation and execution of numerical modeling of experiments with the help of KUPOL-MT code.

Experimental research was carried out with adherence to preset initial and boundary conditions, which were determined with account for scaling of containment model volume to the volume of plant containment. As per results of experimental research, stable stratification of steam-gas mixture along the height of containment model was obtained. Operation of heat-exchanger of system of heat removal from containment with the specified capacity does not provide complete homogenization of the atmosphere along the full height of containment model, above heat exchanger stratified layer of helium and steam remains over a period of full time of operation of heat exchanger.

Pre-test and after-test computational modeling of experiments at installations TOSQAN, PANDA, MISTRA и SPOT ZO were performed by «KUPOL-MT» code of lumped parameters. Computational models of installations allowing to evaluate stratification of components of steam-gas mixture in the volume containment model with account for stream flows were developed. Performed analysis of results of pre-test and after-test computational modeling has revealed that KUPOL-MT allows to perform preliminary qualitative assessment of thermal-hydraulic processes in containment at development of severe accident with hydrogen release and account for operation of sprinkler, cooler, recombiner.

The work has been performed within the framework of ERCOSAM-SAMARA project. ERCOSAM project (Contract № 249691) is being

performed under the Framework Program FP7 of European community. SAMARA project is carried out with the support of SE «Rosatom».

Application of method of improved estimation for analysis of unintended closing of FSIV at NPP «Dukovany» at unit power increase

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Unintended closing of FSIV at steam lines is a limiting design accident (normal operation violation) from the point of view of pressure peak in the secondary circuit of NPP «Dukovany» with PWR-440 (combined method was used for analysis – estimation program of the best estimate and conventional initial and boundary conditions). Already at the first stage of power increase (to 105%) received peak pressure values in SG were nearing the value of acceptance criterion, in preliminary estimation for 108% power, received peak pressure value exceeds limit value.

Safety analysis for this initial event related to normal operation violation is carried out in accordance with requirements of Czech supervisory authority without account for operation of those safety systems of normal operation which improve process execution. This approach has led to such result that at relatively high probability of initial event, the probability of analyzed scenario significantly decreases, and obtained results are very conventional. Therefore, it is reasonable to apply the best estimate method, the results of which indicate performance of criterion of secondary circuit pressure, for analysis of unintended closing of FSIV at steam lines. Results of the estimate the best estimate method and comparison with calculation by combined method are presented in the report.

Section 5

HUMAN RESOURCE FOR NUCLEAR POWER

Subsection 5.1
TRAINING OF YOUNG SPECIALISTS FOR NPPS

Innovative model for young professionals training for nuclear power

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High level requirements for the personnel of nuclear power plants significantly complicate and extend the procedure for preparing of a graduating student to independent work . The job placement process for Rostov NPP may take up to six months or more. During this period, the graduate's life plans may change from shifting to another industry to change of residence .

The proposed innovation model significantly reduces the time of job placement of young specialists at the nuclear power plants due to the transfer the checkup activities, preparing of necessary documents and individual training programs for a job position to the period of study at the university.

The model assumes the addressing of the following tasks:

1) improving the students' performance on core and general education disciplines up to the to the average mark of 4 or higher;

2) system collaboration between Rostov NPP and VITI NRNU MEPhI in organizing and conducting practical training of the students to ensure their in-depth study of NPP equipment and technologies ;

3) active involvement of Rostov NPP employees to conducting classes for undergraduate students on the core subjects using educational resources of the Training center and opportunities of the base department "Nuclear Power Plants";

4) compiling the list of issues to be reviewed in the final qualifying works to address the technical challenges of structural divisions of Rostov NPP ;

5) review and adjustment of educational programs for students training by the heads of functions of Rostov NPP;

6) employment of graduating students in the shops and departments at Rostov NPP at working positions prior to pre-graduation practical training.

Innovative model, developed with the active participation of heads of shops and functions at Rostov NPP was approved at an expanded meeting of nuclear specialists and representatives of VITI NRNU MEPhI held at the nuclear plant and chaired by the plant director A.A. Salnikov.

For its implementation in November 2013 63 graduating students from VITI MEPhI were assigned to shops and departments. The event was hosted by the Training center of Rostov NPP and attended by the heads of structural departments and the heads of the university departments. After the assignment the documents started to be prepared for employment of students for the working positions. Students have successfully passed the examinations and were directed for pre-graduate practical training in the relevant structural units of the power plant. Now, during the graduate designing period, the students, being permanent employees of Rostov NPP undergo individual training program for the job position, and by the end of their study the checkup activities would be completed and all the necessary exams be passed this enabling them to start working in the respective job positions.

Testing of the proposed model shall allow Rostov NPP to modify it to suit the individual requirements of individual departments and divisions of nuclear power plant with the subsequent wide use in affiliates to “Concern Rosenergoatom“. With the concluded cooperation agreements signed with Leningrad, Smolensk, Kursk, Kalinin nuclear power plants VITI MEPhI could become a platform for implementation of the proposed innovative model of training professionals for nuclear power engineering.

Arrangement of training practice at Kalinin NPP: project «8th term»

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- Esteemed profession, career promotion, professional actualization - the goals to be pursued , especially at the very start of professional life.
- Currently the plant actively collaborates with more than a dozen universities across the country.
- The students of Ivanovo State Power Engineering University which is, by the way, one of the main sources of manpower for the nuclear industry in the country, spend the eighth term in the plant Training center.
- Arrangement of the students’ practical training takes place in two stages: the first stage is a three-week course in the Training center , and the second stage – mastering the practical skills in the shops.
- Example schedule of practical training for students for the year of 2014 .
- The students attend lectures delivered by the employees immediately involved in the production process.
- At the end of a three-week “term” the students take an exam at Kalinin NPP to the board of teachers - employees of the plant.
- Dialogue with the management.
- The second stage - the practical training in the shops under the supervision of specialists from reactor, turbine, nuclear safety functions .

- Accommodation conditions and leisure opportunity.
- Problems and possible solutions .

Program of advanced development of human resources "From the new school to the workplace" OJSC "RusHydro"

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The project "Program of advanced development of human resources" From the New School to the workplace " aims at creating an institutional environment for involvement and training of engineers for hydropower . The prerequisites for the program development are: the need for supplying human resources in view of commissioning of 22.2 GW of new capacities by "RusHydro" by 2020 with the consideration of the domestic problem of "lacking" of engineering personnel "," aging "of engineering and technical personnel , the low popularity of engineering professions with the youth when choosing a higher education institution , as well as a number of existing problems of the education system, related to the quality of engineering education at the current moment. The surveys conducted in a number of schools show that preserving the traditional approach to teaching the students leads to the loss of capacity for the development of engineering capabilities by the end of the upper school of practically 2.3 times in relation to intellectual structures and of 1.3 times in relation to creativity .

