MNTK

Tenth International
Scientific and Technical
Conference

SAFETY, EFFICIENCY AND ECONOMICS OF NUCLEAR POWER INDUSTRY

BOOK OF ABSTRACTS

May 25–27

2016, Moscow

ROSENERGOATOM

70 Years of Nuclear Industry in Russia

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NUCLEAR POWER COMPLEX OF RUSSIA: SAFETY AND EFFICIENCY

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To ***the requirements*** to large-scale nuclear power complex (NPC) of Russia are both traditional:

- **guaranteed safety**
- **economic efficiency**

and systematic, ensuring its long-term staying in demand:

- ***By the size of power generation*** the share of nuclear power in electric power market in the country should not be less than 30%.
- ***By export potential*** the volume of nuclear power technologies export (when meeting the requirement of sensitive materials and technologies non-dissemination) should be comparable to the volume of its domestic use.
- ***By the structure of power generation*** NPC should provide for the possibility to expand sales markets, i.e. to ensure thermal power generation for both electric power generation and "non-electric" applications: heating, water desalination, hydrogen and motor fuel manufacture, new technological appliances.
- ***By raw material base*** NPC shouldn't be limited for the historically significant period of time (hundreds of years).
- ***By RAW management*** arrangement of the NPC fuel cycle should provide for safe and final isolation of RAW.

Nuclear power development principles implementation makes for the exact objectives that can be put as follows:

- in short-term prospect:
  - power capacity build-up to achieve NP share in overall electric power generation in the country by at least 25% by 2030 to create the investment basis for development,
  - investigation of demands and ways of development of regional nuclear power industry on the basis of NPP with small and medium capacity reactors,
  - ensuring growth in nuclear technologies export at the level comparable to the one of the same technologies use inside the country,

  - setup of basic elements of new technological platform of large-scale NPC for nuclear fuel cycle closing.
- in long-term prospect:
  - setup and deployment of closed by Uranium and Plutonium large-scale NPC as a basis for power supply for Russia sustainable development in the third millennium.

Thanks to the achieve by today results, Russia now has a high level of technological preparedness for ensuring innovative development of her nuclear power industry. The level of this preparedness is determined by the following parameters.

By reactor technologies:

- There was developed an evolutionary design AES-2006 with VVER 1200 reactor for mass construction of NPPs in short-term prospect.
- At pilot operation level there were successfully demonstrated sodium cooled fast neutron reactors – BN-800 power unit is at the stage of commissioning, there was started the programme of its 100% loading with MOX-fuel and its use for mastering pilot technologies of closed nuclear fuel cycle.
- At various stages of completion there are developmental works for new designs of sodium cooled fast neutron reactors of high capacity, fast reactors with heavy metals, and some projects for small and medium size reactors.

By closed nuclear fuel cycle technologies (NFC):

- At industrial level there was demonstrated a technology of water chemical treatment of spent nuclear fuel (SNF) of Uranium reactors, with Plutonium discharge and glassing of highly radioactive waste (RAW) (plant RT-1).
- At pilot operation level there were demonstrated pellet and vibro- technologies for mixed Uranium-Plutonium oxide fuel (MOX-fuel) of sodium cooled fast neutron reactors.
- R&D works were started for development of alternative technologies of NP fuel cycle with fast reactors (nitride fuel, SNF reprocessing dry methods, transmutation of minor actinides (MA) in fast reactors; elements of Uranium – Thorium cycle).
- There are being considered the concepts of hybrid accelerating-management facilities and molten salt reactors for the purpose of long-life RAW burning-out.

By technologies of nuclear power sources for “non-electric” applications:

- At pilot operation level there was demonstrated the possibility to use sodium cooled fast reactors (BN) for water desalination (BN-350), and also thermal reactors for regional heat supply (Bilibino NPP).
- At various stages of development there are technologies of power generation for “non-electric” and mixed application, and projects on promising nuclear power facilities for these technologies implementation, includ-

ing facilities of transport and space energy complex, heat supply, water desalination, coal gasification, and hydrogen production.

Basing on this status Russian nuclear industry shall now start its systemic development to achieve the set short-term and long-term goals – creation of double-component nuclear power complex with thermal and fast reactors operating in closed fuel cycle.

NEW APPROACHES TO DESIGNING NUCLEAR POWER UNITS

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The important integral part of successful implementation of NPP construction project is effective project information management and data transfer in the required scope and format in-between the stages of NPP lifecycle. Information effective management under the project is achieved through organization of work of all participants to the project in common informational space ensuring data exchange between them, their interaction between themselves and with the project Customer, project management, processes of data management relating to equipment, procurement, supplies, Project documentation management, construction-and-erection and commissioning works.

Works in common informational space (CIS) are based on NPP design informational model developed at the stages of design and engineering documentation issue. All data under the project at power unit engineering and construction stages (including NPP 3D-models, documents, data bases) are gathered in unified system of project information management supporting the processes of project configuration management (including requirements management and modifications management), and then, in structured view, are transferred to the customer to operation systems – to be used at subsequent stages of the power unit lifecycle.

THE «BREAKTHROUGH» PROJECT

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1. The reasons for the global decline in the Nuclear Power Industry (NPI):

- impact of severe accidents;
- deferred decision on the SNF issue;
- competitiveness of the NPI;
- issues of nuclear weapons non-proliferation
- availability of raw material resources.

2. Solving the issues of the NPI within the framework of Federal Target Programme "New Technology Platform – the Breakthrough Project" for closed fuel cycle at fast neutrons reactors:

- No accidents that require the population evacuation.
- the radiation equivalency approach to RW disposal.
- restoring competitiveness of NPPs.
- technological strengthening of the non-proliferation regime.
- use of the raw materials potential to the full extent.

3. The Breakthrough Project completion status:

- Establishment of facilities for production of mixed nitride U-Pu fuel (MNUP fuel).
- Research on the MNUP fuel and reactor testing of the fuel.
- Implementation of the BREST-OD-300 power unit projects, manufacturing and processing of fuel for the pilot demonstration facility (PDF).

4. Construction of the PDF at the Rosatom's Siberian Chemical Combine.

FIRST OF A KIND UNIT OF NEW GENERATION VVER-1200. SPECIFICS OF PUTTING INTO OPERATION

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NVNPP-2 is a nuclear power station of the second generation. It is constructed under design "AES-2006" developed by JSC Atomenergoproekt considering application of VVER-1200 reactor unit. This is an evolution of design "AES-92" with VVER-1000. Design "AES-92" meets all Russian and international safety requirements.

One of the main differences of NVNPP-2 design from both Russian series NPP desing and foreign NPPs, is two-trains system of active safety systems; there are also additional passive safety systems considered. Into RU V-392M design there was introduced a range of constructive improvements if compared to RU V-320 (design "AES U-87"). Significant specific features are seen in reactor pressure vessel of NVNPP-2 design.

In order to improve steam generators reliability during their operation there was introduced a new modernized construction of piping, SG diameter was enlarged, regularity and blowing-through scope was changed, new chemistry for the secondary circuit was introduced.

To improve NVNPP-2 sustainability to hardly probable hypothetic events similar to the ones at Fukishima NPP, and also to increase period of NPP autonomous operation in case of beyond-design basis accidents, in NVNPP-2 project there was introduced a number of additional technical measures.

Application of improved localizing safety systems including among other components double shell of containment with ventilated annular, core catcher, guarantees exclusion of radioactive releases into the environment in case of accident.

The structure of automated process control system (APCS) has a hierarchical principle in its basis in accordance with power unit zoning as a controlled object to technological functional areas and groups, where I&C means ensure fulfillment of tasks of communication with controlled technological object (data collection and signals output), and also implementation of tasks of protection, interlocks, automatic control, etc.

There was completed optimization of radiation protection design using limitation of considered in the project collective dose rate for the personnel when servicing the reactor unit. The programme of ecological monitoring of surface and water natural systems of Novovoronezh NPP-2 area is a programme of monitoring and measurement of the area ecosystem parameters implemented within the frames of post-project ecological support. NVNPP-2 water chemical treatment is a good example of integrated application of new methods and equipment with main technological process being fully automated.

When implementing AES-2006 design at NVNPP-2 the following main tasks complying with the modern level of nuclear industry development are being solved:

- achievement of the required by modern standards safety parameters of NPP;
- consideration of international trends of NPP safety improvement
- maximum use of proved by experience technologies and equipment.

BN-800 REACTOR AS THE PIONEER REACTOR OF NEW GENERATION. SPECIAL ASPECTS OF COMMISSIONING

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The development of the fast neutrons nuclear power industry started with the research reactor BR-5/10 (1959), and, later on, BOR-60 (1969).

The evolution of power reactors started with the commissioning of BN-350 in 1973. In 1980, BN-600 reactor was commissioned, and it has been in operation ever since. BN-800 reactor start-up was conducted in 2015.

The BN-800 design has been developed using the inherent safety principles and the secondary shutdown system that operates passively.

Aside from its designated purpose, the main tasks of BN-800 are optimization of NFC closure on a pilot demonstration scale, and minor actinide burnout in order to reduce the RW activity in the long run.

The turbine generator was connected to the network for the first time in December 2015. BN-800 power start-up was completed in February 2016. Currently, the BN-800 commissioning works are underway. The reactor is at the stage of the pilot operation.

Main peculiarities of commissioning a power unit with BN-800 reactor are associated with its design features:

- sodium used as the reactor coolant, which necessitates construction of a sodium storage facility;
- three circuit heat rejection (the first and the second circuits contain sodium, while the third one - water/steam);
- additional system of heat removal from the reactor core through the second circuit by means of an aerial cooler.;
- compact reactor core.

IAEA ACTIVITIES IN THE NUCLEAR INFRASTRUCTURE DEVELOPMENT DOMAIN

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Nearly every aspect of development — from reducing poverty and raising living standards to improving health care and industrial and agricultural productivity — requires access to modern energy sources. Current forecasts suggest that global electricity use will increase 65–100% by 2030, with most of the growth in developing countries. Many IAEA Member States without nuclear power have expressed interest in introducing it in order to meet their energy needs without increasing reliance on fossil fuels. It is recognized that a nuclear power programme is a major undertaking requiring careful planning, preparation and investment in time, institutions and human resources. While nuclear power is not alone in this respect, it is different because of the safety, security and safeguards requirements associated with using nuclear material. To introduce nuclear power, a wide range of infrastructure issues needs to be considered. The IAEA developed the "Milestones Approach" to assist countries that are considering or planning their first nuclear power plant. The aim is to help Member States understand the commitments and obligations associated with developing a nuclear power programme. The presentation briefly considers nuclear power in the context of sustainable energy and the factors that make nuclear power unique and highlights the IAEA predictions of the trend of global nuclear power capacity up to 2030 and the drivers for nuclear power. The presentation describes the Milestones Approach and the 19 Infrastructure Issues and provides an overview of the activities undertaken by the IAEA to support newcomer countries.

WANO PROGRAMS AT THE PRESENT STAGE AND THEIR IMPACT ON SAFETY

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WANO was founded in Moscow in 1989 as a response of the world nuclear society to the Chernobyl accident in order to implement experience exchange and mutual support worldwide. WANO comprises 125 members / 231 plants / 435 units.

To support its members WANO offers them services through four main programmes: Operating Experience, Peer Reviews, Professional and Technical Development, Technical Support and Exchange.

The WANO BGM in Shenzhen in 2011 marked the beginning of the post-Fukushima stage of WANO reformation. The five recommendations of WANO post-Fukushima commission included the WANO performance expansion, creating the response strategy to events of the industry, increase of WANO credibility, strengthening of WANO authority and openness, increasing the internal consistency and implementation of 12 post-Fukushima WANO projects.

During the Toronto BGM in 2015 the new long term plan called COMPASS was introduced and presented in details to the participants. The major new elements in the WANO strategy and programs are:

- Provide support to members in WANO activities with new built reactors without having developed nuclear infrastructure.
- Monitoring member/plant performance and providing increased assistance to members with low performance (plants of focus).

CONTINUOUS IMPROVEMENT OF REACTOR SAFETY AT THE EDF FLEET. EXTENDING THE OPERATIONAL LIFE SPAN OF THE FRENCH NUCLEAR FLEET BEYOND 40 YEARS

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The initial design for pressurized water reactors assumed and expected a life span of 40 years. In-depth analyses will be required if this time frame is to be exceeded, at least in terms of equipment resistance.

Since 2010, the basic position of the French nuclear safety authority has been that the 40-year stage is a new domain for nuclear safety. This position is informed by the existence of Generation-3 reactors like the EPR.

EDF considers that there are 4 industrial conditions for extending operational life span well beyond 40 years:

- First, a high performing fleet with error-free operation both day-to-day and in the long term;
- Second, to set and meet nuclear safety and environmental protection targets that are acceptable to the general public for operational plants over the long term, against a backdrop of Generation III reactor construction;
- Third, to maintain and renew internal engineering and operating skills; and
- Fourth, to maintain and renew necessary industrial equipment.

EDF considers that enhancing the nuclear safety of the fleet in operation is a continuous process. Evolution of NPP's design has divided the risk of core melts by a factor of 2 between 20 years and 30 years of operation and EDF has implemented in the past international and national event-related OPEX : Design was modified after Three-Mile-Island (TMI), after Chernobyl, after Fukushima, and after the Blayais flooding and heat-wave events in France.

EDF will implement large human and financial investments in improving nuclear safety to go beyond 40 years, it means engineering and purchasing.

In France, nuclear safety objectives for continued operation beyond 40 years have to be reviewed. For example for design-basis accidents, we will aim for having no need to implement protective measures of the population like evacuation, containment or distribution of iodine tablets.

Another example is for core melt accidents where the aim is to reduce the risk of early release and to avoid large releases and long lasting environmental impacts.

For example as regards the aim and objective of design-basis accidents, we consider that the radiological consequences of accidents have to decrease by a factor 5 to 10.

EDF has implemented 3 sustainable processes to ensure the long-term fitness for service of equipment and facilities:

- First, initial ongoing in-service inspection and maintenance process,
- Second, management process for ageing systems, structures and components,
- And third, management process for component obsolescence.

These 3 processes incorporate all skills from upstream to downstream: R&D, corporate design and operations engineering teams on NPPs at site level. They center on an active R&D arm and large-scale EDF's involvement in international programs, and these processes are in line with best international practices, as confirmed by the 2014 Corporate OSART that EDF received.

EDF has embarked on a deliberate and challenging industrial program. For non reprocessible equipment, we demonstrate that such equipment is able to perform as required until more than 50 years and for reprocessible equipment, we justify its continued use in service or its replacement or renovation where its expected life span is less than 50 years.

In conclusion, the program undertaken by EDF with a view to continuing the operation of its reactors beyond 40 years is an ambitious and challenging, but attainable step forward for nuclear safety for the nuclear fleet in operation in France. It will make it possible to reach a very high level of nuclear safety that is close to the world-wide requirements on new reactors in terms of the radiological consequences of an accident, meaning that the extension of the operational life span is indeed a possibility.

THE STRATEGY TOWARDS ZERO FUEL FAILURES

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The global experience in achieving the industry goal of zero failures shows that it is possible to gradually achieve that by applying a systematic approach to the ensuring the cleanliness of circuits at power units and monitoring production and use of nuclear fuel.

In the period up to the 2010s, there were repeated cladding leaks and damage to the fuel assemblies, which resulted in implementation of some technical measures (changes to the FA design or the requirements for its operation). There measures showed positive results in operation of RBMK power units of NVNPP and KOLNPP, and power units 1 and 3 of KLNNPP. However, such results are not considered in this paper.

Currently, for 90% of cases, the available statistics of cladding failures during operation and the results of the FA inspection are of no assistance in determining the root reasons for the failures, except for the postirradiation examination of individual failed FAs conducted at NIIAR State Research Center JSC. Neither operation reports by the fuel supplier TVEL JSC, no certificates of NPP-specific commissions state the failure reason conclusively. At the same time, in some cases, direct observations of leaky FAs show the presence of foreign materials at the bundle of fuel elements, which can be the reason for the debris damage due to mechanical interaction of foreign materials with the cladding in the flow of the coolant, and subsequent secondary processes.

The statistics of fuel assembly leakage for the period from 2010 to 2015 shows that:

- fuel failures occur at the early stages of operation;
- fuel failures occur at power units that have shown no signs of fuel damage before;
- fuel failures occur at new foreign power units of different operating companies.

The failure instances listed show that there may be other factors that can lead to fuel failures, in particular, failures related to fuel manufacturing.

These facts and their analysis have been the background for the Action Plans to Achieve Zero Failures at NPPs developed by the Fuel Supplier TVEL JSC and the Operating Company Rosenergoatom Concern. These plans include the following actions:

- managerial and technical measures to ensure the cleanliness of the NPP circuits;
- auditing NPPs in order to analyze the efficiency of the measures;
- auditing fuel manufacturers in order to analyze the manufacturing and to identify the deviations from the requirements of quality and technical documentation, and to systematically control manufacturing by the Customer;
- introducing highly efficient anti-debris filters at FAs;
- improving the FA manufacturing techniques in order to improve quality and technical oversight;
- improving the inspection method and additional postirradiation examinstions in order to identify the causes of fuel failurs.

DESIGN BN-1200 AS A BASIS OF TRANSFER TO BICOMPONENT NUCLEAR POWER

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In the basis of State Corporation ROSATOM strategic goal achievement for nuclear fuel cycle closing lays creation of bicomponent nuclear energy system founded on VVER and BN type reactors. Namely BN reactor capabilities for fuel reproduction, for Plutonium and Uranium effective utilization produced in process of VVER and BN SNF reprocessing, at nuclear fuel cycle closing allows to realize the balanced nuclear energy system of the NPP even in the short-term prospect.

Experience and competencies get during BN-600 reactor operation and BN-800 reactor development, ensured elaboration of materials for design of BN-1200 commercial power unit where there is introduced a range of new technical solutions provided for improvement of safety and economic efficiency parameters, that is an obligatory condition for transferring to series construction of BN reactors.

Works under BN-1200 RU design are in progress starting with 2007 in accordance with Rosenergoatom programme. From 2010 onwards R&D works under the project of sodium cooled fast reactor BN-1200 creation were continued within the frames of implementation of Federal target programme "Nuclear power technologies of new generation for 2010-2015 and up to 2020".

Commercial operation of fast reactors basing on BN-1200 design considers setup of a number of facilities using closed fuel cycle combining the existing and newly constructed technologically linked clasters of SNF temporary storage, BN and VVER SNF reprocessing, production of Plutonium raw fuel, manufacture of mixed Uranium-Plutonium fuel, preparation of all types of RAW to isolation. When developing BN-1200 design there is considered the possibility to use to types of fuel: nitride fuel as a prospective one, after getting the necessary data proving its economic efficiency, and MOX-fuel demonstrating the possibility of deep burning-up ratio achievement.

For MOX-fuel for BN reactors we gained some experience: when using technology of pellet-type MOX-fuel manufacture implemented then at the integrated mining and chemical plant production site, there were manufactured and successfully tested in BN-600 reactor 42 experimental FA. Part of these FAs were reprocessed at plant RT-1 of MAYAK company in Ozersk town of Cheliabinsk region where there are also Uranium FAs from BN-600 reactor reprocessing is carried out.

As concerns BN-1200 power unit design, the main task is to complete the works for technical solutions optimization that are aimed at ensuring comparable to VVER-TOI capital investment costs. Safety requirements set at the level of the 4^{th} generation nuclear power units are already substantiated in available design materials.

As for closed nuclear fuel cycle enterprises, it is wise to consider, inclusively, the possibility to use already established and being presently setup nuclear fuel cycle facilities.

Accounting for availability in Russia operating facilities for SNF reprocessing after thermal reactors, putting into operation the first BN-1200 power units will ensure effective solution to the task of already accumulated Plutonium utilization.

Section 1

SAFE AND EFFECTIVE OPERATION OF RUSSIAN 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

In Vessel Melt Retention for VVER 1000- Status of Work

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After the Fukushima accident one of the principal safety requirements is final mitigation solution for the severe accident. Already lot of work was done and realized to install many mitigation measures. Even though that probability to reach SA situation with core melt down is extremely low, to find solution for In Vessel Melt Retention (IVMR) for VVER 1000 Units is of crucial importance. For VVER 440 this solution is already applied for most of units in operation, based on thorough research started for Loviisa NPP. The VVER 1000 units in operation have larger power and application of the IVMR needs to be justified with very thorough analytical and experimental work. At present the EC project: IVMR for Existing and Future NPPs started with very strong participation of many European Companies and Institutes. Within this project, TASK 4 is devoted to large scale experiment with external RPV cooling with exact configuration of VVER 1000. Part of this large scale experimental work is also continuation of small scale experiments which provides very significant guidance and results for the follow on large scale experiments. In proposed presentations key results from the small scale experimental program and status of work on large scale experimental facility will be presented. Results already available provide already very significant and positive results for justification of the IVMR strategy for VVER 1000 type reactors.

Updating cladding integrity monitoring methods at an operating reactor to determine fuel burn-out of a leaky fuel assembly

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In order to increase the technical and economic fuel performance NPPs nowadays switch to fuel of increased enrichment (up to 5% of ^{235}U content) and longer fuel cycles (up to 18 months) with the load factor of a power unit up to 0.92 at the capacity of 104% N_{nom}. For NPP-2006 project, the requirements for the design strategy for fuel usage are based on the use of four-/five-year fuel cycles with refuelling once in 12 (18) months with maximum fuel burn-out in fuel assemblies up to 70 MW·day/kg U. Compensation for excess reactivity of fresh fuel brings about the harder neutrons spectrum, changed proportion of concentrations of fissionable isotopes and radioactive fission products generated from burning out. It should be taken into account for the reference radio nuclides used for identifying the presence of leaky fuel elements in the reactor core, assessing their quantity and degree of failure: radionuclides of iodine ($^{131}I \div ^{135}I$), cesium (^{134}Cs, ^{136}Cs, ^{137}Cs, ^{138}Cs) and inert radioactive gases (^{133}Xe, ^{135}Xe, ^{85m}Kr, ^{87}Kr and ^{88}Kr).

In case of the loss of fuel integrity it is important to correctly assess the burn-out of a leaky fuel assembly. It will allow reducing time expenditures for its identification during scheduled repairs at NPPs. The most sensitive indicator of fuel burn-out at VVER reactors is the $^{134}Cs/^{137}Cs$ activity ratio. The burn-out of leaky fuel is assessed by measuring the activity of cesium radionuclides at the spike effect. In view of this fact, it appears relevant to develop the theoretical methods of calculation to assess concentrations of cesium isotopes in fuel during campaign at NPPs.

This report presents a currently developed engineering model of cesium isotopes production depending on UO_2fuel burn-out and assessment of their activity ratio. The model takes into account the evolution of linear heat generation rate of a gadolinium fuel rod and surrounding fuel elements in a fuel assembly with fuel of increased enrichment, harder neutron spectrum, changed cross-sections of neutron reactions in thermal and epithermal regions. The parametric dependences have been selected in the model on the basis of data on fuel use at NPPs and detailed preliminary neutron and physical calculations for the substantiation and selection of the core composition and campaign scenario. The report provides the results of calculations by means of the well-developed model for $^{134}Cs/^{137}Cs$ activity ratio taking into account the evolution of the neutrons spectrum hardness parameter within the first campaign for fuel with the enrichment from ^{235}U 3.6 to 4.95%.

The calculation results for fuel assemblies with increased fuel enrichment and harder neutrons spectrum (up-to-date fuel cycles at VVER-1000) have demonstrated significantly higher values for the $^{134}Cs/^{137}Cs$ activity ratio depending on the burn-out as compared with the ratio for fuel assemblies with fuel enrichment of up to 3.6% during the first fuel campaign. The calculation results have shown compliance with the measurements made at NPPs with VVER-1000.

CHEMISTRY OF THE SECONDARY CIRCUIT IN DESIGNS OF NPP WITH NEW-GENERATION VVER

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The demand for increasing the reliability, safety and economic efficiency of the new-generation NPPs as well as increasing service life of the NPP main equipment up to 60 years calls upon new technologies, in particular, new chemistry.

The exploration of the reasons for corrosion of some elements of the secondary circuit and, first of all, of the steam generator, has shown that the corrosion damage cannot be explained by any single factor, but appear as a result of a combined impact of the design specific features, properties of the structural materials and quality of the working medium. The chemistry of the secondary circuit shall ensure compliance with the following functional requirements:

- minimal amount of deposits at the heat-exchanging surface of the steam generator, in the flow-path portion of the turbine and in the condensate-feed pipeline;
- preventing corrosion and corrosion-erosion damage to structural materials of steam generators, equipment and pipelines of the secondary circuit.

Regulatory framework governing the NPP chemistry has been established in the Russian Federation in the course of development of engineering design documentation of NPPs with new-generation VVERs based on the experience of operation of national NPPs, special calculations, analysis of chemistry standards in some foreign countries.

The report gives the results concerning the selection of the chemistry of the secondary circuit for NPP design with new-generation VVERs based on variants calculations and experience of operation of the existing NPPs.

The report gives consideration to the requirements to the cleanup systems for turbine condensate and steam generators' blow-down water.

Strategy for the management of severe beyond design-basis accident with fuel melting for NPP with VVER

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A molten core catcher (MCC) is one of the elements in up-to-date NPP designs with high-power reactors provided to control beyond design-basis accidents (BDBA) at the fourth level of defense in depth. It is intended to decrease severe off-design accident radiation consequences, where the destruction of the core takes place with subsequent melting of the reactor pressure vessel in case of long-term reactor core drying.

An alternative concept of containing the melt of fuel-containing material inside the reactor pressure vessel in case of an accident with reactor core melting due to external cooling of the reactor pressure vessel is aimed at increasing the NPP safety. The implementation of the concept of the in-core melt containment has a number of significant advantages as compared with the option of using an ex-core core catcher, first of all, from the point of view of reducing the cost of the power unit.

The report presents the design analyses made at the National Research Center Kurchatov Institute for implementation of the concepts of the ex-core and in-core melt containment in VVER reactors of different capacity: VVER-440, VVER-600, VVER-1000 and VVER-1200. Domestic codes SOKRAT and GEFEST-ULR, as well as a West-European code ASTEC have been used for the calculations.

The current results make it possible to come to the following conclusions:

- Use of strategy of the in-core melt containment is a possibility for VVER-440 and VVER-600 reactors due to external cooling of the reactor pressure vessel.
- Implementation of the in-core melt containment without using measures for intensification of the reactor external cooling does not seem feasible for VVER-1000.
- Core catcher shall be expressly used for VVER-1200 (projects NPP-2006 and VVER- product specification) to control heavy ODA with fuel melting.

POSSIBILITIES OF USING FIBER-OPTIC CORRELATION SYSTEM OF MEASURING REACTOR COOLANT FLOW TO INCREASE ACCURACY OF DETERMINING THERMAL POWER OF VVER-TYPE REACTORS

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One of the tasks of the program for increasing NPP energy and economic efficiency up to 2020 is the increase of the capacity of VVER-1000 above the nominal value. Rosenergoatom Concern has finished an ambitious program to increase the nominal power of up-to-date power units with VVER-type reactors up to 104%. A program for increasing the thermal power of power units with VVER-1000 up to 107-110% of the nominal value has been developed and approved.

The report defines the basic technical solutions of the fiber-optic correlation system of a new generation for measuring the flow of the primary coolant of NPPs with VVER-type reactors based on the principle of detecting high-energy gammas being generated as a result of nitrogen-16 decay. The system is intended to significantly reduce uncertainties of determining the reactor thermal power. The conceptual basis of the proposed system is the use of a correlation method with registration of high-energy gammas by means of annular fiber-optic sensors.

The proposed fiber-optic correlation system helps to significantly reduce the uncertainties of measuring the reactor coolant flow in the primary circuit up to a level of 1.5%, which is essentially lower than the actual uncertainty (~ 6%) of the measurement by means of the existing methods.

The proposed system is fully compliant with the requirements of NPP immunity to external impacts (seismic resistance, vibration resistance, tolerance to EMC and to ionizing radiation) during normal operation, and the maximum design and beyond design-basis accident.

A system of testing and verification testing equipment has been developed on the basis of a high-current accelerator for investigating serviceability of the fiber-optic correlation system of measuring the reactor primary coolant flow in NPPs with VVER-type reactors. The acceleration method helps in providing metrological support of different devices based on the records of nitrogen-16 radiation.

The incorporation of the fiber-optic correlation flow meter at the facilities of Rosenergoatom Concern OJSC and foreign NPPs with VVER-type reactors will provide considerable economic benefits, which will be, first of all, aimed at increasing safety and reliability of NPPs.

The annual cost advantage from the additionally supplied electricity at one power unit of an NPP with VVER-1000 in case of increasing its capacity by 1% N_{nom} will amount to **~75 mln. RUB**. The program of increasing the thermal power of VVER-type reactors up to 107-110% from the nominal level can be implemented at 10 Russian NPP units and 17 foreign NPP units (China, India, Iran, Belarus).

By implementing the program of increasing the thermal power of Russian NPPs with VVER-type reactors from 104% to 107% Rosenergoatom Concern OJSC will obtain extra revenue in the amount of **~ 2.3 bln. RUB**, while the additional annual extra revenue for the foreign units may amount to **~ 9 bln. RUB** .

INTEGRATED VVER DIAGNOSTICS

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The measures taken by Rosenergoatom Concern to increase efficiency of NPP functioning are as follows:

- extension of unit operational campaign;
- increasing unit capacity;
- extension of the unit useful life;
- introduction of new fuel assemblies (FA)

They shall be compensated for by the increased observability of the reactor . The reserves for extending observability comprise enhancing the on-line algorithmics of local diagnostics systems (LDS) and augmentation of the composite function of integrated diagnostics system (IDS). The IDS integrates information from the functionally independent LDSs and regular signals of automatic process control system (APCS) for acquiring additional diagnostic information by different sources.

The IDS functioning at one unit, without changing the architecture, gets expanded and configured with respect to the new input data obtained from each unit and gets installed as NPP IDS. Itgets further expanded and installed following the same logic as the IDS of the branch diagnostics center. The common computation parts of the IDS of all levels are similar and have a common interface adjusted depending on specific dataflows.