In accordance with the Russian innovation strategy 2020 and in order to implement the technology there was established and implemented a permanent system of "Corporate lifts" that enabled to establish the basis for relationship of the professional community and the educational community, and the mechanisms for short-, medium- and long-term needs of the company for the young engineers.

The program is an example of the implementation of innovations in the field of education, including the establishment of effective learning technologies and organization of the educational process. The unique nature of the projects is in the formation of new innovative institutional arrangements for relations with educational institutions in the development of the practical orientation of educational programs , introduction of innovative educational technologies and modern interactive teaching methods , the formation of a corporate system of continuous education from elementary school to the workplace using the competency approach . Application of competency and problem- orienting approaches in practice forms the engineering vision of the world : the ability to invent, understand and master of the new, to find solutions in difficult situations. The main outcome of the system is to ensure the admission by the Company not only professionally trained specialists according to the Company's requirements, but also the young people concerned with the values of the profession of hydraulic power engineer.

The program allows to closer bring together the education and the labor market, improve the quality of engineering education, resolve a number of challenges of the entire education system, and to ensure the professionalization of students and shape the future generation of engineers able to handle the task of modernizing the economy of the country set by the government. Systematization and replication of technologies and practices that motivate the school leavers to choose technical educational institutions for professional education and create the conditions for competencies needed in strategically important sectors of the economy.

Program is called for by teachers of schools, universities and vocational colleges. It has been awarded by the Ministry of Energy of the Russian Federation in the category of " Innovative approach to training of engineering skills ." Thanks to the use of this technology the competition for specialty profile for the first 4 years from 2009 to 2012 has increased by 4 times , the flow of the young, skilled, motivated personnel has increased by 1.4 times.

Role of instructor in ensuring NPP safety

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In operation of nuclear power plants (NPP) the priority goal is such production of electric power where the life and health of people and the environment do not suffer irreversible damage. To achieve this goal, there is a set of activities , including technical improvements and upgrading with the introduction of the newest developments and application of advanced technologies , and organizational component which includes the implementation of safe operation and maintenance in accordance with the existing rules and regulations in the field of nuclear energy.

In line with the document "Organization of work with the personnel at nuclear power plants " the plant Training center (TC) conducts training for job position and maintains the qualification individually or in group using the programs meeting the relevant standards of the operating utility.

Training of personnel for job position provides professional knowledge and practical skills and is conducted using the training programs for job position. Training of plant personnel in TC has the following stages: theoretical training , practical training , on-job training , the primary test of knowledge , shadowing, admission for independent work. The control room staff prior to admission to the shadowing also needs to get permission from Rostekhnadzor to carry out work in the field of nuclear energy application.

Maintaining plant personnel qualification is conducted to preserve and develop of knowledge previously acquired by the staff, skills and competencies required to perform their duties specified in their job descriptions and qualification characteristics and regulations on occupational safety.

In developing the educational process a systematic approach to teaching applies - phased, logical organization of training of plant personnel , starting with the identification and analysis of training needs, training planning, development of training programs for the job position and programs for maintaining qualifications, teaching materials and educational aids, conduct of training and assessment of training completed. Systematic approach methodology through its inherent possibilities of application of standard procedures and processes at any stage of training, ensures high quality of the education process , and enhanced monitoring and reporting enable to reliably confirm the qualifications of the personnel to the requirements for the safe operation of nuclear plants.

The role of instructor in the educational process is to help the student to acquire, maintain and develop their skills, knowledge, practices , attitudes and competencies required to perform their duties through applying the necessary and sufficient number of teaching materials and educational aids.

New approaches to simulator – based training of operating personnel

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1. Training center of Balakovo NPP: historical background, training and technical resources, organizational structure.
2. Professional training: requirements , directions, the implementation, components, training and methodological support.
3. Training of operating personnel: program for preliminary training, the use of the severe accident module and “Guidelines for severe accident management,” using FSS/AS for maintaining qualification of “field” operators.
4. FSS-based simulator training: general principles of organization , forms and stages of simulator training, the training procedure, post-training critiques procedure, assessment of operator actions on the simulator, responsibility.
5. Simulator –based training using SVT (situational and visualized training).
6. Forms of simulator training.
7. Comprehensive simulator training.
8. Assessment simulator training.

9. Operator skills of trainees and their performance criteria.
10. Annual training of MCR operating personnel , shift supervisors
11. Efficiency issues.
12. Influence of personnel professional training on the number of unit shutdowns.
13. Improvement of training process.

Up-to-date hardware for simulation modeling, information simulators and virtual reality to improve the effectiveness of training of young professionals for nuclear power plants

K.I. Ozerov, P.A. Bunto

This report overviews the up-to-date hardware for simulation modeling, information simulators and virtual reality to improve the effectiveness of training of young professionals for nuclear power plants developed by CG "NEOLANT."

The events at "Fukushima-1" nuclear plant in Japan proved it necessary to revise the approach for training and re-training of personnel at nuclear power plants . The application of up-to-date teaching techniques which involve the use of three-dimensional simulation models, on-line simulators and virtual reality systems can significantly improve the quality of training / re-training of plant personnel.

The recent application of three-dimensional simulation has deeply rooted itself in the production process at all stages of plant life cycle from designing to closing down of the site.

Implementation of computer-aided teaching systems based of three-dimensional modeling enables to train personnel more effectively starting with the students being educated at the institutions of higher education and finishing with the personnel of the nuclear power plants being trained / re-trained in the Training centers.

CG "NEOLANT" is engaged in creation of information databases on decommissioning. At present, the information systems have been developed for the following nuclear power plants:

- Leningrad NPP – the industrial sites of Stages 1 and 2, the internal content of the power units 1 to 4 ;
- Bilibino NPP – the industrial site and internal content of the power units 1 to 4;
- Kursk NPP – the industrial site and internal content of the power units 1 to 3;
- Smolensk NPP – the industrial site and internal content of the first power unit;

- Smolensk NPP – the industrial site and internal content of the power units 1 and 2;
- Novovoronezh NPP – the industrial site and internal content of the power units 1 and 2;

The simulation models were developed by CG ”NEOLANT“ to train the operators for robot-aided devices and verification of performance of the complex technological processes.

Simulation model for dismantling of reactor structures for AMB-100 reactors at Beloyarsk NPP is designed for mastering the technology of reactor dismantling for AMB-100 reactor structures. Simulation model contains a three-dimensional model of the cavity for AMB-100 reactor, as well as equipment and structures used in the process of dismantling.

The simulation model simulates the operation of the mobile robot-aided device (MRD) BROKK 90 and its attachments .

Thanks to the physics of solids implemented in the simulation model , the simulation model enables to maximum reliably handle various technological operations .

Additionally to technology verification, the simulation model can be used as a training simulator for training of operators for MRD BROKK thanks to the MRD control process accurately implemented in the simulation model and video surveillance system which is designed to monitor the work of MRD . Additionally, simulation model allows to simulate various emergency situations , which provides operator training to response to emergencies .

The simulation model for dismantling of metal structures of under-reactor space between the schemes “O” and “P” of reactor PUGR AB -1 became the further development of the simulation model for dismantling of reactor structures for reactor design AMB-100 at Beloyarsk NPP.

This simulation model made it possible to verify the technology of dismantling of metal structures of under-reactor space between the schemes “O” and “P” of reactor PUGR AB -1. In future, this simulation model can be used for training of operators MRD BROKK 60 on dismantling process .

To ensure timely and adequate personnel response to plant emergency situation it is very important for the employees to get the understanding of the process of emergency situations development at the nuclear power plant. Software systems that simulate the development of emergencies at nuclear power plants were developed for this purpose.

In collaboration with the Institute of Safe Development of Nuclear Power Engineering a software module has been developed that visualizes the process of development of radioactive contamination in the air during the accident at nuclear and radiation hazardous facility . It allows the students to evaluate the contaminated area , which is dangerous for people location, and to understand the escape route to be taken in the event of a particular accident.

The further development of this module was the development of simulator for propagation of gas and aerosol emissions with radioactive substances in areas of location of nuclear and radiation hazardous facility and the actions of quick-response teams during the development of an emergency situation.

This software package enables to simulate the propagation of radioactive contamination in NPP surveillance area, the response of ARMS sensors, the process of notification of the concerned departments, as well as the consequences of the accident for the population living in the plant surveillance area.

Using of the visualizer will allow the specialists to evaluate the radiation doses for the population as a result of the accidents at nuclear power plants and to develop measures based of this data to protect the population .

A simulator, which simulates the progress of emergency situations inside the plant unit, is developed for the purpose of plant personnel training . It simulates the response of personnel and automated systems in the course of emergency development as well as displays the dynamics of changes of the most critical parameters at the reactor plant and other systems.

An important part of training is the mastering of skills for working with the sophisticated equipment. For the purposes of learning the working process for the nut-driver with the simultaneous extraction of studs of main joint for VVER -1000 reactor the simulator was developed to control the nut-driver with simultaneous extraction of studs of main joint for VVER -1000 reactor.

This simulator models the work station for the central board operator to control the nut-driver. It enables to master the sequence of operations during mounting /dismounting of the reactor main joint which allows to improve the level of experience of the personnel working with the nut-driver .

For plant personnel training the software package of Interactive electronic operating manual " Isolation device 484" was developed.

This software package enables to study the structure of the Isolation device, its technical specifications using a three-dimensional model. Additionally, the software package enables to demonstrate on the model the process of mounting / dismounting of Isolation device 484 with indication of the most important issues. The testing system , built into the software system allows to check the knowledge of the specialist undergoing training.

The application of the software systems using three-dimensional modeling in the teaching process of students in higher education institutions will enable:

- Shorten the period of adaptation of the young professionals to the working environment ;
- Shape and foster the safety culture as early as at the education stage;
- Train a professional possessing skills and competencies.

The application of the software systems using three-dimensional modeling in the process of training and re-training of the personnel on the basis of Training centers enables:

- Significantly reduce training time through more extensive use of visual approaches and clear presentation of material;
- Significantly improve the basic knowledge of employees;
- Allows to master the skills of equipment operating under normal and emergency situations;
- Allows the professionals being trained to understand more clearly the development of emergency situations and response actions to be taken in such situations.

Training conducted within the framework of practical arrangements with the IAEA: lessons and experience gained by Rosatom central institute for continuing education and training

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The present article reviews the activities conducted by ROSATOM Central Institute for Continuing Education and Training (ROSATOM CIPK) in the framework of the triple Practical arrangements signed between the IAEA, OJSC "Concern Rosenergoatom" and ROSATOM CIPK at 55th Session of the General Conference of the IAEA in 2011 . These Practical arrangements were signed in order to promote the joint initiatives in training of human resources, education and personnel training for the development of nuclear infrastructure in the countries which have embarked on the creation of new or expansion of existing nuclear power program . It is a recognized truth that at an early stage of development of the national nuclear power program the IAEA plays a dominant role in supporting of newcomer countries. Within the framework of the above-mentioned Practical arrangements, IAEA, inter alia, provides financial and administrative support for scientific visits to ROSATOM CIPK of specialists from newcomer countries.

Based on experience gained in organizing such visits since 2011 , this article formulates some recommendations for national stakeholders involved in the creation of nuclear power programs (primarily for NEPIO).

The highest priority is to study the structure and staffing schedule of the nuclear power plants having the specifics in each country providing the technology. This will enable to accelerate the self-assessment of development of the national nuclear infrastructure and to compile an integrated working plan for training of human resources, including training schemes for key operating personnel in the vendor country.

Related to this stage is the formation (at the earliest phase of cooperation of the provider country and the recipient country) of the joint working group to prepare a road map for the development of human resources for the operating utility. Of special attention should be the establishing of the integrated information and training center at the NPP site which should be commissioned simultaneously with the first concrete poured for the foundation of the reactor building . This Centre should become a training ground for training of local contractors at the stage of NPP construction and equipment installation.

ROSATOM CIPK has developed a goal-oriented course for NEPIO managers "Initiation of new nuclear power programs", which takes into account the lessons learned from the training of specialists of the national nuclear infrastructure in the newcomer countries and this course is offered as a basis for the creation of a joint working group to develop an integrated working plan in cooperation with the IAEA.

Organization of adaptation of young professionals at Kola nuclear power plant, affiliate to "concern rosenergoatom"

S.V. Berezyuk

Kola Nuclear Power Plant, the town of Polar Zori

Kola NPP leadership pay proper attention to the adaptation and development of young professionals.

For the purpose of adaptation, professional promotion, securing the footing at the plant, increase the engagement of young professionals, Kola nuclear power plant arranges a number of event.

1. For the purpose of acquaintance of the young professionals with the production , their psychological adaptation, in place there is "Young Professionals' School" held 2-3 times a year, whenever the young professionals are admitted to work. The program is approved by the plant director. Duration: 1 week (40 hours).