The report presents the following IDS functions:

- automatic integrated on-line diagnostics with due account of new diagnostics features;
- automatic diagnostics for analyzing the archived data accumulated by different LDSs (off-line);
- data processing by means of program-methodical service (supporting the NPP diagnostic personnel when interpreting the new diagnostic events obtained from LDSs);

The transition of Russian NPPs with VVER-1000-type reactors to an 18-months fuel cycle and the capacity of 104% has not changed the scope of reactor core (RC) monitoring. At the same time, detailed dynamic information can be extracted on-line using neutron and noise methods based on the existing VVER-1000 reactor core detectors. The report summarizes the results of the comprehensive analysis of fluctuations of in-reactor direct-charge detector signals for different units of NPPs with VVER-type reactors. Such signals contain the information both on the vibration events of RC elements and on its thermohydraulic properties (FA power density, 3D field of the reactor coolant flow rates, early stages of the coolant boiling).

APPROACHES TO KEEPING RECORD AND MONITORING FAST NEUTRONS FLUENCE AT THE PRESSURE VESSELS OF WATER-COOLED POWER REACTORS (VVER) AND THE EXPERIENCE OF USING THEM AS PART OF THE PROCEDURE FOR MONITORING VVER EQUIPMENT RADIATION BURDEN

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An important prerequisite for assessing the burn-up life of unrepairable equipment of VVER-type reactors, in particular, the reactor pressure vessel (RPV), is the calculation of the values of the radiation burden parameters (fast neutrons fluence (FNF), fluence accumulation rate, number of dislocations per atom).

According to the requirements of regulatory documents (RD) for NPPs, the neutrons fluence at the reactor pressure vessel shall be kept starting from the start-up and commissioning stages, which determines the endurance operating life of the reactor pressure vessel in compliance with the strength analyses and specifications. In order to implement the requirements of RDs for NPPs with VVER-type reactors, a procedure for fast neutrons fluence monitoring is being incorporated in accordance with the provisions of RD EO 1.1.2.29.0913-2012. Availability of such procedure, including the calculation method, helps arrange for keeping record of fluence and its independent assessment at NPPs.

The monitoring procedure currently undergoes operational testing at Kolskaya NPP for independent assessment of FNF at the VVER-440 RPV in the course of planning the core charging taking into account incorporation of new types of fuel. The results of the testing at Kolskaya NPP have been taken into account in improving the procedure and its application to the calculations of FNF at RPVs of VVER-1000-type reactors.

The procedure for monitoring characteristics of the field of neutrons is based on FNF expected estimates, which need to be experimentally confirmed.

In order to implement the procedure for monitoring, it is necessary to maintain a database sufficient for keeping record of FNF at RPVs.

The report presents the results of approbation of the general procedure for calculating FNF at the VVER reactor pressure vessel by means of comparing results of calculations and measurements at the external surface of pressure vessels of VVER-440 and VVER-1000, and demonstrates a possibility of maintaining a database for keeping record of FNF.

On the other hand, when in use, the improvement of fuel arrangement to reduce the rate of fluence accumulation at RPV as well as due to incorporation of new types of fuel, it is necessary to additionally assess the parameters of radiation burden at other reactor equipment (in-vessel internals (in particular, partition), reactor support structures). Besides, in order to prove authenticity of the obtained FNF distributions, it is also necessary to analyze the actual reactor data taking into account the results of monitoring with respect to in-core monitoring system (ICMS) readings.

RESULTS OF THE RESEARCH ON A LEAKY FUEL ELEMENT WITH FUEL PELLET WITHOUT THE CENTRAL HOLE THAT HAS BEEN USED AS PART OF ALFA FAAD UP TO BURN-OUT OF 41 MW×day/kgU

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Post-irradiation examination of a leaky FAAD (fuel assembly of alternative design),ALFA No.ID02293 VVER-1000, used at Unit 1 of Kalininskaya NPP during 3 fuel campaigns (fuel channels 25-27) up to the medium fuel burn-out of 42 MW×day/kgU was carried out in 2015. FAAD-ALFA has been recognized as leaky following the results of cladding integrity monitoring in FC 27. This fuel assembly was furnished with fuel elements featuring increased uranium charge having pellets of typical size of 7.8x0 mm and thinned cladding of 9.1x7.93 mm. The cladding was made of E110opt. alloy based on sponge zirconium manufactured by Wah Chang.

Fuel element 128 with the signs of leakage was revealed in the peripheral row of the fuel assembly bundle proceeding after the visual inspection conducted as part of the initial examination of FAAD-ALFA. Further examination established that the reason for the loss of integrity was through debris-caused damage by a wire trapped in DR5 cell. A fragment of this wire was found in a damaged cell located near the cell of the leaky fuel element. The results of metallographic examination show that the length of defect was 300 μm, maximum width was 167 μm. The fuel element collapsed at the distance of ~650 mm from the lower edge during its removal.

The diameter of the leaky fuel element was increased as compared with the reference fuel element. The diameter increase reaches 200 μm in some sections. The decrease of count rate with respect to ^{137}Cs in a sector with coordinates of 500-3250 mm of the leaky fuel element as compared with the reference fuel element registered after considering the results of gamma scanning testifies to emission of cesium from the leaky fuel element. The average value of emission has amounted to 6%, the maximum emission equaled 10%. At that, no fuel wash-out from the fuel element was registered.

The cladding of the leaky fuel element is strongly hydrogenated and embrittled through its entire length. The through cracks have appeared in some sectors. The oxide film at the external surface is even, its thickness does not exceed 8 μm. The oxide film thickness reaches 300 μm in some sectors of the internal surface.

The structure of fuel of the leaky fuel element has suffered significant changes as compared with the reference fuel element. A circular area of cracking has formed at the distance of ~ 0.7 of the radius, dividing fuel into two parts, central and peripheral. The content of intragranular and grain-boundary pores is increased in the central part.

RESULTS OF INVESTIGATING FUEL ELEMENTS VVER-1000 WITH INCREASED URANIUM CHARGING AND CLADDING OF E110 AND E110opt ALLOYS BASED ON SPONGE ZIRCONIUM

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Two fuel assemblies of VVER-1000-type reactor: FAAD (fuel assembly of alternative design)-ALFA and FAAD-5M with increased uranium charging that were used at Unit 1 of Kalininskaya NPP up to the medium fuel burn-out of 42 MW·day/kgU and 59.4 MW·day/kgU accordingly were delivered in May 2015 to SRC SRINR JSC for post-irradiation examination.

FAAD-ALFA is furnished with fuel elements featuring thinned cladding of optimized alloy E110 (E110opt) based on sponge, and fuel pellets without the central hole. The fuel elements of two modifications are used in FAAD-5M: 1) with thinned cladding of E110 alloy based on sponge, with fuel pellets without the central hole, 2) with regular cladding dimension and a fuel pellet with the central hole.

The examination was carried out with the aim of determining the fuel elements condition after operation, studying the change in their functional characteristics.

The examination was carried out by means of non-destructive and destructive testing. The geometry of the fuel elements, the thickness of the oxide film at the external surface of the cladding, the amount and composition of the gas inside of the fuel elements were measured. The fuel micro-structural characteristics, distribution of fission products along the length and radius of fuel element column, parameters of structural and phase state, and mechanical properties of the cladding were determined.

The results of profilometry testify to the fact that the mechanical interaction between the fuel element column and the cladding, and, as a consequence, emergence of initial distortion and corrugations at the cladding, takes place in the fuel elements featuring increased uranium charging with lower fuel burn-out, than with the fuel elements of basic design.

The elongation of both types of FAAD-5M fuel elements does not practically differ from the elongation of the fuel elements of basic design. The fuel elements FAAD-ALFA with cladding of E110opt alloy based on sponge have elongated on average by 10 mm more than the fuel elements of basic design with the comparable fuel burn-outs.

No difference was revealed in the degree of fuel elements oxidation with cladding of E110opt. and E110 alloy based on sponge zirconium as compared with the fuel elements of E110 alloy on electrolythic basis.

No essential difference was observed with respect to emission of gaseous nuclear fission products (GNFP) from the fuel element column under the cladding of fuel elements with increased uranium charging and those of basic design.

The examination results show that the service life of fuel elements with the increased uranium charging and thinned cladding of E110opt. and E110 alloys based on sponge zirconium is not completed with respect to the basic operational characteristics: change of the geometry, corrosion degree and mechanical properties of the cladding, and emission of gaseous nuclear fission products.

Some aspects of operation of power units with VVER-1000 in steady-state and transient modes after transition to increased power level up to 104% of the nominal power

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The report presents operational data of the power units of Kalininskaya NPP in the period of pilot operation at higher power level.

The first part of the work is devoted to the analysis of transient mode of Unit 4 shutdown due to a defect in the operation of the control unit of one of the two turbine feed pumps. The analysis has shown that the exclusion of shutdown rod (SR) interlocking by the level in the steam generator (SG) after stoppage of the corresponding main circulation pump (MCP) [1] complicates the transient mode, and it is highly probable that the VVER-1000 reactor will transfer to the mode of natural coolant circulation (NCC). At that, there exists a probability of superheated coolant release from the pressure compensator into the hot header of steam generator No.4 and hot line No.4 of the reactor loop in case of the absence of the coolant flow in this loop. Then, boiling-up of a part of this coolant and locking of loop No.4 take place in the NCC mode.

Consideration is also given to the mode of operation of in-core instrumentation system detector assembly of neutron measurement temperature channels with level indicator in the SR and NCC mode [2].

The proposals are given for improving the algorithm of ACS operation for this mode and for calculation of the boiler water volume in the SG due to faulty operation of the SG level gauges in transient modes in case of rapid change in the SG power level.

The proposals are given for updating the algorithm of operation of the in-core instrumentation system detector assembly of neutron measurement temperature channels with level indicator to exclude false actuation for determining the steam volume in the reactor upper part [2].

The second part of the report is devoted to operation of FAAD (fuel assembly of alternative design)-12PLUS with intensifier grids (IG). It has been experimentally shown, with the use of data of in-core instrumentation system detector assembly of neutron measurement temperature channels with level indicator [2, 3, 4], that full coolant mixing is effected in the upper part of fuel elements bundle of FAAD-12PLUS in the reactor core of VVER-1000 of FAAD-12PLUS with IG. This feature of FAAD-12PLUS removes the "hot" channels in the fuel assembly section in the upper part of fuel assembly, establishes a plane temperature field of the coolant in the fuel elements bundle of FAAD-12PLUS at the exit from the fuel assembly core. The hydraulic resistance of FAAD-12PLUS is linear to the power of the fuel assembly both in the central and peripheral rows of the fuel assembly active core.

This feature of FAAD-12PLUS brings about the fact that the resistance of FAAD-12PLUS and FAAD-PLUS become identical in the fuel assembly active core with fuel assembly capacities exceeding 23 MW (Fig. 2).

The hydraulic resistance of FAAD-12PLUS and FAAD-12PLUS is linear up to the fuel assembly power of 25 MW in the central region of the core (Fig.1).

According to our assessments, FAAD-12PLUS can be operated in the peripheral rows of the core featuring power of 29 MW, while FAAD-PLUS can be operated at 26 MW (Fig.3) at the operational limit with respect to fuel assembly heating equal to 50^0C.

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THE CONCEPT OF ALARM SYSTEM IMPLEMENTATION AND PROVIDING OPERATOR WITH RELEVANT INFORMATION

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From the technical point of view, an NPP is a complex automation facility. The greater part of the actions aimed to control the NPP operation is performed in the automatic mode (interlockings, protection, automatic transfer switches, incremental programs, etc.). However, there remain a number of tasks that cannot be performed without an operator (transition from one mode of operation to another one, accident management, troubleshooting the defects in instrumentation and automation, its ageing, etc.). In such cases, the NPP operation safety is directly dependent on the efficiency of organizing the operator’s informational space.

Currently, the basis of the operator’s informational space are the traditional unit switchboard control panels and automated workstations of the up-to-date computer-control systems.

The informational space based on traditional control panels features a number of advantages and drawbacks. The main advantage consists in visualization of the information. The operator is able to assess the condition of power unit, main process and electrotechnical equipment at a glance, and not a single alarm, or change of the condition at the panels will be left unnoticed. However, traditional panels have a significant drawback, which is the limited scope of the provided information.

The use of advanced digital technologies in the NPP automatic process control system (APCS) helped considerably increase the level of informational support at NPPs. A number of signals transmitted to the automated operator’s workstations have increased thousandfold.

However, since 2004, in spite of considerable changes related to the scientific and technical development of digital technologies, the requirements for the alarm systems have remained unchanged. As a result, a contradiction between a big amount of information, which is presented to the operator, and the limited psychophysical capability for its perception has increased and continues to increase.

In view of this fact, for the sake of creating an efficient alarm system in advanced APCSs, a necessity has appeared not only to reconsider the approaches to designing a alarm system (mechanisms of classification, selection, filtration, algorithms of alarm suppression, etc.), but also to take new factors into account:

- physiological capability of the operator and his place in the alarm system;
- mode of the operator’ work (strategic, tactical, opportunistic);
- measurement of the alarm system readings;
- keeping record of the NPP and equipment mode of operation;

- alarm system control at all lifecycle stages.

This report will present a concept of implementation of an alarm system taking into account the provisions of the latest international regulatory technical documentation, specific features of advanced NPP APCSs, experience of NPP operation in Russia and abroad, as well as the cutting-edge experience of NPP construction and commissioning.

WORKS PACKAGE FOR SUBSTANTIATION OF OPERATIONAL FITNESS OF THE CONTAINMENT VESSEL OF POWER UNIT NO.3 OF ROSTOVSKAYA NPP

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The work package for substantiating operational fitness of the containment vessel has been performed for timely putting power unit No.3 of Rostovskaya NPP into operation. The following works were performed:

- monitoring of the containment vessel during its erection, preloading and acceptance testing;
- full-scale observation of the containment vessel during the preloading and acceptance testing;
- adaptation of the expert system for assessment of stress-strain behavior of the containment vessels of NPP units with VVER-type reactors to power unit No. 3 of Rostovskaya NPP.

The works to monitor the containment vessel of power unit No. 3 of Rostovskaya NPP have been carried out by the Nuclear Safety Institute, Russian Academy of Sciences, together with the Volgodonsk Engineering and Technical Institute, affiliated a branch of National Nuclear Research University of the Moscow Engineering Physics Institute, for the first time in NPP construction practice.

It was noted during the monitoring that the technology used at the initial stage of the containment vessel preloading had not provided the design efforts in the prestressing strands. In view of this fact, the Nuclear Safety Institute, Russian Academy of Sciences, made a proposal to change the technique used for the tensioning of the prestressing strands of the preloading system of the containment vessel of Rostovskaya NPP power unit No. 3. The proposed changes in the technique of tensioning the prestressing strands made it possible to increase the amount of effort at the anchor and bring it to the design value as much as possible.

The works were followed by a consolidated report. The information system of monitoring the containment vessel of power unit No. 3 of Rostovskaya NPP was developed, including all the documentation received during the erection, preloading and acceptance testing of the containment vessel.

The full-scale observations conducted during acceptance testing showed that the width of surface cracks opening at the external surface of concrete of the containment vessel, which meets the standard requirements when exposed to the testing internal pressure and after pressure release was not higher than 0.3 mm. The calculation of strain-stress state of the containment vessel after its preloading and acceptance testing showed that the attained design level of the preliminary structure stress with the use of the technique proposed by the Nuclear Safety Institute, Russian Academy of Sciences, ensures full operational fitness of the containment vessel of Rostovskaya NPP power unit No. 3 .

The materials received during performance of the works on monitoring and full-scale observations of the containment vessel of Rostovskaya NPP power unit No. 3 passed expert examination in the Federal Service of Environmental, Technological and Nuclear Supervision (Rostekhnadzor). Based on these documents, the permission for operation of the containment vessel was received.

The design-basis justification of the maximum permissible losses in the course of anchoring the prestressing strands of the system of preloading the containment vessel was with the use of the expert system for assessment of the strain-stress state of the NPP containment vessels. Based on the results, changes were introduced into the safety report (SR) for power unit No. 3 of Rostovskaya NPP.

Topical area
OPERATION OF NPPS WITH RBMK, BN AND EGP-6 REACTORS

ISSUES IN MODELING DEFORMATION OF GRAPHITE STACKING IN HIGH-POWER CHANNEL-TYPE REACTOR (RBMK)

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One of the main directions of restoration of durability characteristics (RDC) of the high-power channel-type reactor (RBMK) is the repair of graphite stacking by means of cutting the graphite blocks into 2 or 4 parts in a limited number of graphite columns (cells). In order to determine the time of repair start and its efficiency, the deflections of channels are measured, also in a limited number of cells. An important task is predicting the reactor operation time till the moment when the repairs are required and after the repairs. The restricting parameter is the maximum permissible value of channel deflection.

The prediction estimates of deformations are made both on the basis of the sequence of measurements used to determine the rate of deflections propagation, and with the use of computer models comprising information on the physics of the processes taking place in graphite during its irradiation, cracking and mechanical interaction of graphite blocks, graphite columns and channels. Currently, there is a whole number of models (computer codes) of graphite stacking behavior differing in the approaches to the described processes and the detailization.

The report analyzes some aspects of stacking deformations and the related issues. The main attention has been paid to the impact of uncertainties on the accuracy of predicting the channel deflections. The uncertainties can be caused both by the inaccuracy of the source data (fluence, temperature of graphite irradiation), as well as by the dispersion of graphite properties, which determines the moment of appearance of a longitudinal crack in an individual graphite block and the degree of its opening. Besides, there are a number of random factors that are impossible to predict, but can essentially influence the distribution of deflections ("field of deflections") in the horizontal section of stacking. Such factors include: orientation of cracks at the edges of blocks in different cells, displacements of blocks in perpendicular directions and their engagement, degree of joining the cut halves of the blocks, etc.

In order to increase the authenticity of the estimates, the information on the measured deflections in some cells is used in some codes for their preliminary adjustment. However, the number of measurements is limited, as a rule, by several hundreds of cells, which does not make it possible to restore the source field of deflections to the full extent.

The report assesses the impact of the above factors on the results of calculating deflections of channels as exemplified by a stochastic model (program GRAD), and comparison with the measurement results is made. The possibilities of model improvement for increasing the accuracy of describing deformations of the graphite stacking at high-power channel-type reactor (RBMK) are assessed.

STRUCTURAL FACTORS DETERMINING RADIATION STABILITY OF REACTOR GRAPHITE

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Currently, 30 power units are operated in Russia at ten NPPs, including 11 power units with reactors of RBMK-1000 type. LNPP-1 is the first built unit of this type. It has been part of the Russian power-generating system for more than forty years now.

It is known that radiation deformation of graphite takes place due to neutron exposure, which, after a time, results in full degradation of its structure, and then graphite cannot serve as a structural material anymore.

The essence of the graphite destruction mechanism is that the crystallites of various dimensions swell at different rates, which results in internal radiation and thermal stress in the material and the cracking of graphite blocks, formation of longitudinal cracks ultimately leading to the expiry of the service life of the entire graphite stacking.

The repair works carried out at LNPP-1 in 2013 for restoration of durability characteristics (RDC) significantly extended the actual service life of this power unit. Nevertheless, currently we have no understanding of the mechanisms of structural transformations in graphite due to operational factors (neutron exposure, temperature mode of stacking, etc.) causing degradation of mechanical properties of some blocks of the stacking. The purpose of this paper was to establish the interrelation between structure degradation and graphite properties at different stages of operation.

The research was conducted using the methods of structural analysis (TEM, SEM) of the structure of cores taken from the graphite blocks of the stacking at LNPP-1, neutrons irradiated to different fluences (with the maximum fluence of neutrons being $25 \cdot 10^{25} m^{-2}$) in the temperature range of 500-700°C.

It was demonstrated that the deterioration of graphite strength properties at reaching the critical fluence of neutrons was caused by the destruction of boundaries of filler-binder type (formation of a net of micro-cracks).

It was established that the evolution of graphite microstructure in the process of irradiation causes changes in such macroscopic properties as graphite density and ultimate compressive strength. Moreover, these mechanisms can be divided into two stages. . Graphite shrinkage takes place at the first stage (which is accompanied by the increase in density and ultimate strength), then, after passing the maximum value, the stage of secondary swelling and reduction of strength properties begins. Finally, the cracks grow, which results in the deterioration of graphite strength properties as a whole.

The paper also demonstrates that an important factor, which should be taken into account, is the non-uniformity of graphite properties, especially when comparing blocks from the layers of different batches of graphite blocks. In this case, the properties dispersion can amount up to ~ 200%.

RESTORATION OF DURABILITY CHARACTERISTICS (RDC) OF GRAPHITE STACKING OF POWER UNITS 1, 2 OF LENINGRADSKAYA NPP

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Introduction

A characteristic feature of RBMK-1000 reactors at the final stage of the operation is the development of degradation processes in the reactor structural units:

- graphite blocks distortion and cracking;
- axial shrinkage of graphite columns;
- axial radiation and thermal creepage of zirconium parts of fuel channel (FC);
- radial radiation and thermal creepage of zirconium parts of FC;
- change in physical and mechanical properties of the stacking graphite (strength, density, thermal conductivity) [1].

The criteria of safe operation of the reactor core elements and reactor structures influencing the service life of the reactor and the entire power unit have been defined by the effective regulations, methods and safety justifications [2].

The main method to control the service life of the graphite stacking of RBMK-1000 in terms of adherence to the threshold limit values of amount of deflection of FC and FA of the control rods is the use of the developed technique of restoration repairs. The RDC technique tested in 2012-2013 at power unit No.1 of Leningradskaya NPP was successfully used in 2014 at power unit No.2 of Kurskaya NPP and No.2 of Leningradskaya NPP; the second cycle of RDC was performed at power unit No.1 of Leningradskaya NPP.

The second important criterion is preserving engagement in the HARPS assembly.

The processes related to the axial shrinkage of graphite columns and radiation and thermal creepage have been studied quite thoroughly by now, and are predicted with an acceptable accuracy.

Conclusions

1. In order to ensure the best possible planning and optimization of repair campaigns of power units at Leningradskaya NPP, it is necessary to continue improvement of the RDC technique, and the tools to perform the works.

2. It is necessary to continue works for improving analytical models of graphite stacking distortion;

3. It is required to continue works on the development and establishment of online systems for checking FC geometry in during the power operation of the reactor;

4. Investigate the properties of graphite stacking of the RBMK-1000 reactor with the aim of finding similar materials and selecting the best cutting tools and cutting conditions.

SYSTEMS OF MONITORING DEFORMATION OF GRAPHITE COLUMNS OF REACTOR UNIT RBMK-1000 BASED ON FIBER-OPTIC SENSORS

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Beginning from 2008 Prolog LLC works on the radiation-resistant fiber-optic sensors based on Bragg grids have been carried out. The major advantages of such sensors are as follows:

- capability of functioning in high gamma and neutron fields;
- absence of electric parts submerged into reactor core;
- high sensitivity;
- high reliability as a result of absence of movable elements.

The use of these sensors is possible in the instruments for measuring temperature, pressure as well as for linear measurements.

IPO-45 system for measuring deflection of the graphite stacking of RBMK-1000 reactors has been developed by the present time on the basis of radiation-resistant FO sensors. The system certified as a measurement device, which has the qualified methods of measurements, has been introduced and is used at Leningradskaya and Kurskaya NPP.

IKS-49 system intended for measuring deflection of the graphite stacking without fuel assembly discharging from the reactor has been developed presently, passed acceptance tests and is in experimental service at the NPP. The measurements are provided through a hole with diameter of 6 mm in FA carrying tube (assy 49).

The introduction of IKS-49 system will help perform measurements for assessing degree of deformation of reactor core of RBMK-1000 without losing time for withdrawal and subsequent charging of fuel, which will essentially accelerate the RU diagnostics.

The described systems operate under condition of shutdown reactor during repair.

Presently the works are underway to build a system working in the reactor core during its power operation.

Differences of operational conditions of such system from the systems already existing:

- high temperatures (up to 400°C taking into account radiation heating up);
- high neutron fields (in FA center).

This system will help perform periodic monitoring of the process of reactor core deflection in the course of reactor power operation, which will increase safety of RU operation, provide necessary knowledge about dynamics of measuring deflections, help more amply use the residual RU service life up to attaining the limiting values of deflections.

USE OF CODES OF IMPROVED ASSESSMENT FOR INDEPENDENT REVIEW OF JUSTIFICATIONS OF SAFETY OF FAST NEUTRON REACTORS WITH LIQUID-METAL COOLANT

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Fast neutron reactors with sodium coolant (BN type) have operated in our country for more than 30 years. Despite its vast experience in expert evaluation of the safety justification of these reactors as part of the licensing process, as an organization of scientific and technical support of the Regulatory Body (Rostekhnadzor - Federal Service for Environmental, Engineering & Nuclear Supervision), Federal State-Funded Institution RDC NRS, has not possessed a full-scale calculation toolkit for independent assessment of the safety of this type of reactors in their steady-state and transient modes. The expert evaluation was based solely on the conservative assessments.

Recently, due to cooperation with the experts from GRS (Germany), the neutron and physical and thermohydraulic codes have become available and are being developed at RDC NRS for the calculation of fast neutron reactors with the liquid-metal coolant. A necessity for licensing support of power unit No. 4 of Beloyarskaya NPP with reactor unit BN-800 put into pilot operation has become the operative motive for such development. Moreover, the nearest prospect of licensing BN-1200 reactor, which is a so-called Generation IV reactor, as well as BREST-OD-300 and SVBR-100 reactors with lead and lead-bismuth coolants demands that the organization supporting the regulatory body used advanced codes of improved evaluation for independent analysis of different modes of operating of such reactors. It should be noted that the intensive work to build the demonstration Generation IV reactors with liquid-metal coolant is also underway in other countries (PRIZM in the USA, MYRRHA in Belgium, European reactor ALFRED, which is scheduled to be constructed in Romania, French demonstration reactor ASTRID built in consortium with the American and West-European partners, etc.)

The experience in verification of the calculation codes on the domestic critical assemblies of the large physical test facility (BFS), and the new calculation benchmarks established abroad help carry out verification of codes, including improved.

This paper provides the results of verification of stochastic codes SCALE and SERPENT, as well as deterministic codes DYN3D and BARS in terms of reactors with sodium coolant.

ANALYSIS OF COMPLIANCE OF THE RESULTS OF MEASURING NEUTRON AND PHYSICAL AND THERMOHYDRAULIC CHARACTERISTICS OF RBMK-1000 UNITS AFTER THE RESTORATION OF DURABILITY CHARACTERISTICS OF GRAPHITE STACKING TO THE PREDICTION CALCULATIONS

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During massive deformations of the graphite stacking at the RBMK-1000 power units, the authors have made the predictive estimates of the dynamics of the neutron and physical and thermohydraulic characteristics of the reactors after the works on restoration of durability characteristics of the graphite stacking. The predictive estimates have taken into consideration the fact that the repairs of graphite stacking cause the loss of part of the stacking graphite mass. The early predictive estimates have been performed for the extreme events: in case of the loss of 8% and 12% of the total mass of the stacking graphite.

However, the actual graphite losses even for power unit No. 1 of Leningradskaya NPP are still far from the limiting losses after completion of the 2nd stage of the stacking repairs. The results of the measurements conducted after completing the next stage of the repairs help assess the correctness of the predictive estimates of the dynamics in neutron and physical and thermohydraulic characteristics.

This paper provides an analysis carried out for the 1st phase power units of e of Leningradskaya and Kurskaya NPPs. The computer code of improved assessment BARS has been used when making the predictive estimates. The verification of the analytical model for the current charges of RBMK-1000 was performed based on the results of scanning the power density fields at the RBMK-1000 power units of Leningradskaya NPP in 2012-2015.

The current and predicted earlier calculations of the neutron and physical and thermohydraulic characteristics of reactors have been compared with the results of measurements performed at the 1st phase power units of Leningradskaya and Kurskaya NPPs after completing the next stage of the repair works on restoration of the graphite stacking.

PROSPECTS OF DEVELOPMENT OF BN-800 REACTOR CORE

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One of the basic purposes of creating a BN-800 reactor is the optimization of technological elements of the closed fuel cycle of nuclear power engineering with the use of mixed uranium-plutonium fuel. Since the issue of works organization for the production of the mixed oxide fuel (MOX) has been solved with some delay with respect to planning the construction works for the BN-800 reactor building, initial fuel charging for BN-800 reactor basically consists of the uranium oxide fuel, and only some part of the fuel assembly (FA) (16% of the total amount) comprises MOX-fuel produced at experimental production facilities of PA MAYAK and State Science Center Nuclear Reactor Research Institute JSC. Such reactor core has been named hybrid, taking into account the peculiarity of its furnishing. The scheduled period of hybrid reactor core operation equals 4 micro-campaigns (4×155 eff. days), which will be followed by step-by-step replacement of the FA of the hybrid area with the FA with MOX-fuel produced at the facility established at the Gas Chemical Facility.

The reactor core fully loaded with MOX-fuel will be formed after the sixth recharging (for the seventh micro-campaign). In this case, three types of FAs (the same as in the hybrid reactor core), with regard to the contents of fission material, are used (conventionally, by the "enrichment degree"): 18.2%, 20.1%, and 23.0%, as far as the high-background plutonium is concerned. In this case, the boundaries of enrichment areas in the hybrid reactor core and area with full MOX-fuel charging coincide. The duration of the reactor core FA campaign equals 465 eff. days (3×155 eff. days).

The main factor limiting the FA campaign is the radiation stability of structural material of fuel elements cladding. BN-800 reactor core has been developed with the use of steel ChS-68 for fuel elements cladding, the same as in the regular FA of BN-600. F elements with such cladding are serviceable ensured (based on the experience in BN-600 operation) up to the damaging dose ~ 90 dpa, which corresponds to the maximum (local) burn-out of MOX-fuel in BN-800 of 9.7% ha at the average value of the discharged fuel of 6.8 % ha (66 MW×day/kg).

The developed R&D program intended to increase duration of the fuel campaign of BN reactors envisages changeover to the use of EK-164 steel, which is more radiation-resistant if used as the material for the cladding of fuel elements. Based on the results of using the pilot FA in BN-600 with the cladding of fuel elements made of EK-164 steel their serviceability is justified at damaging doses not less than 110 dpa. In case of adopting 112 dpa as the threshold damaging dose it is possible to increase the duration of FA campaign up to 580 eff. days. In order to substantiate such campaign extension, it is necessary to carry out tests and examination of reference FAs (for changing the material of the cladding of the fuel elements being part of regular FAs).

The further prospects of development of BN-800 reactor core get associated with the development and implementation of a new reactor core design with an axial layer of the fertile material (dioxide of depleted uranium). The advantage of such option of the reactor core is the reduction of neutron flux density therein, and, consequently, the damaging dose accumulation rate. Based on the results of preliminary research, there is a possibility of reduction of the damaging dose accumulation rate by ~15%. Accordingly, in this option, in case of using the EK-164 steel as the material for the cladding of fuel elements, it is possible to expect the extension of FA campaign up to ~660 eff. days. In order to justify the option of the reactor core with an axial layer, it is necessary to carry out the experimental research on the functionality of fuel elements as part of special experimental fuel assemblies as well as experimental research on the physical characteristics at the BFS large physical test facility.

The key characteristics of BN-800 reactor core at the main stages of its development are presented in Table 1.

Table 1. Prospects of increasing fuel burn-out in BN-800 reactor

Characteristics	Hybrid reactor core (ChS-68)	Reactor core with MOX-fuel full loading		
		Reactor core 01D (ChS -68)	Future reactor core (EK-164)	Future reactor core With axial layer (EK-164)
FA campaign, eff.days	465	465	~580	~660
Interval between recharging sessions, eff.days	155	155	145	165
Maximum burn-out, % ha	9.8 / 8.1*	9.5	12.0	~14.5
Maximum damaging dose, dpa	78 / 74*	90	112	112
Average burn-out, MW·day/kg	64	66	83	~105

* FA with uranium fuel / FA with MOX-fuel

The further increase in the fuel burn-out in BN-800 reactor is potentially possible in case of using steels of ferrite-martensite class for the cladding of fuel elements.

NEUTRON AND PHYSICAL CHARACTERISTICS OF RMBK-1000 (HIGH-POWER CHANNEL-TYPE REACTOR) IN CASE OF CHANGING PROPERTIES OF GRAPHITE STACKING

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Nowadays, the technique of restoration of durable characteristics of the graphite stacking for RBMK-1000 has been developed and tested. The technique is based on cutting and sizing graphite blocks, which helps create free space for compensation of graphite blocks deformation caused by the radiation damage and ensures reduced deflection of the reactor fuel channels.

The removal of the part of graphite moderator during repairs, change of the graphite stacking properties as a result of shrinkage, and changes in the diameters of the reactor fuel channels affect the neutron and physical characteristics of RBMK-1000.

The report describes the methods of keeping record of the changed properties of the graphite stacking and diameters of the reactor fuel channels when conducting calculations of neutron and physical characteristics according to the engineering and precision calculation codes, and presents the results of their verification.

The report provides the results of design investigations of changing the neutron and physical characteristics under conditions of step-by-step repairs of the graphite stacking and possibility of maintaining them within specified limits due to the changed composition of charging the reactor cores of RBMK-1000.

The results of calculations of neutron and physical characteristics have been compared with the measurement results at power units of Kurskaya NPP and Leningradskaya NPP, which have proven the possibility of maintaining the neutron and physical characteristics within the specified limits after the first and the second stages of the works for restoration of graphite stacking durability characteristics.

Based on the experience of preparing nuclear safety justifications during repair works and design assessments of changing neutron and physical characteristics depending on the quantity of the repaired cells, the proposals have been formulated with respect to the scope of the measurements of neutron and physical characteristics sufficient to justify the nuclear safety during operation.

EXPERIENCE IN PUTTING INTO OPERATION DIAGNOSTICS SYSTEM OF REACTOR CORE OF REACTOR BN-800

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The diagnostics system of the reactor core of the BN-800 reactor (RDS) of unit 4 of Beloyarskaya NPP is an automated system, which is part of the automatic process control system (APCS), intended for comprehensive monitoring of the processes taking place in the reactor in the modes of normal operation and deviations from normal operation, early detection of deviations from normal operation and damage of the reactor core (deformation of core elements, sodium boiling in fuel assembly (FA), fuel elements melting in FA, disturbed cooling of reactor core and other anomalies) [1-4].

RDS includes an abnormal reactivity detection system (ARDS), neutron and noise diagnostics system (NNDS), reactor core temperature monitoring system (RCTMS), integrated analysis system (IAS).

In case of putting RDS into operation at different power levels, the serviceability of measurement channels (MC) as part of RDS has been confirmed (temperature MC, MC of current of neutron chambers and current fluctuations, reactivity MC), determination of reactor neutron and physical characteristics has been carried out, and the efficiency of the functioning of system basic algorithms has been investigated:

- monitoring of abnormal reactivity on the basis of continuous determination of reactivity effects due to the changed reactor parameters, fuel burn-out, displacement of reactor control rods, etc.;
- monitoring of current values of sodium temperature over FA heads, at the inlet to and outlet from the reactor core, temperature fluctuations and power distribution in the heat-exchanging loops;
- monitoring of current fluctuations of neutron ionizing chambers;
- integrated analysis of diagnostic information and forming early indicators of disturbances in normal operation and damage of the reactor core on the basis of the readings of the systems included into RDS and operational information received from the APCS upper level systems (automated radiation monitoring system, system of upper block level).

Presently, the RDS system is in experimental service and functions regularly at unit 4 of Beloyarskaya NPP. A consideration is given to a possibility of expanding the system functions to perform calculation of the neutron and physical characteristics of BN-800 reactor core on the basis of RDS software and hardware facilities.

INFORMATIONAL, DIAGNOSTIC AND TECHNOLOGICAL SUPPORT OF NPP REPAIR CAMPAIGNS

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The activities on informational, diagnostic and technological support of NPP repair campaigns have been conducted by the experts of RDI NPEI (Volgodonsk) since 1981. Information databases containing the results of diagnostic and technological support of repair campaigns of operating NPP power units has been updated since 2000.

1. The objectives of diagnostic support are as follows:

- supporting operational reliability and safety of NPP equipment by timely identifying changes in the performance of motor-driven equipment and pipeline valves of NPPs (MDV) with the use of methods and means of online technical diagnostics;
- optimization of periods and types of the repairs and the costs of such repairs.

2. The objectives of technological support are increasing the quality repair campaigns by means of timely provision of the repair service with technical repair documentation.

3. The following tasks are solved during diagnostic support:

- search for defects, assessment of performance of electrically-driven equipment (refueling machine, pumps, ventilation units) and MDV;
- rapid analysis of the source diagnostic information, comparative analysis of the preceding results of diagnostic testing, compiling (filling) diagnostic certificates by means of online databases containing data on the equipment.

4. The informational, diagnostic and technological support of NPP repair campaigns allows the following:

- increase efficiency of planning maintenance and repairs of the NPP operating equipment;
- elaborate technically justified recommendations for adjusting time between repairs;

- carry out diagnostic certification of both the equipment and a number of process procedures (e.g., such as "Recharging NPP nuclear fuel from VVER-type reactors effected by means of reloading machines").

MAINTENANCE AND REPAIR OF NUCLEAR POWER PLANT COMPONENTS BASED ON THEIR TECHNICAL CONDITION BASIS. CONCEPT

V.D. Gurinovich, Yu.A.Yanchenko

A nuclear power plant is a technically complex facility. Every power unit includes a nuclear reactor, reactor and steam-turbine equipment, hundreds of pumps, tens of thousands of units of valves, thousands of kilometers of cables, large amount of pipelines, electrotechnical equipment, instrumentation and other components incorporated into systems to perform process functions for generation and transfer of heat and electric power, as well as functions to ensure nuclear, radiation, fire and technical safety. All NPP components, including buildings and structures, underground communications, etc. require constant monitoring and control of their technical condition, which implies timely maintenance and repairs, effective control of cost of ownership of nuclear power plant facilities. It necessitates using advanced approaches to NPP components life-cycle control at all stages by means of a system engineering method for addressing the issues of maintenance and repairs, starting from NPP and equipment design and to power units decommissioning.

The approach used at the national NPPs, which is based on scheduled repairs, is convenient from the point of view of future and current planning, preparation of repairs, because spare parts and materials can be delivered tender procedures can be conducted in time, etc. Nevertheless, it has a number of disadvantages. First of all, it means redundancy of of the scope works, which entails excess use of labor, time and financial resources, thus, reducing technical and economical performance of nuclear power plants. Therefore, the strategy of NPP components maintenance and repairs based on the technical condition of the components is a promising one. It is implemented primarily in two main directions based on:

- continuing or regular inspections, control with partial or full disassembly, monitoring, testing, checks of the components, analyses of obtained data, its evaluation and forecasting;
- quantitative analysis of reliability of the components and systems.

The combined use of these approaches is efficient.

The report gives consideration to organizational and technical improvements supported by the best foreign practices in maintenance and repairs taking into account requirements of the main federal safety rules and regulations (NP-001-15, NP-010-98, NP-044-03, NP-068-05, NP-089-15, etc.), requirements of EUR, documents of INPO, IAEA and WANO. The report also presents a brief analysis of organization of maintenance and repairs of foreign and national nuclear power plants. It articulates the objectives of the concept of maintenance and repairs of NPP components based on their technical condition, the key directions of attaining such objectives, and the existing limitations.

The concept implementation will help optimize to a certain extent business processes of NPP components maintenance and repairs on the basis of the *safety - reliability – economic efficiency* criterion, increase competitiveness of Russian NPP designs in the international market due to better adherence to the current requirements for organization of maintenance and repairs of nuclear power plants.

ASSESSMENT OF RADIATION STABILITY AND STRENGTH CHARACTERISTICS OF BELZONA MATERIALS TO USE THEM IN REPAIRS OF SAFETY CLASS 2 EQUIPMENT: TANKS, SPENT FUEL STORAGE POOLS, OIL COOLERS AND OTHER EQUIPMENT OF NUCLEAR POWER PLANTS

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The Safety Class 2 equipment (tanks, spent fuel storage pools, oil coolers and other equipment of the nuclear power plants) operation experience shows that despite continuous monitoring of its condition, the equipment gets exposed to the corrosion wear and has defects of different size, which reduce its reliability and safety.

In order to solve the issue of removing defects and repairing equipment, it is reasonable to use the cold-curable paste-like polymer composite materials (PCM) BELZONA 1111 and 1611. It is reasonable to use liquid PCM BELZONA 5811 for corrosion protection of NPP equipment, since it is resistant to corrosion, erosion, mechanical impacts and chemical reagents.

In order to justify the use of PCM BELZONA for repairing safety class 2 equipment (tanks, spent fuel storage pools, oil coolers and other equipment of the nuclear power plants) under conditions of ionizing radiation, the tests have been conducted at Institute of Reactor Materials JSC (IRM JSC) in water medium in the field of γ- radiation up to accumulation of absorbed doses of ~100, ~500, ~800 and ~1,000 Mrad.

The testing results for PCM BELZONA 1111 and 1611 in the field of γ-radiation with respect to radiation stability indicators: tear strength and compression strength, meet the arbitration criteria of radiation stability according to GOST 25645.331-91 and GOST R 51102-97, which testifies to maintaining serviceability with respect to these indicators of materials 1111 and 1611 after the accumulation of absorbed radiation dose equal to 500 Mrad.

The obtained results make it possible to use materials BELZONA 1111 and BELZONA 1611 under conditions of ionizing radiation for repairing the spent fuel storage pools and radioactive waste tanks.

The testing for adhesion strength has demonstrated that PCM BELZONA 5811 retains its permissible values up to the accumulation of absorbed dose in the field of γ-radiation of 1,075 Mrad.

The obtained results make it possible to use material 5811 under conditions of ionizing radiation for protection of equipment of nuclear power plants from corrosion, erosion, aggressive impact of chemical reagents within the limits confirmed during testing of absorbed radiation doses.

Key words: cold-curable polymer composite materials, BELZONA, radiation stability in water in the field of gamma-radiation, absorbed dose, swelling, adhesion, adhesion tear strength, serviceability of repair putties and protective coatings.

ORGANIZATION AND FULFILLMENT OF ROSATOM PRODUCTION SYSTEM (RPS) PROJECTS FOR RESTORATION OF DURABILITY CHARACTERISTICS (RDC) OF POWER UNITS AT KURSK NPP

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1. Implementation of the RPS project for RDC of the reactor of Unit 2 at Kursk NPP in 2013-2014.

- primary analysis of the current condition;
- issues;
- solutions;
- final results.

2. Implementation of the PSR project for RDC of the reactor of Unit 1 at Kursk NPP in 2015-2016.

- primary analysis of the current condition;
- issues;
- solutions;
- final results.

3. Comparative analysis.

- rate of the work performance;
- expenses.

APPLICATION OF PROTECTIVE COATING TO INTERNAL SURFACE OF HIGH PRESSURE CYLINDER (HPC) BODY

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The erosive defects of the base material appear in the process of operation at the internal surface of bodies of wet-steam turbine HPCs. Repairs by means of cleaning and build-up welding of eroded places for the restoration of thickness and internal geometry of the body do not solve the issue of erosion metal wear to the full extent, and repeated defects appear in the process of further operation, which bring about the dramatic thinning of HPC body and even through defects.

One of the solutions to the issue of erosion metal wear at the internal surface of HPC body, taking into account extension of the service life and ensuring reliable and efficient operation of turbine generator sets is the technology of protective coating successfully used at all HPC bodies at Kola NPP (2007-2012) and at five HPC bodies at Kursk NPP (2011-2014).

The work on the application of protective coating to the surface of HPC body is performed in two stages:

1. Repairing of HPC body and preparing the body surface for protective coating application.
2. Applying protective coating.

The first stage includes the following works that can be fulfilled sequentially or in parallel:

- repair of HPC body in accordance with the requirements of regulatory documents for overhaul;
- preparation of the internal surface of HPC body in accordance with the "Process instructions for protective coating application".

The second stage includes application of protective coating per se.

In addition to the work on application of protective coating to HPCs, the Atomenergoremont's experts have extended the scope of use and now apply protective coating to other main and auxiliary equipment of the turbine shop (at Kola NPP), susceptible to erosion wear, viz:

- low-pressure heaters PND-4,5 in the area of branch pipes of heating steam inlet of 9 m^2;
- separator reheaters, heating steam supply chambers to the 1 and 2 stages of 3 m^2;
- bodies of non-return valves KOS-600 of 5 m^2;
- vertical sections of cold receivers of 36 m^2;
- water-film separators of 18 m^2;
- branch pipes for supplying heating steam to high pressure heaters PVD-6,7,8 of 1 m^2;
- restoration of geometry of rods, shafts, and spindles.

ORGANIZATION OF RBMK-1000 REACTOR PREPARATION FOR RESTORATION OF DURABILITY CHARACTERISTICS OF POWER UNITS AT KURSK NPP WITH SUBSEQUENT RESTORATION OF ESTABLISHED DIAGRAM

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The accelerated process of deformation of the graphite stacking and distortion of reactor channels as a result of radiation and thermal damage to the graphite structure have been registered at the first-generation power units at NPPs with RBMK reactors.

In order to solve the tasks of efficient elimination of deformation of the graphite stacking and damage to the graphite structure, a technique for restoration of durability characteristics of the graphite stacking by means of longitudinal cutting of a limited quantity of graphite columns has been developed and implemented.

In order to implement this technique, it is necessary to perform the following operations:

- Preparation of RBMK-1000 reactor for RDC with subsequent restoration of the established diagram;
- Operations for restoration of durability characteristics of RBMK-1000 reactor per se (measurement of sagging deflection, examination of the graphite column, longitudinal cutting of graphite units with removal of cutting products, inspection, force impact on the cells, measurement of sagging deflections, adjustment of the fragmented GC).

Preparation of RBMK-1000 reactor for RDC with subsequent restoration of the established diagram includes the following basic operations:

- Preparatory works;
- Dismantling the upper circuit holder;
- Dismantling the return bend;
- Dismantling the process channel;
- Adjusting the graphite stacking;
- Restoration of HARPS compensation capability;
- Installation of the process channel;
- Installation of the return bend;
- Installation of the upper circuit holder;
- Preparation of the reactor for operation.

One of the important aspects of achieving high results of the works on restoration of durability characteristics is the optimization of process procedures with the use of the Rosatom production system.

OPTIMIZATION OF PROCESS OPERATIONS ON REPLACEMENT OF CONDENSER PTU-14 OF POWER UNIT 5 AT NOVOVORONEZH NPP

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- 1 Purpose.
- 2 Location, dimensions, quantity.
- 3 Issues in the operation of condensers of K-1000-60/1500 turbines.
- 4 Corrective measures.
- 5 Replacement of condenser PTU-14 in 2015 taking into account the experience in replacement of condenser PTU-13 in 2014.

CONCERNING THE STATE AND THE PROSPECTS OF OPERATIONAL CONDITIONS OF AUSTENITE PIPELINES, DN300 AT NPP WITH RBMK-1000 REACTOR

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The report presents the basic results of using technologies for preventing formation and propagation of the intergranular stress corrosion cracking (ISCC) process in the weld-adjacent areas of welded connections (WC) of austenite pipelines DN300 (WC DN300) at NPPs with RBMK-1000 reactors.

Taking into account that the technology of mechanical redistribution of the residual welding stresses (MSIP technology) has become the most wide-spread at the NPPs with RBMK-1000 reactors as a countermeasure against the ISCC process in WC DN300, the report gives consideration to the prospects of changing conditions of the WC DN300 operation after the use of MSIP technology, including extension of the scope of the technology incorporation.

The results of analysis of the MSIP technology application for WC DN300 at NPP power units with RBMK-1000 reactors prove the reduction in the defect rate level in WC DN300 of "pipe – pipe" type in the process of operation due to stopped ISCC mechanism in the weld-adjacent areas at the inner surface of pipelines in case of transformation of the strain residual welding stresses into compressive stresses.

Currently, more than nine thousand and seven hundred WC DN300 units (which makes nearly 60% of all WC DN300 units) are operated with no reasons for the formation and propagation of ISCC cracks after the use of the MSIP technology, which is the basis for decreasing the scope of non-destructive flaw-detection (usually, ultrasonic) inspection and increase in the frequency of regular inspection of such WC DN300 units.

According to the opinion of authors, the calculation and experimental work fulfilled by this time for justification of the MSIP technology application for WC DN300 of "pipe – branch pipe" type in combination with the practice of USI application proves the necessity of the MSIP technology incorporation as an efficient measure for preventing the formation and propagation of ISCC cracks for such WCs. No additional expenditures and expenses are required as compared with the MSIP technology for WC DN300 of "pipe – pipe" type.

It should be noted that the cost of works and measures related to the provision of performing ultrasonic inspection of WC DN300 at all power units of NPPs with RBMK-1000 reactors approximates 70 billion rubles per year.

The concluding part of the report provides recommendations on incorporating the results of the MSIP technology application aimed both at the flawless operation of austenite pipelines DN300 at power units of NPPs with RMK-1000 reactors and at the significant reduction in material expenses, as well as radiation burden on the personnel in the course of fulfilling the operational non-destructive inspection of WC DN300.

SELECTION OF PAINTWORK MATERIALS FOR CORROSION PROTECTION OF NPP HEAT-EXCHANGING EQUIPMENT

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Introduction

The conditions for operation of NPP heat-exchanging equipment (HEE) determine quite a particular set of properties, which shall be possessed by the paintwork materials and paintwork coatings being formed to ensure reliable and durable operation.

Main properties of paintwork materials

Chemical properties: content of basic substance, content of individual components, content of volatile and non-volatile substances, water-soluble salts, water, ashes, etc., acid number, pH, etc.

Physical and chemical properties: density, viscosity, thixotropy, duration of drying (solidification), spreading capacity.

Paint and technical properties: degree of grinding, applicability, stringability, impurity content, etc.

Properties of paintwork coatings

Physical and chemical properties: adhesion, hardness, elasticity, robustness at straining and bending, impact strength, wearlessness.

Protective properties: resistance to atmospheric effects, resistance to temperature differential, light fastness, thermal-, frost-, tropical resistance.

Chemical properties: resistance to acids, alkali, aggressive gasses, water, oil, gasolene, suds, emulsions and other chemical reagents.

Paint-and-technical and decorative properties: grindability and polishability, color, appearance, gloss value, etc.

Surface preparation prior to applying paintwork materials

Modern methods of treatment of surfaces to be protected

Quality control and adherence to technology of paintwork coating application

Taking into account external and internal factors

Conclusion

Preliminary analysis of properties brings about a particular choice and positive experience in application.

RPS ENTERPRISE, SMOLENSK NPP. TASKS, PROBLEMS, FINDINGS.

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Smolensk NPP became one of 10 Rosatom enterprises in 2015, where a strategy for establishing a model RPS enterprise had been initiated. The production processes with respect to basic types of products and key business processes at this enterprise have been oriented to creating values, and meet the basic criteria of model RPS flows, provide achievement of business objectives through safety, quality, performance, expenditures and corporate culture.

The task is to turn search for and removal of losses into a primary need of every individual.

The purpose is to improve safety, quality and efficiency, influence the prime cost of the generated electric power and labor productivity, increase well-being of employees, and create opportunities for professional development and career progress by means of up-to-date approaches and principles.

The main point is the incorporation of multi-tier engineering management over the entire production front.

The engineering management recognized as the most effective tool for attaining results is used in the course of the program implementation.

By increasing the safety and quality we increase the load factor and solve the main production task of generating the required amount of electric power. By removing the losses in repair operations, in the reporting system, and at the warehouses, we reduce the expenditures, increase the efficiency of processes, thereby, improving safety.

The manager plays the key role in RPS development, because it is impossible to engage other personnel without involvement the chief executive officer (CEO) of the enterprise and CEO of the division.

One of the RPS principles states: set an example for the colleagues, demonstrate an example of implementing the RPS approach and economic thinking at your own working place, and, thus, engaging the entire personnel.

10 enterprises have been competing for the RPS-leader status. Based on the results of developmental peer reviews, the status has been confirmed by 7 of the 10 enterprises, including Smolensk NPP.

In 2016, Smolensk NPP sets even more ambitious targets. After all, while it is easy to become a leader, it is difficult to hold the leadership. Smolensk NPP shall strengthen the deserved status despite tough competition: it is commonly known that another 13 enterprises have joined the struggle for the number-one title.

The main direction of RPS development for 2016 is the use of RPS projects as the tool for attaining the business purposes by SNPP and development of culture of economic production and system of continuous improvement.

The RPS leader, to be honest, sets ambitious goals in 2016: generate 23.250 bln kW • h of electric power, reduce prime cost of the product by 16%, reduce semi-fixed costs by 20%, reduce volume the storage reserves by 23% as compared to 2013. The tools include implementation of RPS projects, optimization of all lines of activity and mass input of proposals on improvements.

In order to implement all the ideas, it is necessary to have the maximum number of zealous supporters of the Rosatom production system, committed to the idea of further development.

ASPECTS OF APPLICATION OF INFORMATION MODEL IN OPERATION, MAINTENANCE, REPAIR AND INSTALLATION OF NPP EQUIPMENT

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Presently, the information models have been built at the majority of NPPs for the purposes of decommissioning. The structure of data of such models encompasses all the basic technological equipment of the plant. Basically, the characteristics of the facilities important for the purposes of preparation for decommissioning and decommissioning per se: mass-dimensional characteristics, physical and chemical and radionuclide composition are taken into account. The information model also comprises an electronic archive of documentation (design, process, regulatory, etc.) in relation to the plant facilities, and most frequently the 2D interactive diagrams and 3D engineering models.

These information models can be used as the software and data platform for shaping operation support systems. The relatively low expenditures for adding the applied functional and extending the list of characteristics of facilities will help the related NPP divisions use the information model at:

- operation (making equipment walkarounds and examinations with the use of means of mobile identification);

- maintenance and repairs (including scheduled operations in case of maintenance and repairs, logging);
- repairs (keeping record of performed repairs in relation to NPP equipment);
- installation (online analysis of the process of equipment installation/disassembly with the use of a 3D model).

BOROSCOPE INSPECTION OF NPP TURBINES FLOWPATH

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This article presents the results of software and hardware complex (SHC) application for boroscope inspection of flowpath of NPP turbine cylinders.

The periodic inspection and control of the changing condition of the blade system of the last stages of NPP turbines rotors by means of SHC helps identify the following:

- Damage to the shroud linkage of the rotor blades,
- Erosion wear of the rotor blades edges, braces in the exhaust section of the high pressure cylinder,
- Integrity of supra-shroud seals,
- Erosion of interfaces of holders and diaphragms with cylinder,
- Presence of foreign objects as a result of partial damage.

The acceptance test of SHC during preventive maintenance in 2011 at the turbine of Unit 4 of Novovoronezh NPP and during preventive maintenance in 2015 at the turbine of Unit 2 of Kalinin NPP have revealed the need for perform a set of additional works for increasing efficiency and decreasing labor content of boroscope inspections.

A need for and possibility of automation of the process of boroscope inspections of the flowpath section of turbines has been established, and the ways of implementing it have been defined.

The incorporation of the complexes and automation of the boroscope inspection process helps master a new technique of inspecting the technical condition of the blade system of the last stages and start the transition to the system of NPP turbine repairs based on the technical condition of the equipment.

Subsection 1.2
ENGINEERING SUPPORT TO NPP OPERATIONS

Topical area
EQUIPMENT LIFE MANAGEMENT AND NUCLEAR POWER UNITS LIFETIME EXTENSION

ASSESSMENT AND CONTROL OF AGEING OF POWER UNIT ELEMENTS

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The management of ageing of NPP power units components shall be performed at all stages of the lifecycle of a power unit (design, manufacture, installation, operation, extending service life and decommissioning). The purpose of ageing management at the stage of operation (including extended service life) is to ensure the maximum efficiency of the operation of power units. A positive result of power unit service life management is a possibility to extend the power unit operating life, which helps attain additional profits without any additional investments for the replacement of facilities under decommissioning.

The experience in regulatory support of service life extension is available in our country and abroad: in the USA, France, Canada, Germany, Great Britain, Japan, etc.

The disadvantages of methods and techniques to ensure the design service life require additional measures for ensuring the service life at the stage of operation. The use of such activities was fragmented, and until recently, they could not be integrated, there was no common method or scientific support. This circumstance, as well as expiry of the assigned service life of some NPP units has drawn the ever-growing number of experts both in Russia and abroad to the issues of assessing the service life and managing components ageing. Alongside with national programs, IAEA and a number of foreign countries have developed and continue to develop special programs of ageing management for nuclear power plants: the International program IGALL (International Generic Ageing Lessons Learned (IGALL) for Nuclear Power Plants) for accumulated experience in the field of ageing (IAEA); program GALL for assessing ageing of NPP components (USA); etc.

The IAEA safety standards require that the corresponding ageing effects and degradation mechanisms be determined for all structures, systems and components (SSC) essential for safety, so that such structures, systems and components are capable of fulfilling the required safety function over the entire service life, i.e. in the field of designing, manufacturing and construction, commissioning, operation and decommissioning. It is also necessary to take into account the ageing effects and degradation mechanisms under normal conditions of operation, under testing, scheduled repairs, as well as under the condition of component state in case of a postulated initiating event. It is necessary to have regulations on SSC testing, maintenance, control and inspection, in order to assess the ageing effects and degradation mechanisms predicted at the stage of designing, as well as to determine the unforeseen behavior or degradation, which can occur in the process of operation. It is necessary to collect and save the data on operation experience to be used as the source data to control SSC ageing.

The systematic ageing management envisages availability of safety functions over the entire service life and decommissioning of equipment taking into account changes that take place in the course of time and due to its usage. For this purpose it is necessary to solve the issues of SSC physical ageing, which brings about the deterioration of their operational properties and SSC moral ageing, i.e. their obsolescence in terms of the current knowledge, regulations, rules and technologies. In order to efficiently manage ageing within the entire service life of SSC, it is necessary to apply the system-wide approach to ageing management, which provides the basis for coordination of all programs and measures associated with understanding, preventing, identifying, monitoring and excluding SSC ageing effects, including maintenance, operational control, testing and inspection, as well as operation, programs of technical support (including analysis of any ageing effects and degradation mechanisms).

VNIIAES JSC has vast experience in theoretical and practical activities in ageing management accumulated in performing works for extending NPP service life and presented in this article.

The components ageing is managed at all stages of the life cycle of an NPP power unit by determining the technical condition of NPP components through checking the technical condition (diagnostics), and extending the service life of NPP components by replacing (upgrading) or performing timely maintenance and repairs .

The ageing management includes observation and registration of parameters, which characterize changes in the properties of facilities required for fulfilling their functions, and the control of components reliability in operation. The measures that facilitate ageing management, include the following:

- non-destructive and destructive testing;
- technical expert examination;

- monitoring operational parameters;
- evaluation and predicting service life of the NPP power unit components.

The critical NPP components are subject to mandatory ageing management. The critical components include the components that are important for safety, whose integrity and serviceability determine the NPP reliability and operating life as a whole, as well as the components whose failure can bring about by far bigger economical losses than the expenditures for the their management and diagnostics. Besides, they include the components whose replacement is impossible or economically unjustified in operation.

The process of NPP power unit components ageing management includes the following:

- identifying primary mechanisms and effects of ageing of NPP power unit components;
- implementing measures on optimizing the operating conditions to reduce the impact of the effects of NPP power unit components ageing;
- performing investigation, assessing technical condition and justification of the remaining service life of NPP power unit components;
- maintaining reliability of components in accordance with the requirements of the technical documentation to ensure the required safety within the entire service life of an NPP power unit;
- comparing expenditures for component decommissioning and replacement with the expenditures for extending service life, including expenditures for additional works on examination, assessing technical condition and justification of the remaining service life;
- keeping record of NPP components whose service life corresponds to the assigned or extended operating life of NPP power unit, and the components whose service life expires earlier than the assigned or extended operating life of the power unit;
- timely replacement of components of power units that have reached their limiting condition;
- fulfilling requirements of the applicable regulatory and methodological documents specifying the ageing management of NPP power unit components.

EXPERIENCE IN PERFORMING WORKS ON RESTORATION OF REACTOR COMPONENTS OF POWER UNITS AT LENINGRAD NPP

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A characteristic feature of RBMK-1000 reactors at the final stage of operation is the development of degradation processes in reactor structural components:

- graphite stacking deformation and cracking;
- axial shrinkage of graphite columns;
- axial radiation and thermal creepage of zirconium parts of the fuel channel (FC);
- radial radiation and thermal creepage of zirconium parts of FC;
- change of physical and mechanical properties of the stacking graphite (strength, density, thermal conductivity) [1].