The professionals from various fields deliver lectures to the newcomers on the following topics:

- adaptation, assessment, development opportunities, professional ethic principles and company-wide values;
- visit to the Information Centre, to the production units ;
- repair and maintenance issues ;
- technical and economic indicators of Kola NPP;
- organizational and functional management structure;
- social policy;
- legal issues;
- principles of safety culture;
- radiation safety rules and environmental protection;
- basic circuit and electrical diagram;

- meeting with the members of Young Nuclear Engineer Community, brief tournament , “What? Where? When ?»;
- meeting with Chief engineer and Deputy director for human resource.

“Young Professionals’ School” was included in the “Simple success stories” collection , issue №2//2013 as a the best practice for adaptation and training of young professionals.

2. To develop competencies and engineering capacity after the assessment of the level of competencies development in the form of a business game “Atomic tycoon“, a young professional together with specialists from staff development department and instructor develops his/her Individual development plan which includes actions to develop the competencies, as well as an individual assignment (“Resulting report“). The results of fulfillment of personal assignment are reviewed by the Board chaired by Chief engineer. Young professionals whose resulting reports are the best take part in the competition of scientific and technical reports of young professionals at Kola NPP and have a priority right to participate in the industry and corporate conferences and fora.

These described activities help young professionals to unite , get acquainted, get a complete picture of Kola NPP in a prompt manner and in an interesting format and smoothly adapt to the plant environment. And what is still more important, to immediately feel themselves involved in the common business.

Current trends in training professionals for the nuclear industry training at the themal and nuclear power institute of NRU “MEI“

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History of training the professionals for the nuclear industry at Moscow Energy Institute (NRU “MEI“) is longer than half a century. Currently, the education of students in the interest of SC “Rosatom“ and related companies is mainly limited by the framework of the Institute for thermal and nuclear power NRU “MEI“ (ITAE). ITAE maintains training programs for bachelors and masters in the areas of “Thermal energy and heat engineering“ and “Nuclear energy and thermal physics”. Directly involved in “Rosatom” activities are almost all ITAE departments which graduates handle all operational matters and NPP design , the secondary circuit, successfully resolve physical and technical problems of nuclear and thermonuclear energy, work in the area of plant safety and automation . Thus , with a total number of ITAE first-year students in 2013 amounting to 300, the same year more than 60 graduates of NIU “MEI“ were admitted to work in SC “Rosatom“.

It is natural, that ITAE system of training is based on the traditions and experience of previous generations , retains its fundamental nature, but should meet the new challenges of which the key challenges are the following:

- the final transition starting from the year 2011 to adoption of bachelor course-master course training scheme omitting the possibility of obtaining a specialist certificate of degree.
- using FSES standard instead of SES-2000. A novelty is also the division of bachelor course and master course programs into applied and academic trends in an attempt for the universities to train highly qualified workforce.
- the requests for graduating students generated by employers.

To the external conditions can be attributed still more apparent shortage of highly qualified engineering staff caused by general trends in society and the prevailing bias towards humanitarian, managerial and economic specialties in education of professionals.

At the same time, the imperfection of current educational programs should be recognized. In the transition to a two-tier system, the natural desire of the academic community (not only in Russia but also in Europe) was the maximum saturation of the bachelor training program, an attempt to accommodate a 5 –year specialist curriculum for a 4-year training. All this has led to an oversaturation with the material of bachelor training curriculum and a kind of “thinning out“ of master degree programs.

Under these circumstances, the obvious tasks of higher educational institutions to improve the quality of training, compliance with all the basic qualities of modern requests: fundamentality; individuality; adaptability and interdisciplinary of education shall be:

- development of new state educational standards at all levels of training, taking into consideration (and partially, taking a leading part) all the modern requirements to higher education and correcting the shortcomings of previous generations of the FSES standard and development on their basis of educational programs and working plans for disciplines;
- expansion of the variable part of programs, providing to the students a real choice of disciplines formed with consideration for the requirements of the industry;
- increase of bases for possible training practices of students, involvement of corporate employees to the educational process.

Psychological support of young professionals training for NPP

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Issues related to students’ training for nuclear energy and industry in the Research University in the current conditions are addressed: underdevelopment of entrants’ professional intentions; the lengthy and complicated adaptation of students to higher school education; inadequate level of learning and professional motivation and lack of progress in studies of the first-year students; student's computer- and Internet- addiction; inadequate psychological and pedagogical competence of the teaching staff and also the issue of contents of programs for new disciplines, and a number of other problems. The package of proposals for psychological and pedagogical support for the resolution of the specified problems is presented .

Key words: higher education, research university, personnel training for nuclear power plants, psychological support, competence.

Engagement, selection and adaptation of young professionals at Leningrad NPP

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Politics of Leningrad NPP in the area of personnel management is determined by the strategic goals and ambitious objectives, which challenge today the Russian nuclear industry. For this purpose, Leningrad NPP introduces and implements innovative approaches to human resource management, which -while maintaining the traditional for the entire nuclear industry high degree of social security –focuses on improving the efficiency of each employee, the structural unit and the station as a whole.

An important source of managerial, engineering and technical personnel for Leningrad NPP are the young professionals – graduating students from the core universities. Successful adaptation of young professionals is the first step in their professional development and career promotion , improvement of their performance. To facilitate the adaptation of newly hired workers, including young professionals, in place at the plant there is a developed Program for adaptation of new employees - a set of measures aimed at accelerating the process of adaptation to the working conditions and the specifics of nuclear power plant, at a smoother adaptation to the working environment, reducing the number of errors associated with the start to work, shaping of the positive image of Leningrad NPP, OJSC "Concern Rosenergoatom" and State Corporation "Rosatom", as well as at assessment of the skill level and capabilities of employees during their evaluation period. The affiliate conducts a regular monitoring of the effectiveness of the current system of adaptation. In assessing of the effectiveness of the adaptation program the focus directed not only on the development of various performance criteria (subjective and objective), but also on the analysis of their effect on in-house processes, i.e. assessment allows to understand the interdependence of the results of adaptation of young professionals, young workers and Leningrad

NPP activity through the formation of loyalty , involvement and understanding of the strategic objectives of the affiliate and the Concern.

Teaching of the students oriented to join the nuclear industry of the fundamentals of safety culture

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In their report , the authors review the basic approaches to determination of the direction of development of the safety culture among students at the university – oriented at training of the manpower for the nuclear industry. These approaches are determined in accordance with the general approaches to the definition of safety culture at nuclear power plant operation , with such competences in safety culture , which should be formed for the staff, with the Concept of educational policy in the field of safety and with those conditions in which the process of education in the high school is developing now.

Experience gained by the university in this area has enabled the authors to identify the practical methods of fostering safety culture commitment with university students , such as the methods of organizing and conducting classes with the statement of issues , organization of the appropriate information flows, contributing to the development of safety culture at the individual level regardless of their content; active methods of conducting classes , trainings to master the necessary actions and forms of an adequate responsible behavior of students demonstrating their commitment to safety culture.