The criteria of safe operation of reactor core components of and reactor structures that affect the service life of the reactor and the whole power unit have been defined by the effective regulations, methods and justifications [2].

The main method of managing the service life of the graphite stacking of RBMK-1000 in terms of adherence to the threshold limit values of amount of deflection of FC and FA of the reactor CPS is the use of the developed technique of corrective maintenance. The technique of restoration of lifetime performance (RLP) proven in 2012-2013 at power unit No.1 of Leningrad NPP was successfully used in 2014 at power unit No.2 of Kursk NPP and power unit No. 2 of Leningrad NPP, the second cycle of RLP has been performed at power unit No.1 of Leningrad NPP.

The second important criterion is the criterion of preserving engagement in HARPS assembly.

The processes related to the axial shrinkage of graphite columns and radiation and thermal creepage are quite thoroughly studied at present and are predicted with an acceptable accuracy.

Conclusions

1. In order to ensure the best planning and optimization of repair campaigns of power units at Leningrad NPP, it is necessary to continue improvement of the RLP technique and the tools to perform the works.

2. It is necessary to continue works for improving analytical models of graphite stacking distortion;

3. It is required to continue works on the development and establishing online systems for checking FC geometry in the course of reactor power operation;

4. Carry out investigation of properties of graphite stacking of RBMK-1000 reactor with the aim of determining similar materials and selecting optimal cutting tools and cutting conditions.

IMPROVING REGULATORY SYSTEM FOR THE SERVICE LIFE MANAGEMENT OF NPP COMPONENTCOMPONENTS IN THE CONTEXT OF NP-096-15 ISSUANCE

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Order No. 410 dated October 15, 2015 of the Federal Service for Environmental, Technological and Nuclear Supervision approved the federal rules and regulations in the field of using nuclear power "Requirements for the service life management of equipment and pipelines of nuclear power plants. Fundamental provisions" (NP-096-15).

NP-096-15 covers the service life management of the following NPP equipment and pipelines:

- all equipment and pipelines that are components of Safety Class 1 in NPP unit design;
- all the equipment of single and job-lot production and reference units of pipelines and NPP equipment that are components of Safety Class 2 in the design of NPP unit;
- individual units of equipment and pipelines that are considered components of Safety Class 3 in the design of NPP unit according to the procedure established by the operator as agreed upon with the agency that designs reactors and NPPs.

The introduction of NP-096-15 requires immense preparatory measures that are considered in this article:

- development of the Standard Program of the management of the service life of NPP thermal and mechanical equipment (first edition)
- revision of NP-017-2000, PNAE G-7-002-87;
- development of standard lists of equipment and pipelines for different types of NPPs covered by NP- 096-15;
- development of Safety Guides (SG) on establishing and methods of monitoring the service life of mechano-caloric, electrotechnical equipment and the automatic process control system of NPP;
- development of the national standard "Service Life Management for components of NPP power units" pursuant to the requirements of NP-096-15;
- development of national and industry-specific standards instead of the effective industry-specific regulatory and methodological documents for assessing the technical condition and remaining service life;
- development of the Standard Program of the management of the service life of mechano-caloric, electrotechnical equipment and the automatic process control system of NPP;
- development of programs of the service life management for equipment and pipelines of NPP power units.

NPP power units shall be brought into compliance with the requirements of NP-096-15 gradually.

The article brings special attention to the development of information and analytical systems to control the service life of power unit components at all stages of the lifecycle.

VNIIAES JSC has the experience in developing regulatory documents on extending and managing service life and takes an active part in the above works on improving the regulatory system.

EXTENSION OF SERVICE LIFE OF NPPS WITH VVER-1000

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Rosenergoatom Concern JSC issued an order to commence the works for preparation of power unit 1 of Balakovo NPP for the extension of its service life in accordance with the "Program of extending service life of operating power units of NPPs of Concern Rosenergoatom OJSC for 2013 – 2023" and in view of expiry in 2015 of the design service life of power unit 1 of Balakovo NPP in 2006.

At the 1st stage of the works on extension of the service life of the power unit, an integrated inspection of the power unit was performed, and the safety analysis report was developed. Based on these, "The summary plan of the works to prepare power unit No.1 of Balakovo NPP for service life extension" and the Investment Project "Measures to upgrade Balakovo NPP with the aim of extending the service life of the power unit" were developed on the basis thereof.

At the 2nd stage of the works on extension of the service life, "The program of preparing power unit 1 of Balakovo NPP for service life extension" was developed. According to the Program, 98 measures for replacement and modernization of equipment and the measures for inspection and justification of the remaining service life of the equipment and pipelines, buildings and structures of power unit 1 have been performed from 2010 to 2015. The In-Depth Safety Assessment Report (ISAR) has been developed.

All the measures have been implemented during scheduled repairs of the power unit without any additional shutdowns, thanks to competent prospective planning and explicit control of performance of all works.

Based on the ISAR, measures for replacement and modernization of the equipment, and the Solutions (technical solutions) for justification of the remaining service life of the reactor, the extended service life now equals 30 years and has been justified up to 28.12.2045.

The license for the extended service life of power unit No.1 of Balakovo NPP will be in effect up to December 18, 2045.

RESTORATION OF LIFETIME PERFORMANCE OF THE GRAPHITE STACKING OF NPP POWER UNITS WITH RBMK REACTORS

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1. Brief characteristics of graphite stacking of RBMK-1000
2. Ageing of graphite stacking (GS).
3. Scheduled operating time of NPPs with power units with RBMK reactors.
4. Criteria of safe operation.
5. Purposes and methods ensuring safe operation at the final stage of operation.
6. Basic provisions of the restoration of GS lifetime performance technique (RLP) Basic process operations:

- qualifying operations in the reactor core;
- cutting of graphite blocks;
- Force impact on graphite columns;
- adjustment.

7. Main equipment used in the RLP operations
8. Scope of work performed at NPPs with RBMK
9. Results of the RLP operations at 4 power units, cost performance

SERVICE LIFE EXTENSION AT RUSSIAN NPP POWER UNITS

The service life extension (SLE) of the operating NPP power units is an efficient way of investing financial resources into improving NPP safety and retaining the generating capacity of the Russian power engineering industry.

As of today, in pursuance of the "Long-Term Program of State Atomic Energy Corporation "Rosatom". Activities (2009-2015)" approved by Resolution of the Government of the Russian Federation No. 705 dated September 20, 2008, and in accordance with the requirements of NP-017-2000 "Service life extension", the works have been performed for the service life extension of 24 NPP power units with total installed capacity of 16.242 MW, and new licenses have been received for the extension of their operation. As of April 1, 2016, the power units with the extended service life have generated more than 600 bln. kWh.

The required safety level has been provided, and the probability of reactor core damage has been reduced based on the results of the modernization work package performed at every NPP power unit in accordance with the investment project for the service life extension extending the operating life.

The service life extension of NPP power units in full compliance with the safety requirements is envisaged in the State Program of the Russian Federation "Development of Nuclear Power Industry Complex" approved by Resolution of the Government of the Russian Federation No. 506-12 dated September 02.06.2014.

Currently, the works are underway on implementation of 9 SLE investment projects: power units Nos 2 and4 of Balakovo NPP , power unit No. 2 of Kalinin NPP , power unit No. 3 of Smolensk NPP , power units Nos. 1 and 2 of Kola NPP (repeated SLE), power unit No. 4 of Novovoronezh NPP (repeated SLE).

The investment projects for the service life extension at 3 power units were completed in 2015. The generation of electric power at the installed capacity of 3.000 MW was preserved. The Federal Service of Environmental, Technological and Nuclear Supervision (Rostekhnadzor) issued the first license for the extended operation of power unit No.1 of Balakovo NPP for a period of 30 years. Taking into account the implementation of programs for in-depth overhaul of the equipment and components at power units No. 2 of Smolensk NPP and No. 4 of Kursk NPP, the licenses for the extended operation for 10 and 15 years, respectively, were obtained.

The results of extending service life of NPP power units ensure the safety and social-and-economic stability in Russia due to the minimum tariff burden, maintaining the power balance in regions before commissioning new power units, and maintaining Russian scientific-and-technical capacity.

ON THE ROLE OF CONCEPTS OF STRUCTURAL INTEGRITY OF npp PIPELINES AND EQUIPMENT AT JUSTIFYING SAFETY AND REDUCING OPERATIONAL EXPENSES

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A consideration is given to the regulatory and methodological approaches to providing structural integrity of critical NPP pipelines and equipment based on the concepts of “leak before break” and “zero ruptures” for practical implementation of the requirements of the upper-level regulatory documents (OPB-88/97, NP-082-07) on ensuring NPP safe operation, reliability and quality at all stages of the lifecycle with respect to critical NPP pipelines.

The basic principles of implementing the concepts of structural integrity of critical NPP pipelines and equipment are considered in the paper.

The high quality of designing, manufacturing and installation of NPP component is one of the basic principles of the system approach to ensuring structural integrity. Therefore, the ambiguity of information on the post-installation condition of NPP pipelines shall be eliminated to the maximum extent through instrumentation control of stress-strain behavior during the installation process.

Online monitoring of the stress-strain behavior of pipelines and equipment component prone to service-induced damage helps reveal the reasons for off-design burden of these components due to hidden manufacturing or installation defects, and prioritize the components for future additional technical activities on a differentiated basis, which helps reduce the operational costs.

The safety control based on the concepts of structural integrity of the critical process components makes it possible to stop using additional supports to limit displacements (which are prescribed by NP-082-07) in case of pipeline rupture at an NPP power unit, and serves as the basis for the justification of increasing the inter-inspection intervals of in-service non-destructive testing (NDT). In order to improve the requirements for in-service NDT, the index of effective nuclear operator's regulatory documents contains relevant applicable methodological recommendations approved by the Federal Service of Environmental, Technological and Nuclear Supervision (Rostekhnadzor).

The use of methodology systematic technique for confirming the structural integrity, and of up-to-date methods of instrumentation control of the technical condition of NPP components and their diagnostics during the lifecycle may reduce operational costs (as compared with the operating power units) due to improvement of the requirements for the in-service non-destructive testing and condition-based maintenance of the critical components.

FOR THE CALCULATION OF THE RATE OF EROSION-CORROSION WEAR AND REMAINING SERVICE LIFE OF NPP PIPELINES BASED ON THE IN-SERVICE TESTING DATA

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A great number of data on in-service non-destructive testing of NPP pipelines prone to erosion-corrosion wear (ECW) helps justify approaches to assessing the ECW rate based on the testing data. The research has been performed based on the data of wall thickness measurements of different pipeline components of NPPs with different types of reactors, which helped identify specific features of the ECW process at the straight sections, bends and heat-affected zones of pipelines at NPP with VVER and RBMK reactors. Deposition of corrosion products on the internal surface of a pipeline results in perceived increase of the remaining service life of the equipment components, while the actual condition of the wall under a layer of depositions and the initial wall thickness remain unknown. This research has made it possible to propose a method for the calculation of the ECW rate taking into account the dimensional tolerance and the influence of the deposits on the initial and minimum thickness. In order to calculate the remaining service life, a safety factor has also been introduced, which is taken into consideration for the sake of the conservativeness requirement of the international practice. A unified approach to calculating the ECW rate in the reviewed components has been developed. Besides, the methods of calculating correction factors taking into consideration dimensional tolerances, peculiarities of the component geometry, and the influence of deposits on the measurement results have been proposed based on the testing data and industry-specific standards.

The introduction of correction factors helps improve the conservativeness of the service life calculations as compared with the calculations based on the nominal thicknesses, and the calculation result depends on the standard size of the component, its geometry, and the type of the reactor. A comparative analysis of the results of ECW rate calculations using different formulas has been performed, including the function recommended by the regulatory documentation, and the scope of their use has been specified.

Topical area

OPERATION AND UPGRADING OF ELECTRIC EQUIPMENT AND I&C SYSTEMS

EXPLORATORY DESIGN IS A BASIS FOR HIGH RELIABILITY AND SAFETY OF I&C FUNCTIONING FOR NUCLEAR POWER PLANTS

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In today's world there are a large number of industries, which have their own specifics, possible problems and development prospects. However, the nuclear industry is one of the key strategically important sectors of the national economy, which requires the creation and implementation of new sophisticated information management systems (IMS), I&C, which are subjects to extremely stringent, both Russian and international standards and requirements for safety and efficacy their functioning.

Therefore, to achieve maximum efficiency and functional safety of IMS for nuclear power plants (NPP) exploratory design on the initial stages of the life cycle must be conducted, and, in particular, an analysis of the IMS project feasibility in the conditions of uncertainty.

Exploratory studies are aimed at identifying possible weaknesses in the project of IMS creation; at prevention of its failure; at identifying advanced management decisions to improve the IMS in hard formalized conditions; at implementation of all IMS safety requirements necessary for the IMS design phase and its manufacturing. The need for these studies is due to the fact that the established IMS without them can lead to a situation where the operational phase will face the fact that the behavior of the system will not or does not fully meet the stated requirements for the IMS, which in its turn will lead to the need for urgent adaptation of the implemented IMS.

Despite the large number of research works by many domestic and foreign authors devoted to the creation of IMS for NPP, the problem of exploratory studies is still an open question. This is related to the fact that the research works devoted to problems that arise during the design and manufacture of IMS are fairly easy to review and to cover by virtue of the experience and availability of statistical data. Exploratory studies require innovative approaches to the analysis and forecasting of the situation, which in its turn makes it necessary to have deep knowledge in the subject area, lateral thinking, painstaking research work, which altogether brings enormous economic benefits upon project implementation.

Thus, during implementation of these studies, it is necessary to apply the methods and algorithms for evaluating of the project feasibility regarding creation of IMS, which is aimed at maximum work adaptation with indeterminate primary data, thereby reducing implementation costs, increasing reliability and ensuring the functional safety of IMS for NPP during operation. To carry out exploratory studies, and in particular, to analyze the feasibility of the project regarding creation of IMS for NPP it is suggested to apply cognitive modeling, as well as the momentum modeling of possible development of the situation scenarios during IMS creation, which allow to design system control strategies and make decisions in accordance with the dictate terms of the external and internal environment. The result of these studies is the system of models, which allows the project managers to increase the validity of the adoption of administrative and technical solutions, thereby reducing the risk of human factor.

ON THE POSSIBILITIES OF UTILIZATION OF THE NUCLEAR MAGNETIC SPECTROMETERS IN DIFFERENT I&C OF A NUCLEAR POWER PLANT

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Currently, due to the severe consequences of the accidents at nuclear power plants (NPP) the environmental requirements, both in the territory of NPP and near it, as well as the requirements to the control systems of the various NPP elements and power units were increased. In such a situation, the role of reliable equipment used for the measurements in a variety of control systems increases . To improve the reliability of the various measuring systems studies are being carried out and new instrument and measurement techniques based on different physical phenomenas are being introduced. One of the ways to increase the reliability of NPP operation and to improve the ecological environment monitoring at the NPP and in the surrounding areas is the use of the devices, the work principle of which is based on the phenomenon of the nuclear magnetic resonance (NMR).

Magnetic flow meters and nuclear magnetic spectrometers have been successfully used to control the flow as well as the various liquid media state. Given the fact that in them, as well as in clamp-on ultrasonic flow meters (e.g., company Fluxus) the current contact with the medium is completely eliminated, so they do not make additional alterations to the structure of the current flow, etc. In addition, nuclear magnetic spectrometer can be operated in two modes as a flow meter and as a relaxometer. In the latter mode, it measures the relaxation constant (longitudinal relaxation time T_1 and transverse relaxation time T_2) of the liquid medium. This is the only device that can measure T_1 and T_2 of the the current flow of the liquid medium. On the basis of the measured values, the degree of deviation from the standard medium state or temperature change with an error not exceeding 0.5% can be determined in real time. The only restriction in the use of NMR spectrometers in the signal reception area of NMR is that a section of the pipeline must be made of low magnetic material.

Other application of NMR spectrometers is a rapid control of the state of condensed medium (water, mud, sand, etc.), the substances that contain nuclei with magnetic moments. The deviation of its standard state can be determined according to the measured in real time in a low magnetic field (spectrometer weight does not exceed 4 kg) in a sample section of test medium T_1 and T_2. Simultaneous measurement of two relaxation constants eliminates the error in determining the state of the medium. This allows to deliver to a stationary laboratory for further studies only those environmental samples, which showed a deviation from the stationary state, as well as empowers the area study possibilities, because the number of sampling is always limited by various factors (eg, the number of containers, their marking, etc.).. This allows to conduct a control of the difficult-to-reach areas and the coastal areas of the water bodies.

TECHNICAL DOCUMENTATION REGULATORY REQUIREMENTS FOR CABLE LIFE MANAGEMENT AT NPP

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As part of the extension of the lifetime of NPP power units a series of technical documentation (TD) has been developed in order to estimate the actual state and prediction of the operating life of the various types of power and control cables, which includes taking into account the external influencing factors (EIF) of the design-basis accidents. As of today, these TD is fully updated in recognition of gained experience and the development of new diagnostic techniques. The TD points out the necessary activities for cable product life management (LM), test standards and diagnostic methods of the state during operation. Currently, regulatory and technical requirements for cable LM includes seven TD. Field of application and brief description of each of them are presented in the table below.

Technical documentation for cable LM at NPP

Designation and identification number of the document	Field of application and brief description
PO 1.2.1.02.999.0184-2013 Provision. Determination of the technical condition and aging management of cables at nuclear power plants	Sets a cable life management methodology, and also determines the activities within life management, the requirements and guidelines for certification testing and technical diagnostics of the cables
TPRG 1.2.6.9. 0072-2011 Diagnostics standard program of the cable lines at nuclear power plants. Requirements and guidelines for the development and implementation	Establishes requirements and provides guidelines for the development and implementation of private programs of technical cable diagnostics at NPP power units using non-destructive testing and installation of surveillance specimens
MR 1.2.1.13.1037-2015 Determination of technical condition and remaining operating life of the safety system cables, control and measurements at nuclear power plants.	Designed to determine an actual technical condition and to forecast the operating life of low-voltage cables at NPP on the basis of non-destructive testing methods or practically non-destructive testing (research of micro samples) of the insulation and sheath plastics.
MR 1.2.1.13.0018-2011 Methodological guidelines for the diagnostics of defects in the control cables of nuclear power stations	It is intended to determine the position of the developed defects in the cable lines of the control cables on the basis of different type reflectometry, route and bridge methods

Technical documentation for cable LM at NPP

MR 1.2.02.0168-2013 Diagnostics of technical condition of the power cables with impregnated paper insulation at nuclear power plants	Designed to estimate and predict the state and to determine a position of typical defects on the cable lines of the power cables voltage rate from 0.4 to 10 kV with impregnated paper insulation by measuring the insulation resistance, the reduced voltage, the partial discharge parameters on oscillating die out voltage
MR 1.2.1.13.1005-2015 Complex diagnostics of technical condition of the power cables 6-10 kV with XLPE insulation	Designed to estimate and predict the state of the power cables with XLPE insulation and to locate the installation defects by measuring the isothermal relaxation current and he partial discharge parameters on oscillating die out voltage
MU1.2.2.05.1061-2015 Overvoltage testing and control of the technical condition of the power cable lines at nuclear power plants	Sets scope and rates of testing for power cable lines for voltage of 0.4 to 10 kV in the operation of nuclear power plants with increased AC voltage, including frequency testing of 0.1 Hz and recommends inspection methods for power cables' state during their maintenance.

TD covers almost all tasks performed as part of the work on cable life management. Further improvement of the regulatory and technical requirements for cable life management will depend on the appearance of the cables in operation, which are manufactured from new insulation materials, or on the introduction of new methods of technical diagnostics. As of today, these new cables should include cables, which conform to regulatory requirements regarding low-smoke burning «LS», corrosiveness «HF», flame resistance «FR» . Initial and limit values condition indicators used for operation life evaluation are not defined for these cables. The prospective new methods of diagnostics should include the method of frequency-resonant-domain reflectometry, which in contrast to the traditional time-domain reflectometry is not only able to locate the defect with signs of a limiting condition, but to record yet developing insulation defects.

ROXTEC MODULAR SEALING SYSTEMS FOR CABLE AND PIPE ENTRIES

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This report provides information on modular sealing systems for cable and pipe penetrations by Roxtec company , and also shows the key benefits from the use of modular sealing systems for entries during not only the construction of new nuclear industry facilities, but also during the operation and renovation of existing facilities.

The report points out in detail the structure and elements of cable entry, the issues of the data usage of the entries in areas with high requirements for the sealing, fire - and explosion protection, as well as the issues of the protection of the equipment of nuclear facilities against electromagnetic interference and electromagnetic compatibility of equipment.

Considering the long operation life of the facilities and the arising need to replace the already laid cables and to lay the additional cables, Roxtec modular sealing systems allow to reduce maintenance costs as well as time and labor efforts required for the modernization of facilities.

All screw construction structure of cable and pipe entries allows to perform the installation on the already commissioned facilities. In addition to the above there is no need to dismantle the laid cables.

The report shows that the entries and penetrations in the premises are an integral component of the integrated security system of nuclear facilities.

SMALL BREAK ACOUSTICAL INDICATION SYSTEM OF THE LARGE-MODULAR STEAM GENERATOR "SODIUM-WATER" RU BN-1200

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One of the most prospective methods for monitoring the integrity of "sodium - water" steam generator heat exchange equipment is a passive acoustic method based on the measurement of noise characteristics generated by the interaction of water with sodium in the presence of a leak. JSC "SSC RF – IPPE" has now developed small break acoustical indication system (SBAIC).

SBAIC is intended for rapid detection of water leaks in the large modular "sodium-water" steam generators, for determination of the leak size, location and for the signal generation for the PG EPS in order to prevent failure of the RU BN-1200 steam generator due to:

- minimization of the leak detection time in the tube bundle;
- improvement of the maintainability of the steam generator through the use of SBAIC.

In oder to solve these tasks, as a result of experimental studies, two informative frequency domains during the registration and analysis of acoustic signals were chosen: vibroacoustic (0.1 to 40 kHz), ultrasound (40 kHz to 500 kHz). The use of modern highly sensitive acoustic sensors and signal processing hardware , as well as the introduction of modern high-speed software and hardware methods of monitoring and analysis of acoustic signals allowed confidently determine the leakage in PG design elements in real-time.

The research works were carried out in JSC "SSC RF – IPPE" on the following benches:

- “Leak simulation model “ - acoustic noise attenuation studies were conducted in tube sheet and heat-exchange tube models. The algorithms of determination of the magnitude and location of a leak were developed
- Bench “Octopus” (PG single-tube model) – boiling background noise measurements were carried out
- Bench “PUSHM” - noise of water injection in sodium was recorded
- Bench “PG multitubular model” - the effect of the coolant presence on the characteristics of acoustic noise caused by leak was studied

The obtained results show that when AD is placed in upper and lower tube sheets, SBAIC steadily registers leakage of the size at least 0.5 g / s and allows to determine the coordinates of a leakage using the developed mathematical algorithms with accuracy better than ± 0,2 m. In addition to the above, SBAIC leakage response rate does not exceed 3 seconds.

EXPERIENCE WITH IMPLEMENTATION AND OPERATION OF OPERATIONAL DC SYSTEMS AT NUCLEAR POWER FACILITIES IN RUSSIA.

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SPE “EKRA” LLC, Cheboksary

One of the main directions SPE “EKRA” LLC in the field of low-voltage packages is the development, production and implementation of the operational dc systems (ODCS) for power facilities in Russia. Since 2011, the implementation of ODCS is being produced by SPE “EKRA” LLC in the Russian nuclear power facilities: Kursk NPP, Beloyarsk NPP, Balakovo NPP and Rostov NPP.

Reliability, maintainability, electrical safety and ease of use are taken into consideration during the design and development of ODCS.

SPE “EKRA” LLC develops the following equipment for ODCS:

- surge protectors;
- thyristor load distributors;
- DC boards;
- EKRA-SKI systems for insulation control.

During modernization of DC boards EKRA-SKI systems for insulation control, analog and digital signals monitoring and recording system were implemented in Russian nuclear power facilities . With the help of monitoring web-based interface alarm events and actions of the operating personnel during operation of DC boards can be objectively and accurately analyzed.

EKRA-SKI systems for insulation control developed by SPE “EKRA” LLC is applied in the DC boards, which allowed to automate the process of insulation monitoring and damaged feeders detecting.

The advantage of the EKRA-SKI systems for insulation control is that it allows to work with the traditional DC board insulation resistance circuit and during detection of the damaged connections it does not cause false alarms of the relay protection and emergency automatics.

In addition, for rapid detection of connections with damaged insulation, a portable EKRA-SKI device was designed and manufactured. EKRA-SKI device is successfully used at Kursk NPP, Rostov NPP and Balakovo NPP.

Presence of sensitive differential current sensors in the EKRA-SKI system allows to identify errors in the connections, that previously were not detected during the application of traditional insulation monitoring schemes and during manual search of "ground" short circuits, namely galvanic interconnection of one or two poles of the batteries, the DC board sections or multiple loads.

Now experts of SPE "EKRA" LLC has gained sufficient experience in the production and implementation of ODCS, as well as in the performance of qualified adjustment works for nuclear power facilities in Russia.

APPLICATION FEATURES OF THE MODERN GENERATOR CIRCUIT-BREAKERS FOR PROTECTION OF THE LARGE POWER UNITS OF THE NUCLEAR POWER PLANTS

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M. Palazzo
ABB Switzerland, Zurich

In modern conditions of the power generation market electrical equipment reliability plays an important economic role. In particular, a power unit downtime caused by failure of certain equipment has a strong negative impact on the payback period of the power plant. Generator circuit-breakers used at power plants to protect synchronous generators, are a crucial element of ensuring of high levels of power reliability.

Operation modes of the generator circuit-breaker in the circuit of the synchronous machine are distinguished by complexity compared to conventional high-voltage circuit-breakers. The relatively high value of inductance of the generator windings, as well as the presence of the rotating masses provide the complex character of the electro-mechanical transition process in the presence of all sorts of disturbances in the circuit (short-circuit at the terminals of the generator, a timing error with the system, etc.), as well as the phenomenon of deferred transition of current through zero. This article describes in detail the operation modes that may cause the abovementioned phenomenas. The real examples of transient simulation of large power units in ATP-EMTP software package were analyzed. Various designs of generator circuit-breakers and features of their usage in a particular mode were also described. Statistics of the significant failures of generator circuit-breakers for different technologies of the arc-interruption chambers and actuators (drives) was collected and analyzed. Finally, conclusions were drawn about the parameters of reliability of certain types of generator circuit-breakers based on their performance during severe transition processes of the NPP large power units. The most recent accomplishments in the field of gas-insulated generator circuit-breakers capable to manage with short-circuit clearing in long-term absence of the current transition through zero were shown.

Use of Information Models of the NPP Electric Power Generating Units for the Purpose of Controlling the Condition of the Electrical Plant Equipment and the Integrated Cable System

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NEOLANT JSC, Moscow

The task of managing the service life characteristics of the electrical plant equipment and the integrated cable system, just as the task of managing those of the other types of the equipment of the NPP electric power generating units, at all the stages of the life cycle, with the construction of the industry-wide system of diagnostics and service life characteristics management, from year to year, has been becoming more and more topical and problematic. This can be, primarily, attributed to the key tendencies, which are in existence in the nuclear power industry at the present time:

- additional electric power production is implemented mainly by falling back upon the action plans aimed at increasing the installed capacity and raising the capacity utilization factor of the NPP power generating units;
- the number of power generating units which have exceeded the project service life has been increasing;
- the ongoing competition in the electric power market has been forcing the energy suppliers to look for reserves to be tapped in terms of cost cutting with respect to the operation of NPP power generating units (particularly, their repair costs, rebuilding costs, costs incurred in extending their operation period).

Despite the seemingly growing role of the tasks relating to the service life management, at the present time, there still exist a number of issues in this area. Among others and not limited to the following is the lack of a centralized information system in place to account for and analyze all the aspects of the activities in the area of managing the service life of the electrical plant equipment. Apart from the obviously increasing role of such a system, its creation is required by both this country's normative documentation and the IAEA recommendations.

The complexity and multiple aspects involved in service life management (identification and study of ageing processes; assessment of the technical condition; accounting for an analyzing the history of repairs, upgrades, replacements; interface of various organizations and exchange of information among them, etc.) would require, in dealing with the issues of setting up the industry wide information system to enable diagnostics of the electrical plant equipment and integrated cable system, that various organizations should participate. Apart from the NPP and the operating organization, it would be required that chief design engineers and general project planners of the NPP power generating units be part of the exercise; just as the organization which provides scientific and engineering support for operation of the power generating units as well as other key specialized organizations.

The report, apart from a brief overview of the requirements of the standard on managing service life of the NPP power generating units elements, deals with the problems relating to service life characteristics management which exist at the present time, also sets forth the prerequisites for automating the tasks of controlling the condition of the NPP power generating units equipment, the aims and tasks of the industry-wide information system for diagnostics and service life management as well as organization and technical proposals pertaining to its creation on the basis of the three-dimensional information models of the NPP power generating units which contain the data on the topology of the power generating units, without knowing which, it is impossible to solve the multiple applied problems in practice.

INNOVATIVE SOLUTIONS PRODUCED BY BRESLER ITS LLC FOR ENERGY SECTOR

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Bresler ITS LLC, Cheboksari

Today, Bresler ITS LLC is not only one of the foremost producers of relay protection and automation devices but also, every year, Bresler ITS LLC carries out research and development (R & D) commissioned by Russia's major energy companies. The Company is capable of offering innovative solutions of multiple existing problems using its own developments, whose implementation would ultimately yield a positive economic effect. The state-of-the-art manufacturing center of Bresler ITS focuses on quality and reliability of the products made, going through different phases of testing components and finished products.

All the relay protection and automation devices offered by Bresler can be used to build «digital sub-stations» and «smart networks», thanks to built-in support MEK 61850 and freely programmable work logic. Aside from that, compatibility is ensured with relay overloads and automation devices of other manufacturers when implementing semi-sets of principal overload protection for 110-750 kV lines on different platforms.

The programs for computing the emergency modes of the «TKZ++» electrical network and selecting the tripping parameters of the relay protection and automation equipment «PSC» allow automation of the computations made, improving the performance quality and minimizing the possibility of personnel's errors. Besides, having been implemented as one single set, the programs dramatically cut down on man-hours spent by computation groups on computations and re-computations.

The software suite for paperwork circulation «Relay Protection and Automation Service» is a powerful tool to centralize all the information and to organize access to it. Providing convenient input, storage and output of the information in the form necessary to the user, the «Relay Protection and Automation Service» allows labor costs to be significantly reduced for work with the documentation by relay protection and automation technicians and interfacing functions.

USE OF NON-PROGRAMMABLE LOGIC IN SAFETY CONTROL SYSTEMS

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At the operating power generating units of the nuclear power plants with VVER-1000 reactors, constructed before the 3-rd power generating unit of the Kalinin NPP, in order to build up safety control systems, and normal operation systems, for that matter, a unified package of technical devices (UPTD) consisting, generally, of a set of the following equipment (excluding primary transducers) were used:

- panels of control and measuring instrumentation with normalizing transducers series «SH78», «SH79», «EP4700», «EP4701» and other secondary instruments;
- power supply panels for Sapfir measuring transducers with extraction of root blocks, 22BP-36 power supply units;
- UPTD RT cabinets with current signal distribution blocks similar to the galvanic division block with current outputs;
- UPTD cabinets (basic) with analogue discrete transducer blocks, type ADP, and logic transducer blocks (type BPN, BLP, BUD, etc.);
- KSO cabinet (based on M-64) to transmit information into the top block and station level system.

This combination of the equipment is called the control system based on «rigid» logic. The safety systems implemented on the basis of such architecture can have, as part of them, up to 70 UPTD cabinets in one channel. On the one hand, this seems to be a very cumbersome structure, requiring huge costs to maintain it and to keep it running, on the other hand, such a system is simple for perception of protection systems logic and interlocks. It is true, though, that regular inspection of safety control systems equipment, pursuant to NP-001-15 requirements, calls for a rather complicated system for centralized testing of protection devices to be available. However, even where the program for centralized testing of protection devices is in place, certain circuits remain unchecked (from the command reception block to the actuator control block). Besides, these systems do not have a self-diagnostics function.

Using programmable hardware, the number of cabinets necessary to build one safety channel is reduced to 15 in the case of the conventional approach to constructing a safety control system, i.e. without division in accordance with the principle of control and protection system-initiating safety control system and the executing section of the engineered safety feature actuation system. Given this, in the system, one can implement both self-diagnostics and regular automatic and automated test of operating status without any impact on actuators.

In terms of developing the idea of creating a safety control system based on «rigid» logic along the same lines as engineered features cabinets and for the purpose of reducing the number of equipment while retaining the general building logic, the company has followed the path of development starting from the engineered features cabinets UKTS-AD, continued with KTPS-PN (package of hardware and software with higher reliability), and ended up with KTS NPL (package of non-programmable logic technical devices).

In UKTS-AD, in certain blocks, microcontrollers are used, and such blocks feature the self-diagnostics function (active and passive test of operating condition), the logic part of the block is reserved. This approach was used for discrete blocks only and did not result in reducing the quantity of the equipment.

KTPS-PN also uses microcontrollers in the logic section of the blocks, self-diagnostics capability has been implemented (active and passive test of operating condition), logic parts of the block are reserved, there has appeared a duplicate information network and the functions of analogue to discrete conversion and command driver have been combined (BFZ). This package, in addition, implements the capability of receiving, converting, computing and multiplying analogue signals by the functional blocks KTPS-PN (type APV, NPT), based on microcontrollers. Use of such package has obviated the necessity to use programs for centralized testing of protection devices, which resulted in a considerable reduction in the quantity of the equipment.

KTS NPL represents a solution of new type in terms of building the engineered safety feature actuation system based on non-programmable logic devices, i.e. without microcontrollers, computers or programmable gate arrays, with respect to implementing the functions of collecting and processing signals, driving protection systems commands, diagnostic testing.

The functions of additional diagnostics, collecting signals and transmitting information into the top block level system, automatic regulation functions are implemented in KTS NPL using programmable devices (microcontrollers and processor modules). Notwithstanding this, the programmable devices do not exert any influence on operation of the «rigid» logic due to use of original circuit engineering solutions involving utilization of unidirectional buffers (amplifiers).

Use of the KTS NPL equipment for construction of safety control system enables implementation of all the required protection related, informational functions as well as the functions of block self-diagnostics and diagnostic testing in the automated mode without acting on actuators, while the total volume of cabinets in one safety channel comprises no more than 15 units.

FRASCAN – SOLUTION FOR DIAGNOSTICS OF TRANSFORMERS TECHNICAL CONDITION

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ALFA-ELECTROTECHTORG TRADING HOUSE LLC, Moscow

- Causes of internal short circuits in transformers
- Methods of diagnosing mechanical condition of transformers
- Russian-made instrument for diagnostics of mechanical condition FRAscan
- System for online-monitoring of transformers' mechanical condition FRAscan-Monitor.

Reliability of emergency power supply system equipment. Highlights

1. Inverter Plant JSC is a Russian manufacturer with its own design engineering, process engineering and manufacturing facilities, which enable it to resolve most challenging issues in a short space of time.

2. The lineup of equipment offered by Inverter Plant JSC for nuclear power plants enable any guaranteed power supply need to be met.

3. Rectifiers of Inverter Plant JSC manufacture are an advanced piece of equipment designed both on thyristor and transistor basis with energy recovered into the network and which operate on up to 4000A currents.

4. Uninterrupted power supply systems produced by Inverter Plant JSC with capacity of up to 1 MVA, which operate in a linkage with storage batteries, located in cabinets or racks of in-house manufacture, provide critical power users with uninterrupted power supply of maximum reliability level.

5. Equipment produced by Inverter Plant JSC has been successfully operating at such facilities as:

Beloyarsk NPP, Kursk NPP, Rostov NPP, Kalinin NPP, Leningrad NPP, Smolensk NPP, Kolsky NPP, Novovoronezh NPP, Mayak PO FGUP, GKHK FGUP, and others.

6. The following features make the equipment reliable:

- Use of tried-out and engineering solutions of uniform type
- Duplicating power supplies for back-up
- Analogue control system
- Modular execution
- Visualization of all the processes with amperages, voltages and temperatures shown on TFT display
- Earthquake resistant execution
- Compliance with normative documentation requirements

NEW LAYOUT SOLUTIONS FOR SWITCHYARD OF ELECTRIC POWER STATIONS AND SUB-STATIONS 35 -750 KV BASED ON ZETO CJSC PRODUCTS», VELIKIE LUKI. IMPROVING RELIABILITY AND SAFETY

1. Introductory part (history of the plant, personnel, etc.).
2. Range of products made from 10 to 1150 kV.
3. Program for diversification and technical retooling of the production facility.
4. Quality inspection of the products made. Conducting acceptance tests.
5. Up-to-date layout solutions for 35-750 kV switchyard circuits. Use of compact solutions based on rigid busbars, block and modular solutions.
6. Installation supervision, warranty and post-warranty (after sale) service.
7. Prospects for development of the company and industrialization of new products.

UP-TO-DATE APPROACHES TO ENSURING CYBER SECURITY OF PRODUCTION PROCESS AUTOMATIC CONTROL SYSTEM

Anton S. Shipulin

The company Kaspersky Laboratory is a Russian developer of information security systems for automated production process control systems. The company monitors global threats and offers Russian and foreign industrial companies solutions and services capable of addressing these threats adequately.

Within the framework of the report, we shall be pointing out topical vulnerabilities and information threats to the up-to-date information security systems for automated production process control systems and to the production processes themselves. The tendencies in the development of the process systems carry new risks to their security, unauthorized access by malefactors, capable of tempering with the management of production processes resulting in as much as their shutdown and causing an emergency.

To eliminate vulnerabilities and to neutralize threats, operators of industrial facilities would need to use a holistic consistent approach to ensuring cyber security of the systems.

The report will talk about such elements of the above holistic approach as the following:

- Existing capability of training professionals in security and raising the operators' awareness
- Analyzing level of protection and study of industrial systems security, security of equipment and software
- Deploying specialized engineering solutions for holistic support of production processes cyber security.

Capabilities of the Kaspersky Industrial Cyber Security in respect of ensuring safety of production process using the example of a live model of an industrial facility will be demonstrated.

NEW PATHS IN DEVELOPMENT OF ELEMER. EXPANDING THE COMPANY'S IMPORT SUBSTITUTION POTENTIAL

Ilya I. Esaulovirector

Today, ELEMER Research and Production Enterprise is a major instrument manufacturing company in Russia established by a group of Russian scholars and entrepreneurs in 1992

Line of business:

- development and series production of automation devices;

The products of ELEMER Research and Production Enterprise, to a great extent, are adapted to these products' operating conditions at Russian production facilities. More than half of the types of the currently serially produced automation devices, (and there are 80 of them), have been designed by us specifically as substitution products.

As part of the company, we have established and have been successfully running such divisions as the research and development center and a laboratory for engineering tests. The results produced by ELEMER's 24 years' track record of innovations are more than 80 types of measuring devices designed and serially produced by the company in-house.

It is noteworthy that:

ELEMER NPP's offering for the automation devices market features the products which are capable of reducing dependence on suppliers from abroad.

Among them, there are such novelties as:

- pressure sensors for severe operating conditions;
- modernized electronic pressure gauges;
- autonomous pressure gauges for accurate measurements MTI-100, including the capability of saving readings into an archive;

- contact indicating thermometers TKP-100 and with independent power supply TKP-100BP
- new platinum and copper resistance thermometers have been designed and type approved which need **only one** calibration after manufacture and no mandatory regular calibrations throughout the entire 15-year long service period;
- work is drawing to an end to secure measuring device type approval for the active spark safety modules, type ELEMER-BREEZE;
- launched into serial production are the new indicators ITTS 420Exd/M3-5 for severe operating conditions; nearing completion stage is the development of new measuring transducers IPM 0499Exd/M2-H;
- a whole spectrum of reference measuring devices have been designed and put into serial production including those which do not have any equivalents in foreign countries.

These products feature properties which are in no way inferior to those inherent in the products made by foreign competitors. Besides, these products have an obvious export potential, which we contemplate tapping as early as April 2016 by participating on our own in the Industry Fair in Hannover! These are the actual fruits of our company's **DEVELOPMENT**!

We are always delighted to host guests on our premises in Zelenograd where our firm is located. Please, all of you, be invited to our plant at your convenience in order to visit ELEMER NPP's expositions at various events

Thank you and welcome to ELEMER!

COMPREHENSIVE DIAGNOSTICS AND TESTING SYSTEM OF SYNCHRONIC GENERATOR EXCITATION. EXPERIENCE IN IMPLEMENTATION AND TEST RESULTS

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NPP RTS-ELECTRO LLC

M.G. Popov
ESiAES FGAJU BPO «StbPU»

R.A. Sokolova, V.Y. Stepankevich
TD Alfa Electrotechtorg LLC

The report is devoted to the issue of creating and implementing into practice of operating electric power stations, this country's high efficiency real time technologies.

Under consideration is the use of the digital diagnostic real time complexes for testing, diagnostics and adjustment of 500 MW turbo generator excitation control systems of the main circuit and stand-by electric power supply circuit for own needs of the Leningrad and Kursk NPPs. Test results of the generator excitation systems are given.

High quality and efficient adjustment of the electric equipment when it is commissioned into operation and during the operation itself (at the stage of routine maintenance) are impossible outside the energy equipment complex of the facility. While, at the same time, the range of test modes at the nuclear power plant is extremely limited or is utterly unacceptable in terms of safety conditions. The use of the digital diagnostic real time complexes (DDRT) as developed and manufactured by RTS-ELECTRO NPP LLC allows the task of testing, adjusting the electric power equipment to be dealt with at a fundamentally new level, allows reliability of its operation and safety of operating the power equipment of the electric power plants to be elevated.

The digital diagnostic complexes make it possible to resolve a broad range of tasks, from project planning to testing, as well as testing for various types of electric equipment, including also remote access mode.

Topical area

MATERIAL SCIENCE AND METAL INSPECTIONS

MODERN STATUS AND WAYS TO IMPROVE THE IN-SEVICE NON-DESTRUCTIVE TESTING OF THE MAIN NPP-VVER EQUIPMENT AND PIPING

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Tasks, drawbacks and ways to improve the in-service non-destructive testing (NDT) systems of the NPP equipment and piping (E&P) have been considered.

Reference documentation regulating the E&P in-service NDT at the operating power units, and the aptitude of the in-service NDT results for the E&P evaluation according to the strength criteria have been assessed.

It is demonstrated that the in-service NDT system:

- has kept the principles of defects rating by elements' thicknesses and does not take into account features of their loading being a part of specific RU components;
- sizes of defects acceptable during operation remained close to admissible ones during manufacturing and are extremely conservative;
- the in-service NDT methods do not assure determination of defects' sizes required for strength analysis;
- the in-service NDT equipment and methods do not provide necessary guarantees for the required defects identification and consistent assessment;

- the system of the safety-related defects identification probability analysis is not available;
- a great share of manual testing is preserved at the operating power units.

It is shown that the in-service NDT key problems and ways to improve it are as follows:

- development of defects' norms which are maximum admissible by strength criteria with the required margins assurance and identification of which for the in-service NDT applied means is assured with the required guarantees, and their inclusion into reference documentation;
- introduction of systems providing not only identification but determination of geometric sizes of discontinuities identified;
- extension of the check-list of E&P controlled by automated systems.

METAL DEGRADATION FEATURES OF THE PGV-1000M STEAM GENERATORS' HEAT-EXCHANGING PIPES (SG HEP). APPLICATION OF ADDITIONAL CRITERIA FOR DECISION ON DECOMMISSIONING THE "DEGRADATING" HEAT-EXCHANGING PIPES BASED ON THE EDDY CURRENT TESTING (ECT)

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Comparison exercise of the SG HEP ECT results received in 2008, 2010, 2012, 2014and 2015 has been performed to analyze the HEP metal condition changes in the zone of intensive degradation. Typical HEP with special attention for the areas where degradation took place have been selected.

Since 2014 a flaw detector OMNI 200R with a possibility of using 8 constant channels (four differential and four absolute ones) and forming own "mixed" channels has been used during the ST HEP ECT. Application of a "mixed" absolute channel allowed for excluding the majority of disrupters and gave the possibility to intrinsically present extended bodies on the HEP wall surface. Use of differential channel on the low frequency (25 kHz) allowed for follow the "discontinuity"-type shallow defects formation (up to 20% of the wall thickness) on the HEP external surface at the initial stage of degradation.

With the purpose of the HEP degradation grow rate evaluation the ECT comparison exercise data are presented in the form of an electronic database. Defects parameters with specification of depth and amplitude, defects and deposits available on the HEP in the zone under consideration have been determined.

In the course of the comparison exercise of HEP ECT data subjected to intensive degradation it was established that formation of typical extended bodies was observed prior to considerable defects outbreak in the zone of intensive degradation.

Based on results of comparison exercise performed recommendations on volumes and control zones, application of additional criteria for making decision on the SG HEP killing have been elaborated.

Additional criteria proposed for the SG HEP killing open the door for differential approach to the SG HEP status evaluation.

APPLICATION OF THE USI METHOD BASED ON THE PHASED GRIDS TECHNIQUE AT LENINGRAD NPP

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Assurance of safe and reliable operation of nuclear power plants' components has always been and remains the top priority task. Important feature in the NPP operation is finding a compromise in solving the issue of repairs or authorization for further operation of equipment and piping with deviations from their initial condition. Keeping in mind that the NPP components repair is carried out under conditions of radiation situation further operation of welded joints and main metal of equipment and piping with admissible deviations leads to reduction of a dose load on the personnel, cost-cutting during the repair period and, as a consequence, enables repair planning during the subsequent power units' shutdown.

The method of ultrasonic inspection based on the phased antenna grids technique which is performed both under normal conditions and optionally is currently widely used in the field of non-destructive examination at Leningrad NPP. This report will contain monitoring of the following equipment and pipelines:

- composite and austenitic SS of NB50 – ND30 pipelines;
- SS of NB800 plated pipelines;
- blade tangs of turbine rotor;
- main metal of process channel and SS of carrier welding to TK;
- areas of components subject to the flow accelerated corrosion (method of blind thickness measurement).

Monitoring results of test samples with artificial reflectors are presented, and comparative analysis of data obtained against results of metallographic studies allowing for evaluating the method potential is given.

EXPERIENCE IN EDDY CURRENT TESTING OF THE NPP EQUIPMENT LIGHT-WALL PIPRES

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Considerable number of heat-exchanging equipment is operated at the NPP. Starting from 2009 the VNIIAES JSC employees perform work on examination of light-wall pipes of heat-exchanging equipment. The main examination method is the eddy current testing with the use of:

- walk-through sensors;
- rotating sensors;
- matrix sensors.

To confirm examination results, as well as to increase the light-wall pipes testing capacity the following is applied:

- method of pulse reflectometry;
- TV control with application of endoscope.

Such methodology allows for significantly increasing reliability of testing results at the expense of confirmation or augmentation of results of one method by the others, as well as to increase actual output.

Such approach allows for performing testing of a great number of pipes with the best output-reliability ration. This ideology gave a good account of itself during testing of the heat-exchanging equipment pipes of reactor and turbine shops operated at the NPP.

Inspection of the 2TPN-1 condenser heat-exchanging pipes condition was performed during PPM-2010 at Kalinin NPP power unit No.2. 1,800 pipes were tested for the full length. Pipes are straight and 7 m long. Prior to testing pipes cleaning from deposits and drying was performed.

Three main turbine condensers were inspected during PPM-2012, 2013 and 2014 at Beloyarsk NPP. Methods of the pipes eddy current testing and pulse reflectometry technique were developed for condenser pipes made of cupronickel alloy. 60,000 unique pipes were tested during two years. More than 15,000 pipes were tested by various methods which gave the opportunity to substantiate a resource for additional service life. Following the resource prolongations the pipes testing was performed in 2016.

Heat-exchanging pipes testing of the TQ10W01, TQ20W01, TQ30W01 emergency and scheduled cooling down heat exchangers was performed at Balakovo NPP power units No. 1, 2, 3 and 4 during the PPM-2014 and 2015. For this purpose temporary methods for the TQ heat exchangers heat-exchanging pipes of 08X18H10T austenitic steel were developed.

Killing of pipes with defects was performed based on the testing results following performance of all the said works. Some pipes were extracted to confirm defects and their quantitative parameterization. Dye penetrant inspection is applied for confirmation in express mode. Metallographic studies which confirmed the defects quantitative parameterization obtained by the eddy current testing with accuracy of ± 15% were carried out for quantitative parameterization at VNIIAES JSC.

Considering the positive experience and importance for the NPP due to equipment aging and necessity of operating outside the designed service life it is proposed to develop and introduce methods of eddy current testing of the NPP equipment light-wall pipes.

PERONNEL SUPPORT SOFTWARE SUITES FOR SOLVING THE PROBLEMS RELATED TO EROSION-CORROSION OF THE NUCLEAR POWER PLANTS EQUIPMENT AND PIPELINES

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Nuclear power plants operation is accompanied by local erosion-corrosion damages of pipelines and equipment components which can result in sudden destructions and, as a consequence, to equipment forced outages and power units shutdowns in whole. Pipelines and equipment of the NPP condensate-feed and wet vapor pipelines manufactured of carbon and lightly doped steel.

Measure on development and introduction of the NPP personnel support software suite (PSSS) for monitoring of erosion-corrosion condition, planning of metal in-service inspection and optimization of repair-preventive work of equipment and pipelines components subject to the flow accelerated corrosion are being implemented within the frames of the Rosenergoatom Concern JSC integral program on the problem of erosion-corrosion (hereinafter Program). Implementation of PSSS provides well-timed detection of erosion-corrosion thinnings close to inadmissible ones and allows for excluding sudden destructions of the secondary circuit equipment and pipelines of the NPP power units.

PSSSs are developed and introduced at the NPP-VVER-1000 pilot power units (Kalinin NPP), VVER-440 (Kola NPP) and BN-600 (Beloyarsk NPP, and adapted at Balakovo NPP and other nuclear power plants. The PSSS usage allows for increasing efficiency and reducing redundancy of measures of the main metal and weld-affected zones in-service inspection of pipelines' components of the NPP power units condensate-feed and wet vapor pipelines.

Application of the PSSS at the new NPP power units shall become most efficient as monitoring of parameters and characteristics (affecting location and velocity of erosion-corrosion) from the beginning of their operation provides the opportunity to considerably increase the accuracy of local erosion-corrosion thinnings prediction.

SYSTEM OF CORROSION PROCESSES MONITORING OF PIPELINES AND EQUIPMENT PROCESS CIRCUITS OF NPPS WITH AQUEOUS ENVIRONMENT

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Monitoring of structural materials corrosion velocity is of great importance in nuclear power generation industry not only for prediction of equipment safe operation duration but also for evaluation of radiation situation and values of corrosion products sedimentation on fuel assemblies and in steam generators affecting the modern power units cost efficient and safe operation.

Modern regulatory documents on the NPP pipelines and equipment life management at the stage of operation establish requirements on arrangement of monitoring and prediction of degradation mechanisms' tendencies.

Similar recommendations were put forward based on results of the IAEA OSART mission work at Novovoronezh NPP in 2015.

However, continuous on-line monitoring of pipelines and equipment corrosion condition at operating and designed domestic NPPs is only performed with indirect methods of chemistry quality indicators monitoring. Numerical evaluation of equal-rate and various types of local corrosion growth rate in real time is not analyzed and is not engaged in the process of chemistry quality control. As a rule, values of equipment and pipelines corrosion damage are determined post factum during the PPM periods with the USI methods. With such approach degradation processes control efficiency at the cost of chemistry quality operational control appears to be reduced.

ATOMPROJEKT JSC proposes considerable expansion of chemistry monitoring and diagnostics capabilities at the cost of use of the corrosion processes on-line and continuous monitoring system (CPMS) for optimal equipment lifetime control, decontamination procedures and repair planning at the NPPs of domestic designs.

The main function of the system is determination (diagnostics) of structural materials corrosion condition and aqueous environment corrosion activity. CPMS is built on the basis of electrochemical sensors of corrosion potentials and currents occurring on the surface of equipment and pipelines of the NPP water circuits which allows for determining the mechanism of equipment corrosion process, its dynamics and predicting further development of corrosion as well as of PK mass transfer and radiation situation (for premises of the NPP primary circuit).

The CPNS implementation allows for justified increasing of periods between scheduled equipment inspections, timely detecting quality violation of the Nuclear Facilities and TPPs aqueous environment, diagnosing the reasons of violations and providing support for making decisions on performance of corrective measures

The CPMS allows for on-line monitoring of the pipelines equal-rate corrosion (from 0.1 mm/y) and growth rate of defects of intergranular cracking under pressure (from 0.1 mm/y) of SG tubes and welded joints of NPPs and TPPs pipelines. The CPMS allows for reasonable evaluating of defects growth depth of the pipelines weld joints with accuracy of ~ 15% as compared against the data of periodic ultrasonic testing.

The report considers the methods of evaluation of corrosion condition of structural components within the limits of the whole coolant circulation circuit based on results controlled by the system in one or several positions. Approach to monitoring positions selection and electrochemical methods of equal-rate and local corrosion velocity monitoring applied are discussed.

ON IMPROVEMENT OF THE NPPSs PIPELINES AND EQUIPMENT METAL MONITORING EFFICIENCY

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Works aimed at efficiency improvement of equipment and pipelines metal in-service monitoring, reduction of the number of unscheduled power units shutdowns due to equipment and pipelines metal damage are performed at VNIIAES JSC within the frames of the in-service non-destructive testing of the NPPs equipment and pipelines information-analytical system putting into commercial operation. Regulatory documentation for evaluation of actual level of equipment and pipelines reliability, as well as for prediction of their remaining operation life is developed for cost reduction of the scope of work on justification of possibility of the NPP power units' equipment and pipelines operation beyond the limits of the design life time based on analysis of data om metal damages.

The third version of the regulatory document "Norms of admissible pipeline components carbon steel walls thicknesses under flow accelerated corrosion" RD EO 1.1.2.11.0571-2015 (hereinafter RD) was put in force in 2015 with the purpose of the NPP equipment and pipelines metal monitoring efficiency improvement. A section on calculation of the flow-enforced corrosion velocity is included in the third version of the RD based on analysis of damages of equipment and pipelines components' metal at the NPP with

VVER and RBMK power units. Inclusion of that section allows for performing forecast analyses of the pipelines operation duration. Unavailability of such section in the first and second versions of the document did not allow for using the RD for the forecast analyses performance.

The report contains the data on metal damages of pipelines and equipment components usage of which allowed for developing the section of on corrosion velocity analysis included into the RD; experience of the document using for the forecast analyses performance is analyzed; ways of the RD improvement and implementation of modern software for evaluation of actual NPPs pipelines and equipment reliability level, as well as prediction of the remaining operation life.

ANNEALING PARAMETERS IMPACT ON THE INTERNALS' MATERIALS PROPERTIES AND STRUCTURE RESTORATION

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Internals' materials operation features, first of all reactor partitions, include high damaging doses and high level of temperatures in the partition block due to γ – quants neutrons absorption. Temperatures and damaging doses in reactor partition cross-sections have big gradients, so occurrence of great internal stresses which may appear to be comparable against the yield limit of internals' materials is possible.

In this regard studying of their properties degradation mechanisms and justification of the internals, materials strength characteristics is required to extend the VVER-100 service life up to 60 years and more.

This report contains integrated studies of austenitic steels structure and properties following irradiation in reactors BOR-60 and VVER-1000.

- It is shown that austenitic steels irradiation results in formation of dislocation loops, pores dimensions and density of which depend on the irradiation dose and temperature and to the secondary phases precipitation (G-phase, titanium carbides and α-ferrite). Besides, irradiation impact preconditions the chemical elements content grain boundary redistribution which shows itself in reduction of chromium concentration and increase of nickel concentration at the grain boundaries and in the matrix adjacent to the field boundary.
- Correlation between the change in steels structural-phase condition and the change in mechanical properties under irradiation is established.

- Method of the irradiated austenitic steels mechanical properties restoration by recovering annealing providing return of mechanical properties to the level close to initial condition is proposed.
- Structural determinations of austenitic steels after irradiation up to damaging doses corresponding to 60 years of operation and subsequent recovery annealing under various modes are performed.

METAL INSPECTION CHECKUP DURING REPAIR CAMPAIGN OF 2016

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1. Four power units with VVER-1000 with capacity of 1,000 MW each are operated at Kalinin NPP: power units' No. 1 and 2 are in operation for more than 30 years, power units 3 an4 are operated for 11 and 4 years respectively.

2. Inspection checkup of equipment and pipelines metal during repair campaign-2016 is performed in line with work programs developed in compliance with standard documentation for each power unit. Conventional methods of non-destructive testing are applied for metal checkup: VDI (visual and dimensional inspection), PT (penetrant test), RGI (radiographic inspection), USI (ultrasonic inspection and ultrasonic thickness measurements), etc.; destructive testing methods: MT (mechanical testing), MMGT (metal micrographic tests), IC test (inter-crystalline corrosion test), etc. The ECT (eddy current testing) is performed at each power unit in accordance with the schedule to determine the steam generators heat-exchanging pipes metal condition. Results of the PGV-1000 ECT during the last 3 years and number of killed defective pipes are given.

3. Updating and implementation of new equipment is planned: a handler for the K—1000=60/1500 and K—1000-60/3000 turbines wheel space (development of the VNIIAES JSC, 2015) and a handler for the PGV-1000 headers preparation and drying (development of the NIKIMT-Atomstroj JSC, 2015).

4. The VUMAN RA-Y closed circuit television systems, REVOLVER 80 INVIZ video endoscopes with revolving camera have been acquired for inspection of equipment and pipeline components hard-to-reach places.

5. Rosenergoatom Concern JSC R&D Program for the period of 2012-2018 aimed at the NPP with VVER-1000 reactor vessels service life extension up to 60 years and more.

DEVELOPMENT OF METHODS FOR RADIATION RE-BLISTERING OF THE VVER-440 VESSEL MATERIALS AFTER ANNEALING

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A 45-years period of the VVER-440 first generation power units (power unit No. 4 at Novovoronezh NPP and power units No. 1 and 2 at Kola NPP) service life terminates in 2016-2018. To extend the reactor service life up to 60 years it is necessary to justify first of all the possibility of safe operation of respective reactor vessel for the whole extended period with account for previously performed and newly planned recovery annealing.

No witness samples are envisaged in the first generation VVER-440 reactor vessels. Monitoring of reactor vessel metal condition during operation is performed with the help of samples manufactured of templets periodically cut off from the vessels internal surface of acting reactors subjected to annealing. Templets examination allows for evaluating current condition of reactor vessel metal.

Obtaining of forecast values for radiation blistering for the 60-years lifetime can only be carried out with the use of results of accelerated re-irradiation of templets' materials and account of experimental data of research programs received during accelerated irradiation of reactor vessel components. Application of such approach requires substantiation of robustness of usage of reactor vessel materials accelerated irradiation after annealing results when getting forecast values, as well as re-consideration and revision of the "Methods of determination of the VVER-440 vessel materials radiation blistering during the re-irradiation after annealing" 1.3.2.06.034.0031-2009 in-force.

The work performed resulted in development and justification of "Methods of determination the VVER-440 (V-230 and V-179) vessel materials radiation blistering after recovery annealing during lifetime extension up to 60 years" (further Methods). In the course of Methods development and confirmation of its conservatism analysis of the updated experimental database including both results of templets' metal inspection immediately after cutting-off and results of reactor vessel materials accelerated irradiation after annealing. In the course of Methods development justification of robustness of use of reactor vessel materials accelerated irradiation after annealing was also performed to obtain the forecast values.

EROSION-CORROSION AT THE NPP WITH RBMK-1000 PROBLEM SOLUTION ROADMAP

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Equipment and pipelines (E&P) shutdowns reduction problem solution is an important task of the reliable NPP operation assurance. Analysis of RBMK-1000 power unit's operation experience testifies to increase of this problem importance while they are aging.