A special emphasis in teaching the fundamentals of safety culture at Volgodonsk engineering and technical institute MEPhI, engaged in training the manpower for nuclear power complex of the Russian Federation in accordance with the needs of the nuclear industry, providing quality advanced training for nuclear power plants, is placed on a new discipline “Safety culture“ which is added to all curricula and for all majors. In this report the authors present the distinctive features in the selection of the content, forms and methods of teaching of a discipline.

Innovative approaches to training of professionals to ensure safe operation of fast breeder reactors

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Today, the prospect of the development of nuclear energy is associated with nuclear technology, based on the "fast breeder" reactors and on the closed fuel cycle.

One of the main conditions for successful implementation of this innovation in development of nuclear energy is the availability of highly qualified specialists in the operation and maintenance of systems and equipment at the plants with fast breeder reactors.

The importance of the human factor in ensuring nuclear safety is no less important than the equipment reliability.

The important role in providing professionals for nuclear industry is played by the so-called regional higher education institutions located in the same region with the power plant and providing for the plant the basic contingent for recruiting young professionals.

University department " Nuclear power plants and renewable energy sources" (before 2012 - "Nuclear Energy ") of Ural Federal University has 53 years of experience in training the professionals for the nuclear industry and traditionally focuses on training of professionals for nuclear power plants with fast reactors. The unique nature of the fast reactor technology requires specific logistics for training the professionals of this profile.

The university department has unique equipment, laboratory benches, simulators for training of professionals for units with fast breeder reactors. The important role in their creation, additionally to the university department specialists, was played by Beloyarsk NPP, OJSC "Concern Rosenergoatom", design and engineering organizations having the experience in this area.

An important role in improving of the efficiency of the educational process at the university department is played by integration of a leading professionals from Beloyarsk NPP, organizations and businesses ensuring the creation and maintenance of equipment for fast breeder reactor. A number of employees from OJSC "Concern Rosenergoatom" and its affiliate – Beloyarsk NPP - are employed part-time at the university department.

Innovative area of training at the university department of "Nuclear Energy" is the organization of the educational process on the basis of real-life problems of nuclear power with fast-neutron reactors. An important condition for the successful implementation of this activity is collaboration of university department with Beloyarsk NPP, OJSC "Concern Rosenergoatom" and other enterprises and organizations of Rosatom.

The subjects of research projects of students from this university department relate to the place of their future work. For the graduating students , this leads to a shorter adaptation period at Beloyarsk NPP. Currently the practice in underway to admit the students for job positions at the plant during their practical (pre-graduation) training with taking exams for the job position, etc.

Training of young professionals for dismantling works during decommissioning of nuclear power units

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Dismantling of equipment and metal structures is a mandatory procedure at the final stage of the life cycle of nuclear power plants (NPP) . In this country, the units of the first stage at Beloyarsk and Novovoronezh NPP are shutdown. This and the next decade another 28 units built in the Soviet era shall be shutdown due to expired service life. Decommissioning is a complicated from technological point of view and economically costly process to be implemented in radiation hazardous conditions. Significant portion of work relates to dismantling of equipment and metal structures.

To the development of dismantling technologies should be invited the professionals having professional competences in this area. For their training the educational organizations engaged in training for core specialties (areas of training) need to revise the basic educational program and supplement them with the new disciplines or allocate appropriate sections in the available educational programs.

Additionally, the development of educational and methodical materials is becoming urgent to address the issued of dismantling technologies and resources for their technological equipment. These training materials are required both for education of students whose majors are related to NPP design and technical operation, and also for training and re-training of professionals being already the plant employees and engaged in this activity.

This report presents some considerations for the design and implementation in the educational process at the university of educational and methodical materials related to this area. . These considerations are based on the experience of their use in the educational process at VITI MEPhI. Currently, the authors to this paper are involved in creating of a textbook addressing this subject. Preliminary, the textbook is structured as follows:

- introductory part, presenting the modern views on the issue of decommissioning of nuclear power units , implementation options for this final stage of their life cycle , including the option adopted for Russian nuclear power plants;
- general part covering the issues of regulation at the decommissioning stage, development of program and decommissioning project , the comprehensive engineering and radiation survey and management of radioactive waste arising from dismantling works;
- main part presenting in more detail than the above the materials for the development of equipment dismantling technologies, namely:
  a) the use of problem- oriented design system;
  b) preparatory works;
  c) providing on-site radiation safety;
  g) methods and techniques of equipment dismantling and breaking;
  d) use of technological equipment;
  e) transport operations.
- final part giving the overview of experience of the demolition work at domestic and foreign nuclear power plants.

Identification of the principles of relationship of data for psychological and physiological examination (PPE) of young nuclear professionals from main functions with their professional promotion

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Kalinin NPP, Udomlya

Introduction

Facilities using nuclear energy are potential sources of industrial accidents with significant negative environmental impacts and human consequences.

In order to ensure professional reliability of plant personnel , and thus to prevent the troubles in the operation of nuclear power plants associated with the improper actions of the employees, the plant workers undergo mandatory preliminary psychological and physiological examination at admission to work which then is repeated periodically on annual basis.

In forming their final opinion the psychology professionals consider the relevant features of motivation of job seeking candidates, the features of their cognitive abilities , personality traits , stress immunity , adaptive capabilities, etc.

The admission of every newly hired employee involves spending plant material resources (salary, training), time (including time of more experienced workers, managers). To justify the personnel training costs, it is appropriate to have a forecast not only for its professional reliability, but also for its professional promotion as early as at the stage of its recruitment. Additionally, the forecast for professional promotion of a young professional is a useful instrument to be used by the heads of departments in the planning of personnel promotion to the higher positions.

The research goal: to identify the statistically reliable interconnection between the data for psychological and physiological examination of young nuclear professionals with their professional promotion at Kalinin nuclear power plant.

The research objectives:

1. Determining of the sample volume and criteria of time interval for professional promotion.
2. Statistical analysis of data.
3. Compiling the recommendations on the use of PPE data.

Hypothesis: there is a relationship between the data for psychological and physiological examination (PPE) and the promotion of young professionals.

Research object: young professionals -152 persons.

Within the framework of research the data was collected from the preliminary PPE of young professionals during the years from 2002 to 2011. The data obtained was statistical analyzed using the Mann – Whitney test criterion (pair-wise comparison of the elements of the first sample with the elements of the second sample for independent samples).

Findings

The hypothesis put forward in this research was confirmed – identified is a statistically valid relationship between the data of preliminary PPE and the promotion of young professionals as well as their career development.

1. The professionals achieving the distinct career goals, professional promotion are distinguished, on the one hand, by their tolerance and reticence, the commitment to comply with the existing rules and regulations, and, on the other hand, by the necessary activity. Developed are such qualities as optimism, self-esteem , stress immunity and expressed intellectual interests.

2. Statistically significant were differences identified by the following PPE procedures:

- MMPI - 2 scales - pessimism, 4 - impulsivity, 6 - rigidity, 7 - anxiety, 8 – individualism.
- «16-FLO questionnaire " (Form C), B - high intelligence , C - self-control, E - dominance, L - suspiciousness, Q4 - tension.
- Raven graphics test

3. The results obtained in this effort will allow PPE psychology professionals, heads of functions as early as at the stage of their recruitment to predict the young professionals' adaptation to work at the plant, the peculiarities of their professional promotion and development.

4. This study not only resulted in scientifically justified need for psychological training, drills based on conventional topics (development of communicative qualities , teamwork , conflict and stress management), but also identified the need for development of self-control skills of plant personnel.

5. It is recommended to attract the attention of the managers to the appropriateness of considering the pre-PPE data in hiring young professionals, as well as in planning their promotion for the higher-level job positions.

Development of human resources: PROS and CONS of ECVET approach in comparison with the russian professional standards

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Globalization raises various challenges in the field of human resource development, thus leading to the need for harmonization and, as a consequence, the study of systems of training and continuing training of the staff.

Within the framework of the European credit system for the vocational education and training (ECVET) general requirements are being developed for the qualification, knowledge and skills of people related to nuclear infrastructure, which leads to higher degree of confidence, transparency and mutual recognition of results of professional training of the EU Member States and associated countries. Main ECVET policies are focused on development of the European passport - a package of documents to be uses for the relevant description of qualification and to be recognized by all potential employers in EU countries.

The basic document in Russia defining the requirements for the personnel knowledge and skills is the Uniform qualifying directory for the positions of managers, professionals and employees.

In this paper a comparative analysis is made of European and Russian approaches to training and refresher training of employees for the key positions at the power plant, demonstrated are the advantages and disadvantages of both approaches, and recommendations are given for the development of personnel training system.

Subsection 5.2
TRANSFER OF THE VETERAN TRADITIONS AND EXPERIENCE TO THE YOUTH

Fifty years of the training experience in the field of protection of metals from corrosion at the department of nuclear power plants NRU "MEI"

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The course on corrosion and protection of metals in NPPs called " Nuclear Engineering Materials " for students majoring occupation 0310 "Design and operation of nuclear power plants and installations" was put at MEI by Doctor of Technical Science Valentin V. Gerasimov, in 1963. The discipline consisted of lectures , exercises and laboratory works. It had a theoretical and applied nature: all three components - lectures, exercises , laboratory works - were aimed at fostering students in corrosion management skills of structural alloys by means of analytical and experimental studies confirming the multiplicity increment time to failure . The course was developed and the report shows how it looks and provides a theory that it contains.

Energy-efficient technology for transmitting natural light by "Solatube® Daylighting Systems" through the building envelope — a guarantee of comfort and security for personnel of nuclear power plants

Kazakov VA
OJSC «RITM-2»

Emphasis is placed on safety, light comfort and energy efficiency with guaranteed transmission of natural light into the interior spaces of buildings. An innovative technology of natural light transfer via Solatube® Daylighting Systems is presented, as an effective way to create light comfort in buildings. This system allows decomposing the tasks imposed on traditional translucent structures, widely used in construction of buildings and facilities.

Natural sunlight is essential for the physical and psychological health. Sun is the main source of life on the Earth, its light inspires optimism and gives new strength. Only natural sunlight provides a due visual comfort, launches a complicated mechanism of biorhythms, rescues from depression.

Exactly because, the presence of natural light in the residence of a person is very important. Lack of this essential factor in the residence of a person is regarded as a harm to the health.

Adopted at the end of last year, the Federal Law № 426 dated 28.12.13 "On special assessment of working conditions" confirms this fact. Here, the lack of natural light in the workplace is qualified as hazardous working conditions.

The problem of natural light transmission into internal spaces of buildings and findings are examined.

Understanding the existing problems of natural light transmission through the building envelope and the inconsistencies, it is necessary to have a fresh look at ways to solve this problem. Technological breakthrough is needed in this direction.

Currently , Russia has developed and prepared for production some first samples of a dynamic lighting module. This module is an option of Solatube® systems, which has allowed the fabrication of a luminaire of a new generation – a "hybrid" lighting system. This system is a source of full light (natural, artificial or mixed), which automatically adjusts light output and is regulated as a function of environmental factors, time and user preferences. This solution allows the use of a hybrid lighting system to illuminate the highly explosive objects and facilities, where operators perform their work requiring high attention associated with high psycho-emotional stress.

Hybrid lighting system has begun to find application in new projects.

Widespread introduction of Solatube® systems in construction would successfully meet the challenges for creating a sustainable environment where people will live and perform their high quality job without harm to their health and without life hazards.

Continuity of generations – economic growth of the state-owned corporation Rosatom

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MSOO MSVAEP

The worth of information is not in the fact that you have heard and recognized it, but in the fact that only experience allows you to apply this information effectively in practice.

Prerequisites that motivate society to integrate veterans in the "Nuclear Energy" business process branch are discussed, in particular, external, intra-economic and social background, etc.

Currently, a positive case to incorporate veterans of "Nuclear Energy" branch in the business processes of industry has been already demonstrated: MSOO MSVAEP (International Union of Public Organizations and the International Union of Atomic Energy and Industry) was put in place. As well, OJSC "Nulian Professional" allowing to conserve and scale the Russian nuclear school through effective methods of knowledge and experience transfer owing to veterans participation in business processes of the industry was created.

Existing Russian legislation and regulations of Rosatom on “ purchases”, including Order placement at a single service provider being studied, series of actions allowing to really embed industry veterans back into the business process as a service provider of engineering nature are proposed:

- to note “Nulian Professional” company in the Order dated 21.02.2014 № 1/157-P as a single supplier (performer, contractor) regarding the nature of engineering services, including “backend” scope according to the Russian classification system of services (goods, works);
- to note “Nulian Professional” company in the Order dated 21.02.2014 № 1/157-P as a single supplier (performer, contractor) regarding the nature of engineering services in accordance with the provisions of Part 16, Article 34 of the Federal Law № 44-FZ “Contract life cycle”;
- to create divisions in areas of engineering services composed of industry veterans under the auspices of “Nulian Professional”;

To ensure the functionality of divisions, it is recommended to create a medical rehabilitation center. Source of funding - profits derived from activities of “Nulian Professional”.