In 2014 the Rosenergoatom Concern OJSC Integral program of measures on destructions prevention and the NPPs equipment and pipelines operational erosion-corrosion hardening (No. NPP PRG-62K(04-03)2012) was amended with measures on the flow accelerated corrosion (FAC) problem solution for power units at the NPPs with RBMK-1000 reactors upon proposal of the NIKIET. In particular such measures envisage development and implementation of modern methods, devices and software assets allowing for obtaining reliable initial data the FAC intensity analyses and reliable prediction of E&P residual service life. In the course of the FAC physical-chemical models formation and methods development special attention is paid to determination of local thinnings places, chemistry impact, chemical composition of steels and reduction of the FAC velocities calculation error, as well as considering conditions of O&P loading.

A checkup of calculated assessments compliance with actually measured values, as well as correction of norms of admissible thicknesses including substantiation of the inter-monitoring period for certain types of E&P shall be performed based on the in-service monitoring data processing and analysis.

The planned work shall result in development of recommendation on applicability or necessity of existing monitoring means and methods retrofitting. Then regulations and programs of in-service monitoring shall be corrected with noticing the required monitoring frequency of specific E&P components with account for norms of tolerance on thickness and the rate of their walls thickness reduction due to the FAC.

Implementation of amendments to the program AES PRG -62***K***(04-03)2012 will allow for assuring safety and economical operation of power units at the NPPs with RBMK-1000 at the final stage of their operation.

STEEL FOR THE HIGH POWER RU STEAM GENERATOR WITH A LIQUID SODIUM COOLANT

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Section-module steam generator is used in operating fast sodium cooled reactor facilities. It allows for bringing out of service only the failed section but not the steam generator itself in case of any problems occurrence. Nevertheless the steam generator of such type possesses significant disadvantages which include: a great specific amount of metal per structure, installation complexity, a great number of high-temperature manifold pipelines, valves. All this requires large premises and increase the cost of a power unit.

A vertical casing-type steam generator consisting of two modules was designed for the new serial high-power sodium-cooled reactor facility at the Gidropress OKB. Such steam generator has no disadvantages typical for a section-module one, however it places strict demands on structural material applied as steel shall successfully resist to destruction in various environments including the case of water leakage into sodium. High temperature of operation and assurance of the steam generator service life of no less than 240,000 hours require application of heat-proof materials possessing high corrosion resistance in the steam-water environment, as well as resistance to chloride corrosion cracking which is also due to the module vertical orientation and availability of additional evaporation area. Martensitic-ferritic 12% chromium steel of mark 07x12NMFB developed at the Prometei CNII KM possesses the required property package, as well as satisfactory thermal conductivity level.

Work performed at the Prometei CNII KM provided commercial development of the 07x12NMFB steel in a wide range of metallurgical semi-finished products: plate stock with thicknesses range from 1 to 150 mm, forged stock with thickness of up to 500 mm, seamless cold-finished tubes.

Filler wire of mark Sv-10x12NMFT has been designed for welding (production is set up at the Elektrostal OJSC) coated electrodes of mark EM-99 designed on its basis (production is set up at the ZiO-Podolsk OMZ JSC and Elektrodnyj Zavod CJSC).

Interdisciplinary approach directed at reduction of negative impact of thermal-strain welding cycle on the main metal, formation of optimal structural seam metal conditions and assurance of weld joints high quality allowed for developing a wide range of welding techniques for manufacturing welded metal structures of 07x12NMFB steel. Techniques of automatic submerged arc welding in thicknesses from 30 to 150 mm, shielded metal arc welding in thicknesses from 16 to 40 mm, as well as technique of heat-exchanging tubes welding to tube sheets with the use of automatic argon arc welding with non-consumable electrodes. Steel welding was successfully tested under conditions of commercial production at ZiO-Podolsk MZ JSC.

The 07x12NMFB mark steel and its welded joints including various combinations with steel of marks 12x1MF, 15x1M1F, 10x18N9, 08x16N11MZ, 09G2SA-A have successfully passed benchmark tests.

Work performed jointly with the GNC RF-FEI demonstrated lack of 07x212NMFB mark steel tendency for cracking in the molten alkali in sodium, as well as increased strength in the flare zone during water leakage to sodium as compared against the 10x2M mark steel.

ПData obtained testify that the material designed possesses high level of supporting features required for the steam generator material, and fully meet requirements of designer on the short-term and long-term mechanical properties. Application of new material in combination with accepted structural solutions allowed for 2-5 times reducing specific amount of metal of a steam generator as compared against the steam generator RU BN-800.

Topical area

RW MANAGEMENT, NUCLEAR POWER UNITS DECOMMISSIONING INCLUDING PREPARATION TO DECOMMISSIONING

OPTIONS OF SPENT ION-EXCHANGING RESIN TREATMENT

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During the operation, approximately 30 thou. m^3 of spent ion-exchanging resins (IER) were accumulated at the Russian NPPs, amounting to around 1/3 of the total volume of the liquid radwaste (LRW) accumulated. This makes joint treatment of the IERs and other LRW impossible because of the IER high share. For the spent IER treatment and conditioning, specialized installations are required. Due to the fact that significant amounts of the spent IERs have also been cumulated at other Rosatom SC and Atomflot enterprises, the issue of developing new installations for the IER treatment becomes challenging.

Currently, there exist no industrial installations for the IER treatment in Russia. The international practice of the IER treatment and conditioning includes thermal methods (drying, pyrolysis, incineration, etc), inclusion in matrix materials (inorganic and polymeric), wet oxidation, and decontamination.

In RADON FSUE, the above methods were tested in laboratory and experimentally with fresh and actual spent radioactive IERs. The technological applicability of various methods was assessed against two selected criteria to which dewatering, embedding in polymeric matrix and supercritical water oxidation corresponded. The first two methods allow deriving an on-specification product meeting the criteria of radwaste acceptability for depositing according to NP-093, with the lowest capital and operational expenditures.

The spent IERs conditioning process straight in the container designed for their further storage or depositing was developed. In this case, after the dewatering the degree of the container filling with the IER amounted to 90-95 % vol. and the free moisture content was less than 1 % wt., which is significantly lower than NP-093 requires.

When combining dewatering and embedding in the polymeric binder, the IER content in the compound was 57-62 % wt. The principal properties of the compound obtained are as follows:

- mechanical compression strength – 40-72 MP**a**;
- ^{137}**C**s leaching rate – 10^{-6}-10^{-5} g/sm^2×day.

Preliminary comparative analysis of the tested methods and RADON FSUE technology efficiency was carried out. It was shown that if KMZ containers are used for the disposal of the conditioned IERs belonging to Radwaste Class 4, then the proposed technology has essential advantages compared to other ways of the spent IER conditioning.

ECOMET-S OJSC EXPERIENCE IN THERMAL INSULATING MATERIAL RADWASTE PROCESSING

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One of the types of the radwaste generated during the NPP operation, maintenance and decommissioning is the thermal insulation material waste contaminated with radioactive substances.

The thermal insulation material (TIM) waste consists mainly of mats in glass-cloth envelopes sized 1200 (2000) x 500 x 80 mm. The material volume density is ~ 100 kg/m^3, melting temperature - 1100÷1400 ^{o}C. The TIMs consist mainly of basalt and mineral wool. The TIMs generally belong to the low-level radioactive waste category. The TIM contamination radionuclide content is as follows: Cs-137 – up to 100 kBq /kg, Co-60 – 0,3÷6 kBq /kg, other radionuclides (Cs-134, Mn-54, Nв-94, etc.) – up to 5 % of the total specific activity. Gamma-radiation dose rate is no more than 10 μSv/h.

In ECOMET-S OJSC, a technology of TIM RW processing by way of their melting together under the temperature of 900÷1200 ^{o}C was developed, with obtaining glass-like materials where the radioactive contamination is transferred to a slow-moving form, which creates the conditions for their ecologically safe storage and disposal. This technology allows reducing the TIM RW volume 10-15 times.

Based on this technology, the design documentation was developed for a radioactively contaminated non-metallic solid radwaste processing shop. The shop shall contain sections and systems ensuring the processing of the TIM RW based on thermal insulation mats with obtaining the RW in the form of glass-like material with defined dispersiveness as a final product. The created secondary RW shall be collected to the primary packing – 200-litre metal barrels sent to temporary storage, with further disposal. The shop design capacity shall be up to 3000 m^3/year.

The shop was commissioned in 2012. The TIM RW suppliers are Kalinin NPP, Novovoronezh NPP, Leningrad NPP. The largest amount of the TIM RW was processed in 2013–2015. Total 4744 m^3 of TIM RW with the weight of 465 tons were treated in the period from 2012 to February 2016. The processed radwaste summary activity is $6.96 \cdot 10^{10}$ Bq. As a result of processing, 352.6 m^3 of RW were produced and packed in 200-litre barrels, total 1,763 barrels. The averaged factor of the RW volume decrease amounted to 13.5.

The works performed showed that the technology developed by ECOMET-S OJSC allows for effective processing of thermal insulation materials radwaste produced during nuclear power plants operation, with obtaining the conditioned RW applicable for final disposal. The TIM RW processing technology is patented.

METAL RADIOACTIVE WASTE DECONTAMINATION EFFECTIVENESS ENHANCEMENT WITH THE APPLICATION OF ULTRASOUND AND ELECTROCHEMISTRY

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The scope of works linked with the treatment of the spent nuclear fuel (SNF) and radioactive waste (RW) including the works during the rehabilitation of the USSR nuclear heritage facilities and NPP power units decommissioning, is growing in Russia from year to year. The major part of solid radwaste (SRW) has a surface contamination with radionuclides. The contaminated layer scope is from parts of percent to 1 – 2 percent of the total SRW amount. A significant portion of the SRWs belongs to intermediate-active waste (IRW). Their disposal cost is 2000 – 4000 \$/m^3.

The decontamination process allows returning a significant part of high-quality metal to further use decreasing the RW handling costs considerably. The SRW activity decrease to the low-level radioactive waste (LLRW) subject to the disposal in special shallow ground repositories reduces the disposal cost of 1 m^3 10 times. Besides, the volume of the remaining part of the IRW category SRW is decreased hugely.

In 2007, the scientific and production company Alexandra-Plus LLC (Vologda, Russia) developed the first plant for liquid decontamination of irregular shape metal SRW using the ultra-sound (US). JSC NIKIET and FSUE MosNPO RADON specialists participated in the development process. Experimental works conducted at the plant proved the high effectiveness of the ultrasonic exposure during the liquid decontamination. The ultrasonic processing in decontaminating solutions of a submarine SNF canister fragment resulted in the decontamination factor (under various treatment conditions and modes) of 90 to 500.

In the following years, Alexandra Plus LLC developed and manufactured installations for radioactively contaminated metal decontamination using the US and electrochemical (EC) processes of metal dissolution under the orders of NPPs and other Rosatom enterprises. Currently, all of them are successfully operating. For reprocessing the generated secondary waste, standard technologies of the enterprises are used.

The experience gained allowed, together with SPbGTI(TU), to develop a technology of combined ultrasonic-electrochemical (US-EC) decontamination which enhances the decontamination effectiveness considerably, widens the scope of tasks that can be resolved, decreases the amount of secondary wastes and their generation in the conditioned shape convenient for storage and final disposal.

The presentation contains a concept of a universal industrial set permitting the decontamination of various metal RWs using the developed US-EC technology and settling-sorption technology for the generated liquid radwaste (LRW) treatment. It is easily adaptable to the customers' technical requirements. The set may be supplied in a mobile version.

RADIATION SHIELD INDUCED ACTIVITY IN THE ISSUE OF NUCLEAR INSTALLATION DECOMMISSIONING

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The results of nuclear installations radiation/thermal and biological shielding materials induced activity research were considered. It was stated that the radiation shielding materials and structures would contribute significantly to the radioactive waste amount during the nuclear installation decommissioning.

The absence of an unambiguous approach during the assessment of the radiation waste amount due to the radiation shield activation was stated.

Scientific, technical and economic issues linked with the activated materials and structures of the nuclear installations are considered from two points of view:

1. The selection of low-activated materials for the shield structures at the stage of new generation nuclear installation design with the purpose of minimizing the radioactive waste amounts at the decommissioning stage.

2. Providing a reliable prediction of a summary activity, category and amount of the radwaste of the first generations nuclear installations.

Approaches to the practical implementation of the above mentioned tasks within the general issue of the nuclear installations decommissioning were proposed.

RADWASTE HANDLING AND ACCOUNTING SYSTEM DEVELOPMENT AND UPGRADING AT NOVOVORONEZH NPP. ESTABLISHING A UNIT FOR THE RADWASTE REMOVAL FROM THE RADIATION CONTROL

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Within the framework of preparing to OSART mission at Novovoronezh NPP, a self-check was conducted for the Novovoronezh NPP RW handling system compliance with the IAEA requirements. According to a number of the IAEA documents, one of the indicators of efficient work in the area of radiation protection is the waste segregation in the controlled area (CA) to radioactive and non-radioactive waste as well as the exclusion from the radiation control of the waste with the activity less than the established levels.

The annual financial assessment of the liabilities concerning the RW handling showed the necessity to decrease the RW generation, *inter alia*, for economic reasons. Amendment No.1 to OSPORB-99/2010 specified the notion of removal from accounting and control and gave clear numerical values for this procedure implementation.

Novovoronezh NPP performed works on establishing a unit for the removal from the radiation control. In this process, the methodology of the waste sorting and removal from the radiation control was developed, certified and approved by FMBA. In Power Unit 5 controlled area, the equipment for the waste segregation in its generation places was installed. The segregation is performed by the gamma radiation dose rate and the waste surface contamination. The Novovoronezh NPP staff is trained in the waste segregation in the CA. The programme for waste separation and removal from the radiation control has been developed and introduced.

A new unit for the removal from the radiation control has been established; it includes an X-ray operator's workplace, the equipment for measuring the waste radiation parameters, and equipment for arrangement of data sheets for LLRW. The unit has been put to operation and since October 2015 the waste transfer to the LLRW category is performed on a monthly basis. A complete removal from the radiation control is also carried out.

Within the Novovoronezh NPP RW accounting automated system improvement, a hardware-software complex "RW data sheet arrangement" which allows to automate the RW data sheet issuing process (automation of calculations performed during the data sheet arrangement, automation of the data storage, both the measuring data and accounting data obtained as a result of measuring).

At Novovoronezh NPP, the NZK containers (non-returnable protective containers) storage is performed in light hangar-type facilities. The effects of ionizing radiation shielding and self-absorption (taking account of all the regulatory requirements and without exceeding permissible levels) are used. With the purpose of the containers shuffling inside the operating storage facilities, a computer code was developed for predicting the radiation situation around the hangar-type facilities. The code using the calculation methods determines the predicted gamma-radiation dose rate around the storage facilities depending on the containers installed therein. In addition, the code allows to model storage facilities of any size or the areas of the NZK accumulation and calculates the predicted gamma-radiation dose rate, which gives an opportunity to provide for a correct placement of the containers at any sites under condition of non-exceeding the established permissible levels of the gamma radiation dose rate.

LIQUID RADIOACTIVE WASTE HANDLING AT KALININ NPP

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At Kalinin NPP, four power units with VVER-1000 reactors are under operation:

Unit No.1 was commissioned in 1984;
Unit No.2 was commissioned in 1986;
Unit No.3 was commissioned in 2004;
Unit No.4 was commissioned in 2011.

Purposeful work for the LRW handling performed at Kalinin NPP provided for a significant reduction of the LRW formation to 6-7 tons per one VVER-1000 power unit, which is the world best results level.

At Kalinin NPP, bituminiziation technology is used for processing the evaporate concentrate (evaporator bottoms) in compliance with JSC NIAEP design.

The bitiminization plant developer and manufacturer is OJSC SverdNIIKhimmash.

At Stage 1, the pilot bituminization plant with the bitumen compound (mixture of evaporated radioactive salts and bitumen) discharge to special tanks was used.

At Stage 2, upon Unit No.3 commissioning, an upgraded industrial bituminizing plant with the bitumen compound loading into containers on the packing table was put to service. The filled containers are transported to the solid radwaste storage faciliy using a biologically shielded container passing a data sheet arrangement unit for determining radioactivity parameters and issuing the packing data sheet. After the data sheet arrangement, the containers are placed to the solid radwaste storage facility.

The plant allows for the processing of various wastes:

- evaporator bottoms obtained after the drain water and specialized laundry waste water processing in evaporators;
- ion-exchanging resins pulps.

The bitiminization plant (BP) consists of two parallel trains each having a rated capacity of up to 200 litres per hour by the evaporated water and is designed for liquid radioactive wastes processing by way of their membrane evaporation to the dry residue (with the humidity of less than 1%) and radioactive components inclusion to the melted bitumen up to obtaining a homogeneous mixture – compound with the salt-bitumen relation of 50/50%, which is further loaded into containers.

At Stage 1 bituminization plant, 2,190 m^3 of the evaporator bottoms (about 570 tons of dry salts) were processed. It was put on standby.

Since the beginning of 2014, at Unit 2 bituminization plant the processing of LRW from Stages 1 and 2 tanks is performed.

140 m^3 of the evaporator bottoms (35 tons of dry salts) were processed, 488 containers with the bitumen compound were produced.

Conclusions

1. The LRW formation results reached at Kalinin NPP are 6-7 tons per one VVER-1000 power unit.

2. At Kalinin NPP, the LRW bituminization technology was implemented. The solidified radioactive waste quality meets the criteria of acceptability for disposal. At the BP, the LRW with the compound filling degree of up to 50 % (salts-bitumen) is processed, and the plant allows for processing the wastes with various salt composition.

REACTOR INTERNALS HANDLING DURING NOVOVORONEZH NPP UNITS 1 AND 2 DECOMMISSIONING. TECHNICAL AND ECONOMIC ASPECTS

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Currently, the works under the project of Novovoronezh NPP Units 1 and 2 decommissioning are under way; during the project implementation, the generation of over 18,600 tons of solid radwaste with the total activity of ~ 10^{16} Bq is expected.

One of the main tasks during the decommissioning of Novovoronezh NPP Power Units 1, 2 is the handling of the reactor internals which make the major contribution to the total activity balance of the RW generated during these power units decommissioning.

The design features of the power units and high radioactivity levels of the internals impose stringent requirements and limitations to the selection of technologies of handling these components of Units 1 and 2 reactors.

The international experience of vessel nuclear reactors decommissioning planning and implementation shows an opportunity to use various technical approaches and technologies of the internals handling.

A detail study of various methods of the internals handling with the purpose of the most effective way selection and application will allow reducing the costs of the generated RW handling including the improvement of general technical and economic indicators of Units 1 and 2 decommissioning.

In the presentation, the technical and economic aspects of using three options of the internals handling during Units 1 and 2 decommissioning are analyzed, which provide for:

- the internals withdrawal and fragmentation, with the fragments loading to containers of NZK (non-returnable protective containers) type and their transfer to the National Operator for disposal;
- the internals withdrawal and loading (without fragmentation) to large-size containers and their transfer to the National Operator for disposal;
- straight transfer for the disposal of the internals located in the reactor vessel.

The results of the internals handling options analysis show the advisability of further consideration of the reactor internals disposal without fragmentation (in the reactor vessel or in separate large size containers).

NON-LEARNT LESSONS OF DECOMMISSIONING PROJECTS

A.A. Sobko
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The presentation contents:

1. Short characteristics of the projects
2. Projects preparation
3. Projects planning
4. Performance and control
5. Project completion

The presentation scope: 10-15 transparencies, 15 minutes.

6. Short characteristics of the projects

Research Building “B” liquidation: SB dismantling and decontamination, the site rehabilitation.

Beloyarsk NPP Power Units 1 and 2 decommissioning: TG-1 dismantling, gas holder accident localization system liquidation, removal of RW from dry waste storage facility (HSO-1), removal of spills from reactor stack drainage storage tanks, the metal RW and removed solid RW transfer to a specialized company.

1. Preparation of projects:

- conducting the integrated engineering and radiation survey – IERS (evaluation of the RW initial parameters reliability);
- design specifications and estimates development (design documentation approval and expert study, SAR and EIA development, working documentation development);
- obtaining by the Operating Utility of a license for decommissioning.

2. Planning of the projects:

- project contents and boundaries (to be determined by the project and regulatory guides and work scope list; the lack of clear boundaries and contents brings extremely high risks of gross errors during the work planning);
- planning regular actions on the interaction of the project stakeholders.

3. Performance and control:

- project developer participation at the work performance stage allows for the project contents monitoring (with timely correcting the project errors), submitting the operating data to the Operating Utility and the Customer for the monitoring of the work scope, budget and terms;
- regular interactions on the project provide for timely risk diagnostics and control.

4. Project completion:

- significant effect to the decommissioning projects is made by IERS correctness and the work performer’s observance of the technologies provided for by the project;

- the interaction between the Operating Utility, Project Developer, General Contractor at all steps of the project implementation allows to give the necessary flexibility to the project and timely compensate for the drawbacks of the IERS and design specifications and estimates.

ISPU OPERATION EFFICIENCY ENHANCEMENT

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The application of new generation inorganic sorbents (manufactured by EKSORB Ltd, Ekaterinburg) opens up wide opportunities for the enhancement of ion-selective purification units (ISPU) operation efficiency.

High variability of sorption properties and selectivity to various radionuclides as well as the ability to change the sorbents physical and chemical parameters depending on a specific technological process requirements allow to effectively resolve the issues of purifying the liquid radwaste (LRW) with practically any composition.

A doubtless advantage of inorganic sorbents manufactured by EKSORB Ltd is the possibility to work with the LRW without preliminary filtering the solutions and decomposing organic contaminants, which allows to remove the major part of radionuclides from the solution to the sludge decreasing the number of filtering containers required at the closing stage.

The principal sequence of the ion-selective purification using the EKSORB Ltd sorbents is as follows:

LRW pumping – adding finely-divided sorbent – mixing for 1-2 hours (sorption in a static mode) –separation of sludge with the spent sorbent – sorption in dynamic mode using a filtering container with granulated sorbent.

The ozonization stage is excluded, no LRW heating is required.

At the static sorption stage, 95% of activity are removed from the LRW, the remaining 5% are removed at the dynamic mode or at the static sorption 2nd stage.

It is possible to control the activity distribution between the sludge and the filtering container for minimizing the amount of a corresponding class waste.

The sludge containing the sorbent is separated at the ISPU unit standard equipment; it is cemented easily with the inclusion degree of up to 40%. The produced cement compound corresponds to GOST R 51883-2002.

Therefore, the ISPU operation is simplified greatly, its productivity, safety and reliability are enhanced.

PROCESSING OF EVAPORATE CONCENTRATES GENERATED AFTER IPSU (KOLA NPP)

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At Kola NPP, the ion-selective purification unit (ISPU) for the NPP evaporate condensate purification is under operation since 2006. Similar units are now under design and construction at other plants and in the long term all the NPPs in Russia and other countries will be equipped with them. The decontaminated evaporate concentrates belong to the low-level radwaste (LLRW) category and are not subject to moving away from the plant site. A decision has been made to additionally evaporate these solutions and pour them into barrels – containers that shall be stored in free spaces of the plant after the melts cooling and crystallization. It is clear that this decision is a temporary measure.

The chemical composition of the decontaminated evaporator concentrates contains 95 % of sodium nitrate and sodium tetraborate, i.e., mainly ions of Na^+, NO_3^- and $B_4O_7^{3-}$. Of this mixture, pure alcali NaOH, acid HNO_3 and boric acid H_3BO_3 can be obtained, which are widely used at the NPP. The contents of any of these components in the concentrate manifold exceed their contents in natural raw materials and they are in a soluble form convenient for processing.

For these purposes, the method of electrodialysis with bipolar membranes is proposed. The electrodialisys was intensively implemented at liquid radwaste (LRW) processing organizations (VNIIHT, Radon, VNIINM), then the works were terminated. The ion-selective membranes are manufactured by Russian enterprises (JSC Shchekinoazot), there exist the designs of membrane devices with bipolar membranes.

The principal process is as follows: immediately after the ISPU the brines are delivered to chemical correction where the acidity and concentration optimal for the electrodialysis are reached. At the next stage of the electrodialysis, the components separation is performed: Na^+ and NO_3^- ions are moved through the membranes to adjacent chambers where the acid and the alkali are produced and the borate-ions are left in the initial chamber. After that, the concentration and purification of the products using the membrane and crystallization methods are performed.

Thus, two tasks are solved: the problem of salt solutions or melts storage is resolved and the chemicals reuse at the plant is provided: for ion-exchangers regeneration, for pH adjustment at the ISPU, for decontamination, for making various chemical agents solutions, etc.

The laboratory experiments with model solutions showed a principal possibility of the technology implementation.

BELOYARSK NPP STAGE 1 DECOMMISSIONING: PREPARATORY WORKS

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NPP power units decommissioning is one of the most important short-term issues for the nuclear industry. Power units of Beloyarsk NPP Stage 1 with AMB reactors are at the stage of the preparation for decommissioning.

The purpose of the presentation is to review the planned and executed works for Beloyarsk NPP Stage 1 decommissioning.

The key condition stipulating the permit for beginning the decommissioning works is the spent nuclear fuel (SNF) removal from the NPP power unit. Due to the lack of the AMB reactor SNF cutting and processing technology, the main task after the unit final shutdown is safety provision during the SNF long-term storage in Stage 1 cooling pools. Since 2010, FSUE Mayak has been preparing for AMB SNF receipt; the expected period of the wide-scale fuel delivery is 2020. Within the framework of the fuel long-term storage in Stage 1 cooling pools, the works on safety justification for the SNF storage in dried cooling pools and the SNF assembly life time extension were carried out. In compliance with the newly adopted regulatory documentation, the preparation for the SNF removal from the Beloyarsk NPP is under way, namely: safety assurance during the SNF transportation in TUKs (transportation canisters) and upgrading of the fuel-transportation equipment for the fuel remote reloading from the cooling pools to the TUKs.

The dismantling of individual systems and equipment in line with the preparatory work list of the Decommissioning Project is performed; the liquid and solid radwaste handling and processing units are created.

During the period of the decommissioning preparation, it is also necessary to dismantle the gas holder accident localization equipment and systems, Turbine Units No. 2 and No. 3, decontaminate a number of storage tanks and liquidate Stage 1 dry waste storage.

ABOUT IMPROVEMENT OF CONDITIONING TECHNOLOGIES OF NPP LRW AND RADIOCHEMICAL PRODUCTION BY THE FLUIDIZED-BED PROCESS

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The present report is devoted to handling the LRW including those with salt evaporator sludge, with pulps of ion-exchange resin, as well as with liquid organic waste at NPP because of necessity of their disposal to comply with Federal Law No.190 “Radioactive waste management”.

Main task of initially accepted provisions of handling the LRW at NPP was to exclude non-controllable radionuclide dispersal outside the facility. Safety and reliability was ensured by project solutions. However, due to service life extension, the LRW storage expansion became necessary, for which reason a decision was taken to begin developing technologies and equipment for conditioning of this waste. When applied, it resulted in development of processes of asphalt based solidification, cementation, and evaporation to the maximum salt concentration of LRW in the facilities.

Now a process of their treatment in the course of thermo-chemical transformation (reforming) of waste during fluidization is considered as a promising method for bringing of LRW to standard conditions.

Usage of the industrial fluidization began during the Second World War, in 1942 together with catalytic cracking. Since then it has been used in many fields of engineering. In the course of the process a layer of granules in the plant is converted into fluidized state, when liquid or gas pass through them. Now more than 500 industrial gasoline crackers, coal gasification plants, mineral production plants, pharmacology plants, and waste treatment plants with several units of equipment such as drying and evaporation units, heating/cooling units, units for aggregation of particles and chemical reactions are used in the world.

Usage of fluidization technology for production of nuclear fuel, opening-up and treatment of radioactive waste began in the sixties of previous century. At this particular time the fluidization was used for recovery and hydrofluorination of uranium concentrates [1] and calcination of liquid high-level radioactive waste [2] dispersed with the use of nozzles to fluidized bed with heated spherical particles, evaporation of water and nitric acid from solutions, with extraction of solids. For example, several thousands m^3 of nitric-acid solutions were treated in two calcination plants [3] in Idaho Falls, and then they were converted into dense calcium intended for further treatment and disposal (vitrification).

A laboratory facility was manufactured in the VNIINM early in present century. In the course of validation for technology of inert organic matter thermal conversion in the fluidized bed it was tested, and it constituted a ground to obtain a patent for treatment of ion-exchange resins mixture and pearlite and development of design documentation for production of the plant prototype [4].

In the beginning of this millenium the Swedish company Studsvik and Westinghouse company (USA) concluded venture agreement for development of patent application for process of thermal conversion of medium-level radwaste. In general, the developers have shown that treatment of waste being stored in the DOE containers (Hanford, Idaho, Savannah River) with the use of so-called THOR process in the fluidized bed can contribute to reduction of capital expenditures and costs for treatment of LRW compared with other approaches: vitrification, cementation, and asphalt based solidification. [5]. It is the author's opinion that advantages of these conversions for asphalt emulsion using bituminous component LNPP as an example are listed below:

- The bituminous components are extracted from NPP canyons and converted into emulsion.
- The reforming plant is heated with the use of superheated steam.
- All liquid components of waste (mixture of filler slurry and asphalt emulsion) are converted into steam.
- Effectiveness of conversion of nitrates, nitrites, and nitric acid into free nitrogen with the use of carbon, CO and H_2 is more than 98%.
- Sodium, potassium and aluminium from waste are converted into non-water-soluble mineral product, mainly sodic aluminum silicate, where content of Na_2O can be more than19% (as for its weight).
- Sulfates, chlorides, phosphates and fluorides from waste form a part of Na-Al-Si mineral structure and only less than 5% come to exhaust gases.
- Radionuclides from waste are solidified (99,9%) in the solid mineral product except for 3H, carbon ^{14} and iodine which sublimate at temperature of the process.
- The second stage of conversion (complete combustion) provides effectiveness of organic matters removal at a level of 99,99%.
- In the course of process the emulsion is converted mainly into light hydrocarbons such as methane, carbonic oxide, hydrogen, carbon dioxide and water in the lower part of fluidized bed. In the course of the process an oxygen is inleted into upper part of fluidized bed in order to get stronger oxidation of gaseous components.

Thus the steam conversion of LRW at NPP is related to generation of hydrogen, when hydrocarbons interact with water

$$CH_4 + H_2O \rightarrow CO+3H_2$$

However, if a carbon source is used, it is initially exposed to pyrolysis or evaporation and then reacts with superheated steam as per following reaction

$$C+H_2O\rightarrow CO+H_2$$

Then the hydrogen interacts with oxygen so to avoid excess hydrogen in the system.

In general the reforming plant consists of two units with fluidized bed. The first one is used mostly in the reducing environment, and its function is an evaporation of liquid waste, destruction of organic matters, recovery of nitrates, nitrites, and nitric acid to elemental nitrogen and formation of stable solid products. The unit of the reforming first stage is called as a denitration and mineralization unit. This unit uses the superheated steam as a fluidizing environment.

The layer consists of granules of solid additions and coreactants such as carbon, clay, silicon and/or catalytic agent.

The liquid waste comes to the fluidized bed after a minor preliminary treatment (i.e. concentrating or dilution procedure, addition of clay). Similarly to the above chemical transformation, the carbon is supplied to the fluidized bed to obtain H_2 and CO. For organics of waste which are subject to pyrolysis, formation of different hydrocarbons with getting of reducing environment occurs as per following reactions:

$$C_n H_m + n H_2O \rightarrow nCO + (n+m/2) H_2;$$

Similarly for nitrates in the liquid waste:

$$2Na NO_3 + 3C \rightarrow 2NO + 3CO + Na_2O$$

In the steam environment the sodium oxides react to form alkalies:

$$Na_2O + H_2O \rightarrow 2NaOH$$

Overall reactions:

$$2Na NO_3 + 3C + H_2O \rightarrow 2 NO + 3CO + 2NaOH$$

$$2Na NO_3 + 2C + H_2O \rightarrow 2 NO + 2CO + 2NaOH$$

Nitrogen oxides NO and NO_2 are later recovered to nytrogen in case of interaction with C, CO or H_2, which were formed in the course of reaction of steam and organics.

The nitrates can also be recovered in case of addition of catalytic agent or metal. For example: $2NaNO_3 + 5Fe + H_2O \rightarrow H_2 + 5 FeO + 2 NaOH$;

The second unit of the fluidized bed in the operational process is in the environment of oxidizing medium, and it is called as CCR.

The fluidizing agents of the second unit of the process are exhaust gases of the second stage unit with addition of air (oxygen). Peculiarity of the process in this case is related to gasification of superfine carbon particles flowing from the first unit, oxidation of CO and H_2 to CO_2 and H_2O, and transformation of tracks of acid gases into stable alkali compounds when these acids interact with medium layer consisting of calcium carbonate and/or calcium silicate.

When alumino-silicate containing mineral is brought in the process, different mineral components including anhydrides and sodalites are formed in the fluidized bed. Family of sodalites (including noselite) is a unique one, as they have cellular structure consisting of aluminosilicate triangular pyramids. Bringing of volumetric aluminum silicate in the fluidized bed create conditions for generation of different components such as nephelite comprising SiO_2. The radioactive nuclides in the soda aluminosilicate minerals are arranged in the cubic shape structures, and all these internal cubic shape structures are ion-coupled with oxygen atoms. These minerals are formed of sodium, solutions of waste and additions to be treated (kaolinitic clay), which include SiO_2 and Al_2O_3.

The geopolymers are ceramically similar polymers formed on the basis of aluminum silicates which are basically coupled by alkaline metal ions (M_2O)

in $4SiO_2\ Al_2O_3\ M_2O$. The clays (heated to amorphous state) or amorphous fly ashes are used as a aluminosilicate initial material for their conditioning. Alkaline or alkaline-earth elements in the waste activate the amorphous aluminosilicate structure transforming it into crosslinked inorganic polymers.

A number of recent studies of conditioning of mineral products, transformation of LRW into solid mineral matrix in the course of their treatment in the fluidized bed have shown that filling of final products by waste is 70-80% (for comparison: during cementation with the use of portland cement the filling is maximum 20%).

In general, similar approach to the formed and accumulated LRW can be implemented at any NPP by substituting of asphalt bitumen by other reducing agent (activated carbon, molasses, sucrose, ethylene glycols, etc.). Having regard to the above we consider it necessary to develop draft of appropriate design documentation and produce prototype of equipment (for example, for Experience center НВNPP -1,2 , ЛСК etc.) to carry out the tasks.

List of the branch facilities, where the above technology could be used: for conditioning of LRW

- all existing NPPs and those which are put out of service.
- radwaste storage facilities with LRW.

SOLID RADIOACTIVE WASTE TREATMENT AT KALININ NPP

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Introduction

Solid radioactive waste (SRW) management system is a system of process and administrative procedures for handling the solidified and solid radioactive waste generated in the course of normal operation of NPP, during repairs, as well as in case of emergency situations.

The radioactive waste (SRW) management system is designed to ensure radiation protection for attending personnel and to exclude radioactive contamination of environnement in accordance with requirements of regulatory documents on safety in the nuclear power engineering.

Solid radioactive waste treatment

Facility for treatment of radioactive waste (solid radioactive waste storage facility) is constructed at the Kalinin NPP in order to reduce amount of radioactive waste and for its transformation into state comfortable for storage.

The facility is located in the area of Kalinin NPP site. The solid radioactive waste storage facility includes unit for treatment and unit for storage.

Radwaste compactor.

Characteristics:

- force, kN - 950
- capacity (source material), m^3 /h - 3

- compression ratio - up to 6
- packing to be used: special barrel of 200 l (Ø 560 mm, L=871 mm).

Hydraulic plant of the radwaste compactor includes 3 pumps of following delivery heads:

2,5 MPa
10 MPa
25 MPa.

The radwaste compactor is also equipped with a crimping mechanism for special barrels which is hydraulically driven from pump of delivery head of 10 MPa.

Milling plant.

Characteristics:

- output, m^3 /h - 0,4÷1
- packing to be used: special barrel of 200 l (Ø 560 mm, L=871 mm).

The plant is designed for breaking of large-sized solid waste, and it is able to mill metal products of thickness up to 10 mm.

Incineration plant.

Characteristics:

- rated temperature, °C - 700/1100
- output, kg/h - 50
- rate of volume change - up to 50
- packing to be used: special barrel of 200 l (Ø 560 mm, L=920 mm).

Following low-level solid waste should be treated with the use of the incineration plant:

- working clothes, personal protective equipment (PPE), rags, shoes
- wood material, paperboard, paper
- plastics of all types, plastic compound (if mass fraction of chlorine-containing plastics does not exceed 5%
- sludge of water treatment facilities
- rubber.

The plant is able to incinerate low-level radioactive oil waste.

Incineration of SRW containing explosive matter is not allowed. Content of polyvinylchloride and other matters in the solid waste to be incinerated which incineration results in generation of aggressive and toxic agents in the amount exceeding limits specified in the regulatory documents should be limited to 5%.

Incineration of SRW together with non radioactive waste is not allowed.

Incineration of SRW causes formation of radioactive ash which is directed to cementation unit, where it is mixed with the cement, bentonite clay, and salt solution from gas treatment system of the incineration plant, and grouting occurs after a time delay.

IMPLEMENTATION OF NUCLIDE VECTOR TECHNOLOGY AT NOVOVORONEZH NPP

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As per requirements of federal rules and regulations (RF Government Decree No. 1069 d/d 19.10.2012, NP-093-14 "Acceptance criteria for RW disposal", a detailed radionuclide composition and radionuclide activity in the RW including H-3, alpha-emitting, beta-emitting, and transuranic radionuclides should be specified for classification and arrangement of data sheets (radiation level). Direct measurements of the above radiation characteristics require comprehensive content analysis of radionuclides in each RW container including works on selection and preparation of samples, radiochemical analysis, alpha-, beta-, gamma spectrometry. An alternative approach widely used abroad (ISO 21238-2007) which enables to cardinally reduce costs and improve efficiency of arrangement of RW data sheets consists in revealing of stable and/or conservative ratios between content of radionuclides (nuclide vectors) in different types of RW. The above technology reduces monitoring of radionuclides content in the RW (including even hardly detectable radionuclides which decay is not accompanied by gamma-radiation), to measurement of content of only easily detectable reference gamma-emitting radionuclides (for example, ^{60}Co, ^{137}Cs).

From 2013 to 2015 years the JSC "VNIIAES" carried out a range of experimental and analytical&theoretical works on development and implementation at NVNPP of methodological support for arrangement of RW data sheets on the basis of nuclide vector technology.

In the course of works:

1) Experimental study of radionuclide composition and specific activities of radionuclides in different flows of RW was carried out. About 300 samples of solid and liquid RW were selected and examined in the course of the study. Based on examination of the obtained package of experimental data, the nuclide vectors for different flows of RW were determined, and reference radionuclides were selected.

2) List of radionuclides subject to control in the course of arrangement of data sheets for RW which are generated during operation of NVNPP Units was developed and coordinated with National radioactive waste handling provider.

3) Methodological support for arrangement of RW data sheets on the basis of nuclide vector technology including "Procedure for determination of nuclide vector for arrangement of RW data sheets of Novovoronezh NPP"

and и “Procedure to control radwaste activity at Novovoronezh NPP with the use of nuclide vector technology” was developed and certified .

Implementation of nuclide vector technology at Novovoronezh NPP enables to comply with requirements of PP RF No. 1069 and NP-093-14 to classification and arrangement of RW data sheets as for control of radiation characteristics of RW, to reduce costs, and to improve efficiency of radiation control in the course of arrangement of RW data sheets.

Topical area
SNF MANAGEMENT

ROSENERGOATOM CONCERN OJSC NPP SPENT NUCLEAR FUEL HANDLING PROGRAM 2013-2015 PROGRESS AND 2018 EXPECTED PROGRESS REPORT

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One of the main environmental problems of nuclear energetics is the spent nuclear fuel production. In order to organize and ensure safe spent nuclear fuel handling VNIIAES JSC is developing a SNF handling program for Rosenergoatom Concern OJSC NPP. The latest SNF handling program implementation period came to an end at Rosenergoatom Concern OJSC NPP in 2015.

The increase of index of SNF disposal out of NPP territory towards the volume of spent nuclear fuel produced by the NPP during the period of SNF handling program implementation, the reduction in the number of failures and irregularities in the operation of SNF handling equipment and systems have been specified as target SNF handling program efficiency indicators.

A change in trend of SNF production at NPP was indicated in 2014: for the first time the SNF volume decreased due to the provision of SFA disposal out of Leningrad NPP and Kursk NPP territories. In 2015 this positive trend has become stronger. In just three years of SNF handling program implementation the SNF volume accumulated at the sites of NPP, decreased by almost 380 tons. The number of failures and irregularities in the operation of SNF handling equipment and systems has also decreased from nine in 2013 to five in 2015. Thus, target SNF handling program efficiency indicators were achieved.

The main results of SNF handling program implementation are:

- launching and running of “RBMK-1000 dry-storage SFA control” method under No. MT 1.2.1.02.999.0002-2014;
- reissuing of certificate of approval for TUK-109 container transportation under No. RUS-0103-B(U)F(Rev.3);

- equipping Leningrad NPP power units with facilities for examination of irradiated fuel assemblies before recycling, executing WCD for FA flushing equipment applied before recycling;
- Leningrad NPP took all measures to achieve the designed capacity for dismantling department, ensured off-test SFA disposal to FSUE "PA "Mayak";
- installing measurement units to register the SFA burnup fraction for RBMK-1000 reactors: in pilot operation phase at Leningrad NPP, in integration phase at Kursk NPP, in acquisition phase at Smolensk NPP;
- including assemblies with initial uranium-235 enrichment level of max. 4.92 wt.% and burnup level of max. 58 GW×day/tU to the VVER-1000 list of FAs approved for delivery to fuel recycling plant;
- VNIIAES JSC developed leakage criteria for fuel elements of fuel assemblies of VVER-1000 reactor of B-320, B-338 and B-187 projects, including new types of fuel (without central holes in tablets and with extended fuel stack). These criteria were included to VVER-1000 reactor SSE in order to apply them when reasoning the storage of SFA with failed fuel elements such as those with "gas leakage" in FP out of hermetic canisters;
- disposing 23 failed SFAs from Kalinin NPP unit No.1, 4 failed SFAs from Kola NPP to Novovoronezh NPP, recalibrating the 4 failed SFAs in the shielded box of Novovoronezh NPP for the purpose of shipping them to FSUE "PA "Mayak";
- displacing FP stacks by stacks of compact storage in FP of Balakovo NPP unit No. 1 (in full), in FP of NPP unit No.2 (two compartments); in FP of NPP unit No.3 (one compartment), thus extending FP capacity to 34 %;
- producing 6 TUK-84/1 for AMB SFA, obtaining changed license conditions necessary for Beloyarsk NPP units No.1, No.2 operation in a part of changes of safety case for Beloyarsk NPP AMB SNF storing and handling, executing certificate of approval for AMB SNF transportation in a unified TUK-84/1;
- integrating a unit for the release of technological shafts and FPs of Beloyarsk NPP units No.1, No.2 from the long components ASK 501.00, issuing nuclear and radiation safety case for the works on release of technological shafts and FP-1, 2 under No. 09-06/1148 dated 30.11.2015;
- developing program and implementing engineering surveys for the formation of SNF handling complex during preparation for removal from the site of Bilibino NPP, developing specification for the SNF handling complex project.

At the same the problematic issues were also identified. Due to the lack of funding the measures on development and implementation of technological regulations on repairs of NPP FP with determination of necessary conditions for carrying out works on the control of leaks and relining (elimination of the comments from the targeted inspection check certificate) and to liberation of Novovoronezh NPP separate spent fuel storage facility in the part of development of method of handling with the wasted adjuster rod and burnable absorber rod of VVER-1000 have been not taken.

A new SNF handling program for Rosenergoatom Concern OJSC NPP must be developed under the provisions of the sectoral documentation and new federal of ensuring nuclear and radiation safety, approved by the Russian government in November 2015 and should include steps envisaged by the decisions taken, by the protocols of the Rosatom State Corporation and Rosenergoatom Concern OJSC, and by suggestions of nuclear power plants.

RBMK-1000 DRY-STORAGE SPENT FUEL ASSEMBLY CONTROL

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Criteria of meeting specifications for spent fuel assemblies (SFAs) which are transferred for dry storage in containers, monitoring and analyzing methods together with methods of handling of data on the state of SFA in RBMK-1000 reactor units are specified in MT Guide 1.2.1.02.999.0002-2014 (RBMK-1000 dry-storage spent fuel assembly control.)

On the basis of the Guide the “Instruction on dry-storage spent fuel assembly acceptance control” is issued.

The dry-storage spent fuel assembly acceptance control consists of the following measures:

- selection of SFAs for setting to dry storage;
- checking the selected SFAs for compliance with technical specification “RBMK-1000 Spent Nuclear Fuel. Requirements for transfer to central storage”;
- control of water activity in container with SFA;
- visual monitoring of SFA state for the purpose of determination of SFA damage degree (mechanical damages, the destruction of SFA elements, a change of a clearance between the upper and the lower fuel assemblies);
- determination of SFA factory marking.

According to the results of visual monitoring the following RBMK-1000 SFA types are not approved for further setting to dry storage in 95x2 mm FA capsules:

With mechanical damages of the spacer grid such as rim burr, of over a half of a spacer grid rim’s width (~ 9mm) in order to avoid sticking of FA in capsule.

With destruction of spacer grid 10 (outer spacer grid in the upper FA near the clearance between FAs) in order to avoid sticking of FA in capsule. Main symptoms of spacer grid destruction are spacer grid rim rupture with displacement of the rim edges to the place of rupture or the loss of spacer grid cells. A crack of spacer grid rim without displacement of the rim edges to the place of crack is not a symptom of destruction;

Following damages of spacer grid 11 (outer spacer grid in the upper FA near the clearance between FAs):

- crack of the spacer grid 11, destruction of the spacer grid 11, burr of the rim, of over a half of its width. Spacer grid 11 is the place of attaching SFA by the clamper.

Occurrence of dropout of components from the spacer grid in the clearance between FAs;

SFAs with a clearance between the upper and the lower fuel assemblies that is smaller than 11 mm in order to avoid fuel element' destruction during dismantling process.

By present moment 11,000 SFAs were visually monitored. Over 2,000 of them are not approved for fuel assembly dismantling:

- over 1,500 SFAs have defects of spacer grid 10 - spacer grid 11 such as dents, burrs, cracks, destruction;
- fuel leakage through the clearance between fuel element is revealed in 596 SFAs.

About 100 SFAs were not approved for dismantling for other reasons.

The report shall contain the description of technical means necessary to carry out visual monitoring of the dry-storage SFA state.

ANALYSIS OF THE ADEQUACY/REDUNDANCY OF NORMATIVE-TECHNICAL DOCUMENTATION REQUIREMENTS IN THE PART OF NUCLEAR SAFETY CASE FOR FUEL HANDLING AT NPPS WITH VVER

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The criterion for nuclear safety for fuel handling at NPP is the condition of non-exceedance of the value of effective multiplication factor (Keff), a normative value taking into account the methodological inaccuracy of the calculating program, production tolerances and accuracy of a particular calculation. Keff value which is generally calculated by Monte Carlo method is the leading eigenvalue for solution of asymptotic conditionally critical problem when solving the linearized kinetic equation, and foremost represents a mathematical entity. The physical quantity which is the ratio of the number of neutrons generated in the system per unit of time to the number of neutrons absorbed by and leaving the system in the same unit of time is sometimes used generally and in regulation documents. This value is called dynamic factor of criticality or simply criticality factor. The report shows that when the system is known to be subcritical, the value of Keff is conservative and exceeds the criticality factor that is sufficient for nuclear safety case. Also supercritical situations when Keff is not a of conservative value is considered and the need to further evaluate the criticality factor when considering physical processes in such systems is determined, meaning in general that the problem should be solved with due regard to the time of delayed neutrons and feedbacks.

NUCLEAR SAFETY FOR STORAGE OF NEW FA TYPES WITH ENHANCED URANIUM CAPACITY AND ENRICHMENT OF MAX. 5.0% AND THEIR TRANSPORTATION TO NPP

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Originally the means of transportation and storage at nuclear power plants with VVER-1,000 were designed for the fuel assemblies used in the 90-ies of the last century, with maximum enrichment of 4.4%. Therefore, the use of modern fuels requires an in-depth accounting of nuclear safety cases for storage and transportation throughout the entire NPP fuel cycle.

Occurrence of dangerous nuclear situations during fuel storage and transportation is in principle possible for the entire NPP fuel cycle and can be conditioned by the use of enriched uranium as the fuel and of water as the moderator. The modern stage of nuclear energetics development from this point of view is characterized by factors that potentially increase the risk during handling with both fresh and spent nuclear fuel.

Increasing NPP technical and economic parameters at the modern stage of development is to a great extent provided by the use of new fuel cycles and improvement of the fuel assemblies. At the same time to the present day the VVER-1000 fuel enrichment had reached 5.0% and fuel assembly design has been improved by increasing uranium capacity and reducing amount of construction materials and contaminants.

The Report contains the results of K_{eff} calculations for VVER-1000 fuel handling with the use of modern software and reasonable omission of conservative calculation models.

OPERATIONAL CONTROL ON LEAKPROOFNESS FOR RBMK AND VVER SPENT NUCLEAR FUEL

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The SFA acceptance control is one of the phases of preparation of VVER-1000 spent fuel assemblies' transfer from wet to dry storage In accordance with the technological process of transfer of VVER-1000 spent nuclear fuel from wet to dry storage. The SFA acceptance control is conducted for the purpose of dividing spent fuel assemblies to conforming, suitable for the transfer to dry storage in accordance with the standard technology, and nonconforming, transferable to dry storage in accordance with special technology. Basic requirements for SFA acceptance control are specified "RBMK-1000 dry-storage SFA control" method under No. MT 1.2.1.02.999.0002-2014. One of methods of SFA leakproofness control provided by MT is based on the direct spectrometric measurements of water activity in the capsules.

The report contains the results on development of operational leakproofness control for RBMK-1000 spent fuel assemblies with direct spectrometric measurements, as well as tests of the installation prototype in the at-reactor FP of Smolensk NPP.

In accordance with the technological process of transfer of VVER-1000 spent fuel assemblies from wet to dry storage the phase of leakproofness control must be provided in order to ensure radiation safety when handling with SFAs. In order to ensure leakproofness control a system for measuring leakproofness of the VVER-1000 was designed. The measurement method is based on the continuous measurements of volumetric activity of beta-active gas (^{85}Kr) using gas radiometer.

The report contains the results provided by the system designed for measuring leakproofness of the VVER-1000 fuel assemblies at FSUE Mining and Chemical Combine (GKhK).

ON THE USE OF MKS UNITS FOR MEASURING BURNUP, ISOTOPE COMPOSITION AND RESIDUAL HEAT OF SPENT NUCLEAR FUEL

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In accordance with "Safety rules for storage and transportation of nuclear fuel at nuclear facilities" (NP-061-05), the burnup is considered a nuclear safety parameter if the nuclear fuel control is carries out with the help of special units. The measured values of nuclear fuel burnup can be used for nuclear safety cases and radiation safety cases if SFA is placed in fuel pools, also compressed or in capsules during nuclear fuel storage and transportation.

In order to provide measurements of nuclear fuel burnup of the spent fuel assemblies (SFAs) in an industrial environment the MKS units were developed in Russia. VVER ISS-01 units are integrated at nuclear power plants with VVER-440 reactors (Kola and Novovoronezh NPPs), VVER-1000 reactors (Kalinin NPP). RBMK MKS-01 units are integrated at Leningrad NPP and Kursk NPP. VVER-1000 and RBMK-1000 burnup control units are produced and installed at FSUE Mining and Chemical Combine (GKhK) for the purpose of using them with dry storages. MKS-01 DAV unit is designed and integrated at FSUE "PA "Mayak" to ensure DAV-90 highly enriched fuel control. A complex of equipment based on the MKS-01 AMB unit is designed in order to ensure the isotopic composition control during AMB fuel decomposition at FSUE "PA "Mayak".

All MKS units are certified and registered in the State Register of measuring instruments and approved for exploitation on the territory of the Russian Federation. Measurement procedure (MP) with the use of burnup units is certified and included to the Federal Register of measurement procedures. MP with the use of VVER MKS-01 unit and RBMK MKS-01 unit are considered and approved by the "Commission on the methods and means of control of nuclear safety parameters and alarm systems on the occurrence of a self-sustaining chain reaction in nuclear facilities, storage facilities".

Reports on VVER MKS-01 unit description, measurement procedure and VVER-440, VVER-1000 and RBMK-1000 SFA nuclear fuel burnup measurement results were presented at previous International Scientific and Technical Conferences. This report provides:

- the results of modernization of VVER MKS-01 unit at Kola NPP for the purpose of increase the unit's capacity with measurements 10 days after reactor discharge;
- the results of RBMK-1000 nuclear fuel burnup measurement under conditions of at-reactor fuel pool of Kursk NPP;
- the results of RBMK MKS-03 burnup unit design;
- the results of FSUE "PA "Mayak" DAV-90 units nuclear fuel burnup measurements.

ON THE USE OF UNITS OF MEASUREMENT OF RBMK AND VVER SNF STORAGE SUBCRITICALITY FOR NUCLEAR SAFETY CASES

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It is necessary to comply with the requirements of nuclear safety, as set out in NP-061-05 during handling with SFA in FP ISFSI. A burnup is approved to be used as a parameter of nuclear safety if the control is carried out before placement of FA to FP of the storage with the help of special units. There are several tens of thousands of SFAs kept in ISFSI of NPP with RBMK, thus the measurement procedure is problematic.

Under these conditions a subcriticality measuring unit layout was designed in 1996 as the compensatory measures. It allowed to experimentally (with the support of or in combination with calculation methods) prove theoretical background of the possibility to determine the absolute subcritical value in the area of interest of the fuel pool of the ISFSI. To determine the multiplying properties of spent nuclear fuel a pulse method of direct measurement of subcriticality was applied, provided that Kef.≈0.3 under normal operating conditions. The method is based on measurement of the temporal distribution of the neutron flux density in the environment after it is injected with a pulse of fast neutrons. However, for various reasons (mostly due to lack of funding) the proposed method was not brought to the commercial introduction at NPPs.

A new experimental-industrial UIP-006 unit sample for FP ISFSI subcriticality measurement was created in 2009 at Leningrad NPP where it functioned until 2013. From 2013 till 2015 UIP-006 unit has been upgraded and in 2015 it was tested in terms of FP ISFSI at Leningrad NPP.

A similar SKP-HOT unit was developed in 2015 for the ISFSI of VVER-1000 SFA at FSUE GKhK. SKP-HOT unit passed pilot tests at the HOT-1 ISFS.

Both units are certified and registered in the State Register of measuring instruments and approved for exploitation on the territory of the Russian Federation.

The methods of subcriticality measurement at the ISFSs with the help of units were developed. The methods are certified and included to the Federal Register of measurement procedures.

This Report is dedicated to a brief description of a pulse method of direct measurement of subcriticality and of UIP-006 and SKP-HOT subcriticality measurement units, and also to the results of subcriticality measurement for the specified areas of FP ISFSI at Leningrad NPP and HOT-1 at FSUE Mining and Chemical Combine (GKhK).

USE OF COMPRESSION FOAMING TECHNOLOGY IN MODERN FIRE EXTINGUISHING MEDIA ON NPP

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When extinguishing fires, foam formulations of different consistencies are widely used as a fire extinguishing agent. For preparing such compositions, foam-generators are traditionally used in which the surfactant solution (foaming agent) mixing with the ambient air is converted into a foamy mass.

Recently, a new foam producing method is increasingly implemented – using the so-called compression technology – when the foam is formed in a special mixing chamber by mixing the foaming agent solution with compressed air coming from the compressor. This method provides a number of objective advantages that will improve the effectiveness of fire extinguishing. The following advantages are the most significant in this regard:

1) Increased maneuverability of the fireman due to the fact that not foaming agent solution but already finished foam is provided to the fire hoses, causing the sleeves being very light and supple.
2) It becomes possible to supply the foam to the long distances and higher altitudes while working at low pressures.
3) The range of the foam jet increases several times.
4) The efficiency of fire extinguishing agent greatly increases, as in direct firefighting and in the protection of nearby objects from the effects of thermal radiation.

In 2015, by the order of Rosenergoatom Concern, a research work was carried out, during which these benefits actually received the confirmation and have been studied in terms of quantity. As a result of this work the practical recommendations on the use of compression foam to extinguish fires at NPP have been developed, as well as recommendations on the functional characteristics of foam systems, which produce such foam. These recommendations have been designed and implemented in the form of branch Methodology instructions of Rosenergoatom Concern.

In particular, it was found that when extinguishing different types of combustible substances the use of compression foam ensures 5...6-time saving of fire extinguishing agents (compared with the usual medium expansion foam).

The low level of hydraulic loss in fire hoses is of great practical importance when using compression foam. In comparison with water, the difference on this parameter can be from 4 to 30 times, depending on the section of the hoses used and the expansion rate of compression foam. Due to low hydraulic losses and due to the small weight of the foam, moving on hoses, you can supply it to a height of 300 m, with the outlet system pressure of not more than 1 MPa.

The increased resistance of the foam and its special adhesion properties allow using it as a temporary fire-protective coating of facilities, close to the fire seat. For example, in case of impact of heat radiation on dry wood, the period of its spontaneous combustion can be increased by 25...30% by covering the surface with a protective layer of compression foam.

The detailed regulatory data on specifications of foam compression systems are also of great practical importance, which were obtained as a result of the work done.

In our country, many people know the JSC "KAMPO", which is a developer and manufacturer of respiratory devices for different purposes and the company JSC "Breathing systems-2000", which is not only a distributor of KAMPO, but also designs and manufactures products in its facilities, for example, mobile compressor station on the KAMAZ chassis, which was established a few years ago and is successfully supplied for the needs of the Ministry for Emergency Situations, or our latest development - stationary air compressor BAROS with a capacity of 330 l/min (this capacity is selected based on most of the plants supplied). When developing, we always consider our own experience and wishes of our customers, who know not only the domestic equipment, but foreign counterparts as well. As a result of consultations, we were able, together with our Chinese partner company ECOWELL, to create a compressor that matches the most closely the wishes of the Ministry for Emergency Situations, as along with multi-component electronic charging process control system, we were able to offer more adequate cost of the product. I hope that such compressor will catch the interest of the relevant services of ROSATOM.

Especially for Rosatom companies, KAMPO reworked the AP "Omega-C" breathing apparatus operating manual, is now servicing and inspection of the apparatus become easier.

For example, visual inspection of the apparatus efficiency, without the use of control devices is carried out directly before inserting in the apparatus or once a month. Routine work is carried out once a year, also visual, without using the control facilities, but it is necessary to clean the mask with alcohol, i.e., to disinfect. According to the documentation, pressure check is carried out once a month, at the same time if the cylinder valve is equipped with a pressure indicator; the test simply consists of the indicator inspection. For a more complete control, the indicator can be checked more often.

The apparatus does not provide any routine replacements of any parts during the lifetime, and any user can use the device after reading the operating manual. I want to emphasize once again that in order to reduce time spent on checking the apparatus the cylinder valve must be equipped with an indicator of the air pressure. In the documentation the checking is minimized, but it is impossible to remove it completely, because a person's life depends on the condition of the apparatus, so you should be sure at any time that the apparatus is in working condition.

This year, we have developed and launched certification of a new harness for breathing apparatus with compressed air, which is not an upgraded version, but a completely new structural design using new materials. New rest will allow the height adjustment of the lap belt, which will distribute the weight of the apparatus comfortably for users with different height and build. The mount of the cylinders is versatile, allowing you to set different cylinders, both one and two, without additional purchase of accessories. We decided to keep the air duct system in the same form as for 10 years of operation we have not received any complaints from customers, and the development of this design does not require exceptional knowledge and skills of the staff of operating organizations, as our system does not have adjustments and is dismantled into individual parts by hand. I want especially to note that this innovation does not entail an increase in the cost of production. We expect the first deliveries of the system at the beginning of next year.