Lifetime extension of the process pipelines in nuclear power plants

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Lifetime extension of components and assemblies is widely used nowadays in various sectors of the economy, as it allows significant budget and material resources saving. In nuclear power, lifetime extension has its own features related to safety associated with further operation of these facilities. The main problem is to extend the lifetime of process piping (PP), as its repair is associated with the excavation works on the landscaped sites and impede the normal operation of a nuclear power plant. This problem has two items, the solution of which varies in their methods of implementation.

1. Diagnostics of the pipeline corrosive condition.
2. Installing cathodic protection system by pulse currents.

The most complete and direct information about the development of (hazard) of defects in the metal components and elements of process equipment (TO) and PP can be obtained by acoustic emission (AE) as AE control cracking in its resolution far exceeds the capabilities of traditional methods of non-destructive testing (NDT). On the other hand, even the presence of macroscopic defects, “embedded» during manufacturing or assembling equipment and pipelines and detected by conventional NDT methods, is not generally a reason to reject products after their operation over the long term (20-30 years). Therefore, AE control can be used as a determinant to establish criteria for TO and PP admittance found to be defective (providing the necessary level of safety and time of operation) .

Diagnostics of pipeline corrosive conditions are performed through electrical measurements (to evaluate electrochemical potential of pipelines) over its entire length. According to the results of electrical measurements and AE testing, sectors to be trenched for pits are identified, condition of the insulation coating, depth, area and character of corrosion damage etc. are examined. Then, conclusion on damaged pipeline is given for its further operation, repair or replacement. At this stage, it is highly desirable to implement X-ray examination of welded joints and connections in the trench.

Upon receipt of conclusion on the further operation of pipeline in the most corrosive point (and other vulnerable places) a measuring comparison electrode and a corrosion witness are installed. Then, an anodic grounding (anodic groundings) and a cathodic protection station by pulsed current, in a predetermined, suitable and good for further operation place, are mounted. Mounting a cathodic protection station by pulsed current is an indispensable reliable pipeline protection against soil corrosion. If installing DC cathodic protection station on a site with a multi-line pipelines, there are conditions to screen a pipe by an another one (which leads to loss the protection), or allow the current to flow from one pipe to another (that leads to accelerate corrosion process and result in pitting at the outlet of current from the tube) .

New commissioned underground metallic structures are recommended to be protected only by pulse techniques and pulsed cathodic protection system are to be included in the project documentation at the design stage.

Feasibility and necessity of participation of veterans of nuclear power in Russia in international cooperation

Poroikov V.S.
MSOO MSVAEP

The results of the International Union of nuclear energy and industry veterans are discussed. Issues of eventual efforts of veterans in social and industrial activities, their contribution to support the positive image of nuclear energy and industry are considered . It is emphasized the importance of their contribution in the struggle for international markets to build new nuclear power plants outside of Russia, and especially in a time of tension in international relations. Potential actions of veterans to promote Russian advances in the design and construction of nuclear power plants are considered.

It is proposed to develop industrial (company level) concept of veterans participation in international cooperation under regular supervision from Management (JSC Concern “Rosenergoatom”) side.

Forming pre-project proposals for the development of specifications related to the implementation of project "System of electrochemical corrosion protection of pipelines by pulsed currents in technological systems of Kalinin NPP"

(method: cathodic protection of underground pipelines by pulsed currents)

Yu.K. Terekhin
Secretariat MSVAEP

1. DESCRIPTION AND PURPOSE

To protect underground pipelines from corrosion impact, various coatings protecting surface of pipelines are used. Over time, under the influence of temperature changes, mechanical stress or ground movement some micro cracks in protective coatings appear, and stray currents penetrate in the pure metal. They cause a particularly dangerous form of corrosion – a galvanic corrosion . The nuclear site is an energy saturated site, here the stray currents are induced and reach significant values, thus, protection of underground pipelines and steel structures from galvanic corrosion has become a priority .

To overcome these shortcomings, inherently present in DC electrochemical protection, a cathodic protection system by pulsed current is developed. Its functioning is different from the traditional one.

In recent years, new techniques of cathodic protection by **pulsed** currents are encouraged.

Since the development of a new cathodic protection system for underground pipelines, there are already about a dozen installations protected against corrosion by using this method. Some of them were re-inspected. Undoubted usefulness of pulsed cathodic protection everywhere was observed.

To perform tasks associated with a reliable operation of this system and a level of protection on all controlled pipelines, special equipment and materials have been developed. These means adequately respond to the task to ensure protection of process pipelines from corrosion in industrial energy saturated sites.

Electrometric measurements and investigations near emergency diesel generator station of Kalinin NPP, unit 1, to assess the potential application of pulsed cathodic protection of pipelines and to provide a preliminary assessment of electric fields around industrial pipelines, were conducted.

Experiments on the basis of new developments demonstrated not only the opportunity, but also the efficiency of cathodic protection system by pulsed currents; the implementation of this project would increase the operational lifetime of process piping without maintenance and improve operational reliability and safety of systems having underground pipes.

Currently, based on the exhaustive implementation program "Electrochemical Protection Systems Using Pulsed Purrents Against Corrosion," "Nuclear Professionals", together with the developers of JSC "Katod" and OJSC "Alitir" and involved engineers of Kalinin NPP started to install this system at its industrial site as the most advanced and promising facility to protect underground pipelines of technological systems of Kalinin NPP.

Treatment of Waste – the Key of Nuclear Progress

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The priority problem of the nuclear operation today is the generation of nuclear waste. The safety culture with management control and review techniques, technical solutions and control systems is subject to permanent progress and development. The problem is nuclear waste. Nuclear waste today is of different dimension than it was decades ago. The challenge the nuclear industry is facing today is the taking care of "invalid" elements with destroyed elementary structure, result of nuclear based energy generation.

All our conventional human activities and technologies in fact are about the use of the *Weak Interrelation* of the elementary world, utilising the "benefit" of burning, melting and froze; mixing, steaming and modifying the elementary structure, producing waste, toxic, non-toxic and any kind, but without impacting the *Strong Interrelations* and with that the proton-neutron process balance.

Nuclear based energy generation is about the impact of the *Strong Interrelation*. It destroys the elementary balance of the fuel element of the operation. The destroyed balance of the fuel results in the chain of damaged elements with destroyed balance. Elements impacted become nuclear waste – with invalid elementary structure.

Nuclear operation results in damaged elementary processes. The more we operate the more is the damage.

Does the nuclear industry need solution on this? Yes it certainly needs.

There is however no conventional way for finding the solution. The solution is the *process based* quantum approach with the practical examples of the rehabilitation for experiments.

The need to introduce at NPPs "Omniscan" flaw detector based on phased arrays for metal and pipeline testing

Kozin Y.N.
MSOO MSVAEP

Application of "Omniscan" flaw detector in the process of metal and welded joints of NPP equipment and pipelines control, based on phased arrays, will reduce the radiation burden on operators by reducing the residence time in the radiation field produced by the equipment and pipelines, thus, increasing the quality of control. For wide application of "Omniscan" flaw detector, it is necessary to develop control techniques , tried and duly approved. Implementation of such works can be carried out with the participation of specialists : veterans of "Rosenergoatom" company.

The contents of the publication:

1. Physical basis of excitation, in the controlled facility, of ultrasonic waves using phased arrays.
2. Detection principle of discontinuities in the tested facility using "Omniskan" flaw detector based on phased arrays.
3. Records of control and evaluation of discontinuities based on the results of control.
4. Contents of the project on control techniques using phased array

Conservation and transfer of knowledge on NPP units configuration

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Length of a NPP power unit life cycle is about 50-70 years or more. The longest period of this life cycle is the decommissioning. At the beginning of this stage and in connection with significant number of operational staff departure, a real loss of key knowledge concerning the configuration of power unit and its operating history can happen, if the operating organization will not put in advance a set of measures related to the conservation of knowledge.

NPP power unit decommissioning is a complex process composed of several stages having a local (on-site) concept on decommissioning, a decommissioning program , measures for conducting a comprehensive engineering and radiation survey, a decommissioning project, decontamination and dismantling, RW handling etc. At any stage of the decommissioning, sound decisions can be taken only on the base of required appropriate and comprehensive information.

The main problem for the nuclear power unit decommissioning is the problem of processing and removal of radioactive waste for its further storage or disposal that will be generated when dismantling reactor and radioactive structures of the unit. Currently, Russia has not RW repositories and burial grounds required during decommissioning of nuclear power plant unit . This factor , as well as difficulties related to financing of operations for immediate dismantling of the reactor radioactive structures requires SAFSTOR option that is the main in decommissioning of nuclear power plants in Russia.

SAFSTOR requires a sufficiently long retention time of unit radioactive structures under surveillance. For example, the defined length of LNPP Units 1,2 decommissioning period is not less than 50 years. In such circumstances, preservation and transfer of all relevant information for experts coming generations, who will carry out installation and equipment dismantling, becomes the most important mission , as the availability of reliable information directly affects the safety and efficiency of decommissioning works. All the more, under some reasons, Russia operating nuclear power plants currently storage design and operational documentation, as a rule, only on a hard copy. During operations one part of information is irretrievably lost; the hard copy comes in poor condition. Often, important information about the history of nuclear power unit operations is kept only in specialists memory and it is lost when they leave the station.

Clearly, fundamental changes in such situation to support information on decommissioning works are only possible when creating special purposefully updated and maintained decommissioning database of NPP units design and operational documentation that ensures its long-term and reliable storage. Early deployment and filling such decommissioning database allow, in centralized manner, preserve and transfer all necessary documentation and required data for the practical implementation of decommissioning works to future generations of experts.

Report provides a practical and methodical experience gained by the authors in creating databases for NPPs decommissioning. The paper explains possibilities for NPP units to use three-dimensional engineering models to solve problems related to preparation and implementation of decommissioning, as well for operation of existing nuclear power plants in their lifetime extension.

Students vocational guidance visits: lyceum 1547, IAE Kurchatov, first nuclear power plant in the world, IATE MIFI - in October 2013 and practical conclusions

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MOOVK

In the framework of career-oriented activities, the Inter-Regional Organization of Rosenergoatom Veterans (MOOVK) organized a tour for students of 9 - 11 grades: Petrozavodsk (school number 34), Olonets (Karelia, school number 2) at the Scientific Research Center "Kurchatov Institute", the Memorial " The world's first nuclear power plant" of Nuclear Energy Institute IATE MIFI and at Info Center of Rosatom Lyceum, initiated by MOOVK, 15 – 17 October 2013. Organization and tour management were performed by V.P. Kalinin , who is veteran of MOOVK having an experience of tours organization for schoolchildren at NPP and the nuclear icebreaker "Lenin" in 2011-2012. In early July 2013 , together with teachers of physics Lesnitskaya A.V., Bogdanova N.N., a time schedule of tours was prepared and agreed, a list of students was established, in total 21 people, 4 accompanying teachers from two schools

Later, in schools, Executive Order were issued on the responsibility of accompanying persons concerning safety, health and life of students during the travel and excursion, Attachment 1.

In the railway ticket office of Petrozavodsk, in October 2013, information was obtained on the cost of tickets for travel to Moscow and from the station Lodeinoe Pole to Moscow. The timetable of excursion with information on the cost for travel and the covering letter were submitted to the PetrozavodskMash Rosatom plant Management. Its response on the career-oriented tour sponsorship, according to the presented documents, was positive, Attachment 2.

At the beginning of September 2013 a program on tour activities agreed by hosts was developed and approved, railway tickets were acquired, a hotel for children was booked. Cherkasov V.G. and Sofienko G.I, both from MOOVK, achieved allocation of a bus for 3 days tour. After preparatory work with school teachers and parents of students, Lesnitskaya A.V., Lobskaya V.V. (school 34) and Bogdanova N.N., Shishkina G.N. (school 2) held a discussion about the safety and behavior of students during the travel and excursion.

15.10.2013 V.P. Kalinin met students at the railway station, took a car-bus for Ostankino TV-tower visiting, then the Lyceum 1547, accordingly to the Programme: movies "Meet the conventional and nuclear energy", "Radioactive Waste Management", an interactive game, an excursion through Lyceum.

Tour through Lyceum left a lasting impression on students and the disciples spoke about this in their writings. 10.16.2013 a tour was made in " Kurchatov Institute" : an exhibition on the institute and the Soviet / Russian science and technology achievements, the first Research Reactor , Kurchatov's house. Excursion to the "world's first nuclear power plant". Students consciously understood what is the nuclear energy , atomic and hydrogen bombs , why scientists and industry employees, under the leadership of Kurchatov I.V., promptly completed the Government's directive during the war and difficult postwar years ... Attachment 3 .

As a result of the tour at important nuclear industry facilities and during the meetings with industry veterans, many students had arisen a thought to get a profession in ad-hoc institutions and become, after training, an NPP employee in order to generate safe and cheap electricity for conducting interesting and wanted researches. Schools should resume such excursions in which students, veterans of the industry and public are involved, install exhibition stands and libraries. Such career-oriented activities should be carried out continuously in order to obtain the goal – arrival of the younger generation to replace highly qualified outgoing specialists.

V.P. Kalinin, MOOVK responsible for organization and conduct of career-oriented tours.

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