For convenience of the checks of breathing apparatus parameters, we created and introduced in the composition of our test facility KU-9V which was checked many times an inflatable test head, which greatly simplified the test of the apparatus for leaks from a full-face mask put on the model. The very same system for checking apparatus parameters is a standalone product, mechanical, without electrical connections and electronic sensors, which could complicate the operation and increase the price of the product (for on-site repair it is not necessary to have special knowledge of electrical engineering and software). I want to draw attention to the fact that the check facility is only necessary for our AP "Omega" breathing apparatuses which are supplied for the needs of the fire service, as the check of the parameters using the control equipment is required in accordance with our legislation. Exactly the same breathing apparatus with the addition of letter "C" in the name does not need any inspection using the control equipment, because all checks are carried out visually, as reflected in the operational documentation. This difference is due only to the presence of the certificate of conformity to the requirements of the technical regulations on the fire safety and its lack in another apparatus. All our apparatuses have no routine replacements, i.e. the replacement of any parts is carried out only when it fails. AP "Omega-C" has the same design and the same technical characteristics as the AP "Omega", but it includes a wide range of cylinders, as it is not limited to the requirements of technical regulations on fire safety on the term of protective action – at least 60 minutes. Therefore, if the apparatus is not intended for operation by fire service of the enterprise, the choice of AP "Omega-C" is more preferable.

For ROSATOM companies it may be interesting to consider the purchase of self-rescuers with compressed air, which are also manufactured by KAMPO, for untrained users, and for trained staff with skills of using breathing apparatuses and responsible for the evacuation of personnel or having the obligation to shut down any production units or other actions under evacuation. The apparatus for an untrained user is in operation immediately when you open the bag where it is stored, and the air from the cylinder flows into the hood completely covering the head and having a transparent front part. The air in this apparatus is supplied continuously, regardless of whether or not the person is breathing. Time of protective action of the apparatus is 20 minutes (using metal composite 2L-cylinder with a working pressure of 300 bar). The apparatus for a trained user is an apparatus with a full-face mask and a lung demand valve, which provides automatic pulmonary air supply, i.e. the apparatus delivers air at a person's breathing, but both apparatuses have excessive air pressure under the hood and a mask that completely eliminates the ingress of harmful substances in the respiratory system.
The time of the protective effect of the apparatus is 27 minutes (using metal composite 3-liter-cylinder with a working pressure of 300 bar).

We can add to the above a few words about the accessories for testing the cylinders included in the breathing apparatuses.

This equipment is a set of products that provide the whole process of re-examination of the cylinders up to 10 l:

- Hydraulic bench for water pressurization in the cylinder up to 450 bar maximum.
- Bath with cylinder rotation mechanism to study for the emergence of a leak.
- Installation for testing the volumetric expansion of metal composite cylinders.
- Installation of drying cylinders.
- Installations for cleaning and rinsing cylinders
- Drilling machine with pneumatic vise for assembly/disassembly of valves or valve removal of residues in case of breaking.
- Table with equipment for pumping cylinders with compressed air blow-off and check them for leaks.

The entire complex is specially designed for the Ministry for Emergency Situations order for small cylinders of breathing apparatuses.

In conclusion, I would like to say that the format of the presentation does not involve a detailed description of the specifications of our products, I inform you about our products and about our capabilities, for more detailed technical information and questions on maintenance, you can always contact our technical support team by sending a request on our website or simply call our office where our staff will explain in detail all the characteristics and differences of the equipment and advise the most suitable for your condition. In addition, we can bring and demonstrate our equipment and answer your questions on the spot.

MEASURES FOR EVALUATION OF FIRE RESISTANCE OF THE STEEL LOAD-BEARING STRUCTURES OF MACHINE HALLS OF OPERATING NPPS, DUE TO THE EXPIRATION OF THE SERVICE LIFE OF FIRE RETARDANT COATINGS

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The innovative solutions for retardant treatment of steel load-bearing structures in order to ensure regulatory requirements for fire resistance are considered. The guidelines for the choice of fire protection for steel structures using structural and methodological scheme are set out in details. The factors that affect the field of application of different methods of fire protection are determined.

Providing the required fire resistance of load-bearing structures of machine halls of operating and newly built nuclear power plants (NPPs) is one of the main components of the NPP fire safety.

Machine halls are located in framed buildings where all vertical and horizontal loads are supported by steel load-bearing structures: columns, girders, beams, trusses.

To achieve the required fire resistance of load-bearing structures involved in ensuring the overall sustainability of the building, the fire protection of these structures plays a decisive role.

Fire retardant intumescent coating was applied as the fire protection of metal load-bearing structures being in service for 15 years or more. Guaranteed service life of such coatings, set by coatings manufacturers, ranges from 10 to 20 years. However, manufacturers can only guarantee the preservation of the appearance of the coating.

When determining the service life of intumescent coatings for steel structures, the accelerated environmental tests are used according to the same procedure as for conventional lacquer coatings on steel plates measuring 150 × 70 mm and 1.0 mm thick. The coating is considered to pass the test on the artificial ageing, if after a certain number of cycles; there are no cracks, destruction, or delamination.

Fire retardant efficiency of intumescent coatings after the accelerated environmental tests is not defined. In accordance with GOST 53295-2009 “Fire retardant compositions for steel constructions. General requirements. Method for determining fire retardant efficiency" fire tests should be conducted on a steel I-section columns, 1,700 mm high and steel plates measuring 600 × 600 × 5 mm coated with fire retardant.

To determine the fire retardant efficiency of the coating, fire protection of steel structures the machine hall should be prepared in the same time as control samples of columns, or coated plates, which should be stored in a room where the fire retardant composition is operated. Then, at regular intervals it is necessary to determine the fire retardant efficiency of the coating and compare those figures with the original ones.

If control samples were not made, it is possible to assess the efficiency of the fire retardant coating after the guaranteed lifetime of intumescent coating by cutting the test samples out of steel structures (connections, secondary beams) and conduct fire tests according to GOST 53295-2009.

When designing and building new NPP units, more reliable coatings should be used, which relate to methods of structural fire protection.

When heated to high temperatures the thermal conductivity of fireproof materials varies depending on their composition and temperature.

Many substances contain a considerable amount of water, whose intensive evaporation slows down the heating of the component being protected. The application of flame retardant composition which includes gypsum and lime leads to getting an evaporation pad and slowing down the heating of the steel. Endothermic decomposition reactions can also take place in fire retardant compositions.

In heavy concrete, cement-sand plaster and sand-lime brick, the thermal conductivity decreases, due to the dehydration of crystalline of cement and silicate stone. In the other fire retardant materials the thermal conductivity increases when temperature increases.

In order to improve the fire resistance of steel structures to the normalized values, following fire protection methods are now applied: heat removal and heat insulation.

The *heat removal* is provided by cooling the hollow steel structures with the circulating fluid and filling the hollow concrete columns.

The *heat insulation* is provided by application of the plaster, wrapping and screening. The heat insulation is carried out in two ways: wet and dry.

The *wet method* includes fire insulation that uses sprayed mineral materials with an inorganic binder, fireproof plasters, intumescent fire retardant paints and coatings.

The *dry method* includes fire insulation in the form of slabs or prefabricated components such as: mineral wool plates, vermiculite plates, slabs and prefabricated parts (shell) made of plaster, etc. Suspended ceilings are also a dry method of fire protection.

Fire-resistant suspended ceilings are used for the protection of the horizontal structures. Suspended ceilings are structural and functional components, and are used mainly for the construction of roofs and ceilings with steel beams, girders, trusses and structures.

The application of different methods of fire protection is determined by taking into account:

- the value of the required fire rating;
- the type of the protected structure and orientation of the surface to be protected in the space (columns, pillars, girders, beams, connection);
- the kind of load acting on the structure;
- the temperature and humidity conditions of exploitation and work on fire protection;
- the degree of aggressiveness of the environment with respect to fire protection and construction material, as well as the degree of aggressiveness of the fire protection material to steel;
- increasing the load on the structure due to the weight of fire protection;
- the start time of the fire protection installation (during the construction of the building or reconstruction);
- the ability to recover after damage;
- the manufacturability (complexity) of application;
- the cost of material and production of works
- the aesthetic requirements to the structures.

THE LATEST TECHNOLOGY OF FIRE AND EXPLOSION PREVENTION AT THE FEC FACILITIES WITH FAST HARDENING FOAMS ON THE BASIS OF STRUCTURED SILICA PARTICLES

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Closed corporation Science production joint of The Contemporary Fire Fighting Technologies

We have achieved great results in fire extinguishing technique and technology of the most destructive Class A fires for people (about 95% of the number of deaths on fires) as a result of the invention of fast hardening foams, the technique of their generation and technology of supply in the fire seat. Depending on the concentration and percentage of the hardening component, the controlled time of hardening for such foams is from 2-5 to 30 seconds. These foams maintain a very high heat capacity and heat vaporization and acquire further 3 or 4 new, quite unusual properties which are especially important when extinguishing Class A fires. 1. Increased spreadability, wettability and surface opacity on almost all kinds of Class A fires. 2. They have great adhesiveness to the surface of almost all types of solid inflammable materials, creating a resistant foam layer on them 3-5 cm thick even on vertical and horizontal ("ceiling") surfaces, when extinguishing fires. 3. These foams have fantastic fire resistant and fire retardant properties, preventing the ignition of solid inflammable materials, even under the influence of a gas burner with a flame temperature of over 1,000°C for 3-5 minutes. 4. When applied on almost any surface of solid inflammable materials as a fire retardant measure these foams retain their volume, shape, and all the fire retardant properties for a few days (up to one month!).

These miracle foams are obtained through the use of sol-gel transition of polydisperse multiphase water-containing systems, which is well-known in colloidal chemistry, to solid state. And incredible thermal and fire resistant properties of foams are obtained by introducing SiO_2 silica to the liquid shells of foam bubbles, with the use of nanotechnology in the formation of polydisperse water-air mixture in the process of generating the hardening foams. The foams we designed for separate supply of the ingredients of mixture composition with the capacity from 2-5 to 30-35 l/s for solution allow extensive use of fast hardening foams for extinguishing Class A fires. From the forest and steppe fires to internal fires in buildings and on special facilities: ammunition warehouses, production and storage of explosives, rubber products warehouses, chemical warehouses and so on.

Briefly described miracle properties of fast hardening foams during certification tests with the participation and under the guidance of specialists of VNIIPO of EMERCOM of Russia in extinguishing a fire of a model pile 1A showed that when extinguishing this pile under all other conditions being equal the flow rate of the fast hardening foam solution is 6-7 times less than the required water flow and 4-5 times less than the number of conventional foam and foam based on the film-forming foam and wetting agents manufactured abroad. And it is the most important, that the time of extinguishing a typical wood pile was also 6-7 times less than extinguishing with water and 3-4 times less than with all other foams. And more importantly, the process of extinguishing with hardening foams lasted only 5-6 seconds, with other foams or wetting agents - 15-20 seconds, and plain water - 35 seconds! But even more striking result of these tests is that re-ignition of the extinguished wood pile with the flame of gas burner with a flame temperature of 1,000 °C was possible for about 15-20 seconds after extinguishing with water for 30-35 seconds after extinguishing with conventional foams or wetting solution and practically not possible to burn even after 3-4 minutes of direct exposure to the flame of a gas burner after extinguishing with fast hardening foam! That is, the wood becomes almost non-flammable after extinguishing with fast hardening foam.

These latest advantages of the new technology of extinguishing Class A fires with fast hardening foams provide incomparable superiority in extinguishing internal Class A fires, particularly in fire fighting with the performance of the combat operation for the rescue of victims in domestic fires and in extinguishing and localization of forest and steppe fires. Fast hardening foams are particularly effective in Hand-held extinguishers as the primary fire extinguishing tools at hand, in stationary automatic and semi-automatic fire extinguishing systems, as a means of fighting fires installed on the ordinary and Special fire engines.

Cardinal improvement of the efficiency of fire extinguishing Class A fires with fast hardening foams is especially important for Russia, at least for two reasons: 1. Russia is a the unattainable First place in the world for the number of human victims on fire. Even according to Official statistics in Russia, each year, fires kill 10-12 times more people per capita in the country than in the USA! (According to unofficial data of the Direction of the EMERCOM of Russia, it is 15-20 times more!) But it was not always like that! In the 1970s, when the population of the USSR and the USA was about the same, about 250 million people, in the USSR the fires killed about 3,500 people a year, and in the USA - about 10,000 people. That is about 3 times MORE than in the USSR. (Draw your own conclusions!) 2. Losses from forest fires in Russia EVERY YEAR are much more than in ANY OTHER COUNTRY! But it is more difficult to prove in the figures, for obvious reasons, than the numerical data on human deaths. So please believe "on the word", because we are engaged in the problem of the efficiency of extinguishing forest fires in Russia and in the world for over 35 years! This allows us to state that such technology and technique of extinguishing Class A fires and especially forest fires are unknown in the world.

RESEARCH OF THE POSSIBILITY OF APPLYING COMPRESSION FOAM TO EXTINGUISH ELECTRICAL EQUIPMENT UNDER VOLTAGE AT NUCLEAR POWER FACILITIES

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To date, Russian nuclear energy is one of the leading sectors of the country's energy. 10 nuclear power stations operate with a total production capacity of 26.3 GW in the territory of the Russian Federation. These facilities must meet all safety standards, including standards of fire safety. The main fire damage at nuclear power facilities is a failure of expensive power engineering equipment, as well as indirect damage to property, which is failure of power supply, power cutoff to ensure people's activity.

According to the existing regulations on liquidation of fires at nuclear power facilities, the priority of fire departments in conjunction with the emergency services of the site is to carry out measures aimed at removing the voltage from the current-carrying parts and units of the power equipment.

However, at the nuclear power facilities the difficulty lies in the fact that there are 30% of the premises in which, for safety reasons, it is forbidden to de-energize electrical equipment, as this may affect the safe operation of a nuclear reactor. Consequently, extinguishing fires of electrical equipment should be carried out under a voltage from 0.4 to 6.0 kV.

The main affecting factor when extinguishing fires of electrical equipment under voltage is a leakage current flowing through the jet of extinguishing agent through a human body. To reduce this risk it is necessary to use the protective means against electric shock, as well as to take into account the safe distances from the nozzle to the electrical equipment. Safe distances are selected according to the passage of leakage current through the jet with the value not more than 0.5 mA, which is not perceptible to humans.

The primary extinguishing agent, used to extinguish fires at nuclear power facilities is water. However, given the fire load of the facility, the water cannot always be effective in extinguishing the fire, and the use of all types of foam agents and other surfactants to increase the extinguishing ability of the jet is prohibited, due to a possible sharp increase of jet conductivity.

The development of fire extinguishing technologies has allowed creating new technology for foam supply. The technology of manufacturing compression foam allows supplying not foam solution but finished mechanical foam through the pump and hosing system of the fire engines to extinguish the fire. When extinguishing the fires, the compression foam has the same properties as simple foam obtained by conventional foaming; however, the compression foam has a number of differences that suggest its high efficiency along with foams obtained by the conventional foaming.

The program to study the supply technology of compression foam is developed to determine whether it is possible to electrical equipment under voltage at nuclear power facilities.

A number of significant parameters will be evaluated, when conducting studies to determine if it is possible to use the compression foam to extinguish electrical equipment under voltage. For the research, the dedicated bench will be upgraded to determine the leakage current on the jet of extinguishing agent. The mathematical analysis of the data will be held according to the experiment results, which will determine the laws on the basis of which it will be possible to create software for pre-planning of forces and means to extinguish fires at nuclear power facilities.

Thus, the studies will allow determining the conditions for the safe application of modern fire extinguishing technologies, such as the supply of compression foam to extinguish fires of electrical equipment under voltage, and thus to increase the efficiency of the fire brigades.

APPLICATION OF ROBOTIC SYSTEMS FOR PROVISION OF FIRE EXTINGUISHING TECHNOLOGIES AT NUCLEAR POWER FACILITIES

M.V. Aleshkov, I.A. Gusev

State Fire Academy of EMERCOM of Russia

Fires at power engineering facilities entail large-scale negative consequences, and therefore the security of these facilities is a strategically important area for the state.

The most flammable facilities in the power engineering complex are areas with oil-bearing equipment, machine halls represent special danger where fuel oils are used in turbine generators and in the cooling system there is flammable and explosive hydrogen.

The development of a fire in the machine halls may be accompanied by the destruction of building structures, the metal trusses of the cover are particularly vulnerable to the adverse effects of fire.

An example is the fire that occurred on the night of December 31, 1978 at the Beloyarsk NPP, which originated in the machine hall. A short time after the beginning of fire, the cover above the machine hall with the area of almost 1,000 m^2 collapsed.

In practice, there are a number of ways to protect metal structures from the effects of high temperatures and open flame burning, but cooling metal structures with water is the most effective in terms of fire.

However, the water supply through the manual and portable fire monitors is linked to the risk of getting the personnel involved in extinguishing the fire in the zone of probable collapse.

The issue of improving fire extinguishing technology at power engineering facilities using mobile robotic systems. This will greatly protect the personnel from exposure to fire hazards, while the effectiveness of fire extinguishing will remain at a high level.

The report analyzes the works connected with fire extinguishing at power engineering facilities where full or partial replacement of the members with the robotic firefighting tools is possible.

We consider the performance characteristics of existing robotic systems used for fire extinguishing and special applications.

APPLICATION OF INNOVATIVE SOLUTIONS FOR FIRE PROTECTION OF NUCLEAR POWER FACILITIES

S.A. Kabeda

Group of companies "Rubezh", Saratov city

Nuclear energy in terms of fire protection deserves special attention. Taking into account the consequences that may occur in the case of fire, the facility should use the equipment that meets the latest achievements in science and technology. One of such systems is an integrated system for safety in large and distributed facilities of any complexity "GLOBAL-NPP".

Through the implementation of a complex of 3 interrelated decisions: the development of their own innovative communication protocol between addressable devices and control unit RSR2, the creation of software adapted for protocol (hereinafter referred to as the software) and appropriate architecture «GLOBAL», the system has received a number of unique features:

- 50,000 addressable devices on one Group controller (hereinafter the Group controller) with an unlimited number of Group controllers per Automated work place;
- 1 km between adjacent addressable devices which, given 250 devices per one address communication line (hereinafter the address communication line), provides the length of 250 km;
- 1 second to survey the status of all devices, and thus the system response to any alarm event. This setting remains the same even in highly distributed security system;
- Automatic addressing and configuration of addressable devices;
- Three hardware-level of backup devices.

ISS «GLOBAL» is an innovation itself, because for the first time it brings together, within the integrated application of the solution, which led to the characteristics described above. Specially designed software algorithms of Group controller allow the use of the same physical devices for different subsystems of the ISS in the ISS "GLOBAL-NPP".

The interaction of the system components is based on the "logging" principle when the System of configuration of addressable devices watch themselves and captures their state and send only the information about the changes to the Group controller. Thus, the efficiency of information delivery is maintained even when the system has a large number of devices.

A comfortable feeding mechanism for the address communication line is implemented which provides in any desired location the connection of addressable fire alarms with a total current consumption more than address communication line is capable of providing.

The system is based on the retransmitting principle implemented in each device. Each address device is able to filter the relevant information among the information received from the device the information, and the rest information destined for subsequent address devices, to send it further and strengthen the repeated signal.

All these abovementioned properties allow you to position the ISS "Global-NPP" on a par with the leading world-class solutions.

MODERN TECHNICAL SOLUTIONS RESPIRATORY PROTECTIVE EQUIPMENT IN CONDITIONS OF EXTENDED TIME WORKS IN UNBREATHABLE ATMOSPHERE

1. Overview of the main design solutions for Dräger AirLine hose systems

Company Dräger is one of the few manufacturers offering a full range of technical solutions for improved battery life in environment unsuitable for breathing. Different approaches

1.1. Recommendations and technical solutions for the use of various sources of compressed air

- regular air lines with operating pressures up to 10 atmospheres;
- mobile balloon assembly of compressed air with working pressure of 300 atmospheres;
- compressor equipment of various types;
- organization and servicing of stationary storage of compressed air

1.2. Hose breathing apparatus, face pieces, lung machines:

A feature of the application of hose breathing apparatus is their use as a spare source of air supply needed for evacuation in the event of termination of the air supply in the main system.

- PAS Micro, PAS Colt and PAS Lite Breathing apparatus
- Panoramic masks and lung machines, cylinders and valves

1.3. Equipment for the organization of pneumatic distribution when dealing with hose systems

The switching equipment, pneumatic lines, splitters. The basic methodology for determining the possibility of using the hose systems

2. Insulating self-rescuers

Insulating self-rescuers. Easy to maintain and easy to use.

Dräger Saver compressed air self-rescuers are designed for different application scenarios:

- The use of Draeger Saver self-rescuers as the RPE during the evacuation. Increased battery life when filling self-rescuers from stationary storage of compressed air
- The use of self-rescuers of the Dräger Saver range Type A, as a hose breathing apparatus as part of the hose systems.

3. The use of compressed-air breathing apparatus (CABA) of Dräger PSS and Dräger Plus series as spare breathing apparatus.

All Dräger breathing apparatus may be used as spare breathing apparatus as part of the hose systems. Features of the application. Economic aspects of the technical solution.

Overview of design decisions on the organization of isolated self-contained rooms with air recirculation.

- Applications of self-contained sites – protective shelters

- Dräger equipment to ensure battery life for up to 36 hours
- Air quality control tools within the site and outside the containment
- Case studies

MOBILE ROBOTIC INSTALLATION FOR FIRE EXTINGUISHING. EXPERIENCE AND SPECIFIC APPLICATIONS TO USE AT NUCLEAR POWER FACILITIES

N.P. Kovalev
LLC "Innovative Technologies of Salvation"

While developing, the energy industry of the country determines the range of the primary tasks, which its stable development and operation will depend on. One of these tasks is to ensure fire safety.

The extinguishing success depends largely on the applicable fire and rescue equipment. But even modern fire-fighting equipment will be useless if there is no one to operate it.

When extinguishing fires and conducting rescue, the personnel is located in the area of thermal effects, in the environment which is unsuitable for breathing, exposed to the impact of fire hazards. It is essential that at nuclear power facilities the participants of extinguishing fire may be exposed to lethal levels of radiation.

In this connection the question arises about the extinguishing efficiency – how to achieve its desired level, while ensuring the safety of personnel.

The answer to this question is the ability to use mobile robotic systems, which are widely used at present.

The need for mobile robotics was shown in the accident at the Chernobyl NPP, which has served as a powerful impetus for the development of civil robotics in the USSR. Robot STR-1, carrying out works on cleaning the roof of the fourth power unit, was one of the first samples of robotics in that period. STR-1 was not the only robot-liquidator of consequences of emergencies, there was a series of others, but because of the strong radiation they broke down one after another.

The main thing is that they carried out a part of their work, replacing a person, thereby saving his life.

Most of the robots that have been used at the Chernobyl NPP had narrowly focused skills. Therefore, the developers of fire robots in many ways try to equip the robot with the systems capable of producing both extinguishing and monitoring the environment, detect the seat of the fire, light the fire place, and others.

The representative of this type of robotic tools is mobile robotic installation for fire extinguishing - MUPR-C-SP-E-IK-TV-UP-20(15,10) model 001 manufactured by LLC "Innovative Technologies of Salvation".

Mobile robotic installation for fire extinguishing is designed for reconnaissance and fire extinguishing in the areas of “emergency situations”, in closed-plan rooms, at sites where human presence is dangerous and others.

When using with a pressure hose line from the fire trucks, pumping station, a hydrant or a powder vehicle for the following purposes:

- the movement in the target areas along the route, for remotely provided by the operator’s real time command,
- lighting of the target area during the movement;
- transfer the image on the operator’s monitor during the movement;
- scanning of objects in the nominally specified planes (according to the regular program) with the discovery of fire (the fire seat);
- supply of water or low expansion foam to the fire seat, to the point specified by the operator;
- supply of the powder to the fire seat, to the point specified by the operator.

Due to its dimensions the installation is able to move through the corridors and common areas.

The report also describes mobile robotic installation for fire extinguishing, basic tactical and technical characteristics. It demonstrates the installation itself, and describes a method of its use in fighting fires.

In the future, we plan to expand the functionality of mobile installation for fire extinguishing, by increasing its tactical possibilities.

We plans to develop a mobile installation for fire extinguishing based on mobile robotic installation for fire extinguishing, to supply gas-generated foam, as well as with the possibility of work with the designs of high-strength materials, due to the waterjet cutting system.

Another example of fire robot is a mobile robotic installation for fire extinguishing MRUP-SP-G-TV-U-40-17KS, which is designed to solve complex operational tasks in the event of fires in railway systems for, mass transit, road and railway tunnels, train stations, indoor parking lots, in pedestrian tunnels, facilities of energy complex and industrial enterprises.

The report also describes the main tactical and technical characteristics of the robot MRUP-SP-G-TV-U-40-17KS, functions, specific use and application.

PROMISING TOOLS FOR AUTOMATIC FIRE DETECTING AND EXTINGUISHING

V.I. Fomin

State Fire Academy of EMERCOM of Russia, Moscow city

The concept of fire safety system of NPP provides an exception or minimizing the harmful effects of fire due to their elimination at an early stage, which leads to the use of safe and effective means of fire detecting and extinguishing.

Different types of fire detectors traditionally used nowadays provide detection of fire occurred when there is a naked flame, which is accompanied by a change in the physical properties of the environment (smoke, temperature rise).

One of the methods for early detection of a fire is to control the chemical composition of the air, changing due to thermal decomposition (pyrolysis) of overheated flammable materials which begin to smolder. At present, gas fire detectors responsive to changes in the chemical composition of the atmosphere, caused by exposure to fire are developed and produced by domestic industry.

One of the advanced technologies for early detection of fires to date is aspiration using forced air bleed from the protected volume with monitoring using ultrasensitive laser smoke detectors. This principle allows detecting combustion products at a very early stage - before the appearance of visible smoke: when smoldering products of combustion or their heating (evaporation of cable insulation, etc.).

Fire alarm systems are constantly being improved. There are new algorithms and methods for early fire detection. Great importance is attached to a reduction of false positive activation; the result was the emergence of multi-sensor fire detectors. Four independent sensors are integrated into the detector: carbon monoxide CO sensor, photoelectric smoke sensor, thermal sensor and light infrared sensor, which are controlled by an embedded CPU according to complex adaptive algorithms.

The most common, effective and universal extinguishing agent, according to domestic and foreign experts, is the water mist. The mechanism of flame suppression depends on the mode of combustion and the water supply method. Domestic manufacturers have developed devices allowing to eliminate “dead zones” when supplying water mist.

The production of sprinklers with a positive start and sprinklers with forced start and operation control is commercially developed. In conjunction with the launch of targeted alarm system any sprinkler or a group of sprinklers can be arranged. Extinguishing efficiency thus arises repeatedly as a sprinkler system is not “catching up” the fire seat and works in advance.

One of the modern means of fire extinguishing is nitrogen unit. Nitrogen units for fire extinguishing are based on the latest generation of membrane technology. They are effective systems for quick fire suppression by feeding nitrogen gas into the room, where there was a fire or explosion.

THE MAIN PROBLEMS IN THE DESIGN OF FIRE SAFETY SYSTEM OF NPPS AND POSSIBLE SOLUTIONS

D.G. Lanin,
ATOMPROEKT JSC

In recent years, due to the expansion of the scope of international cooperation of State Corporation "Rosatom", as well as anti-Russian sanctions and other crisis factors, the issues of competitiveness of the Russian nuclear industry come to the fore. Against this background, the problems of fire safety in the design of NPPs become even more relevant.

The main problems at the moment are:

- the presence of many supervisory and regulatory bodies, as well as other subjects of relations in the field of fire safety of NPPs with different and often conflicting interests, while the absence of decision-making center, for example, special or inter -agency body for fire safety regulation for such facilities;
- the lack of regulatory framework, which would provide clear and transparent mechanisms to ensure the fire safety of NPPs;
- the lack of universally accepted, legitimate conformity assessment tools of NPPs to fire safety requirements, including the risk-based techniques which meet the modern level of development;
- the lack of evaluation mechanisms for adequacy and necessity of technical solutions in the field of fire safety in view of the economic component in the various stages of the life cycle of NPPs;
- the lack of a clear concept of "innovative product", as well as implementation mechanisms for innovation at the NPPs.

All these problems "tangled", and every year they "tie the designers' hands" more and more resulting in an increase in the cost of design, construction and operation of facilities in terms of ensuring their fire safety.

As a result of working with international, foreign and domestic regulatory framework, taking into account the existing expertise in designing NPPs in Russia and abroad, as well as interaction with the representatives of the professional community there are a number of proposals for a possible solution to these problems, some of which are:

- a proposal to the Government of the Russian Federation on the establishment of an interdepartmental body regulating fire safety issues for NPPs;
- the development of a normative legal act regulating the general issues of the NPPs fire safety and defining its basic criteria in the context of nuclear and radiation safety, as well as the people's life and health. Giving the appropriate status to the document;
- the development of a set of documents (requirements or voluntary use of standard technical solutions) on the basis of existing regulatory instru-

ments that can be applied selectively and targeted for the implementation of specific provisions of the above normative legal act;
- the development and legitimization of high-tech tools for conformity assessment of NPPs to fire safety requirements on the basis of deterministic and probabilistic analysis of fires and fire simulation methods and testing safety conditions for people;
- the development of the evaluation mechanisms of economic efficiency of technical solutions in the field of fire safety, including innovative ones;
- the intensification of work in the field of industry standards for goods and services of fire-technical purposes, and also in terms of development and use of such a way of conformity assessment as full-scale fire tests.

Upon successful solution of the above problems, SC "Rosatom" can become an engine of technical fire safety regulation, both in Russia and abroad.

FEATURES OF NPPS' FIRE PROTECTION WITH LIQUID SODIUM COOLANT BY THE EXAMPLE OF THE BN-800 AND BN-1200

N.A. Lobanova
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For fast reactors BN-800 and BN-1200 a three-loop circuit is adopted in which the coolant of the 1st and 2nd loops is liquid sodium heated to very high temperatures that can in various leaks of pipelines and equipment to burn on contact with air. There is also a danger in case of contact of sodium with concrete or thermal insulation, since it may lead to the destruction of the building structure and/or the reaction of sodium and its combustion products with water from the heat insulation and the concrete due to thermal degradation.

When forming the design basis accident scenario, the following has been postulated: opening a through hole in the pipeline or in the vessel, equivalent to the diameter equal to two thicknesses of the pipe wall; jet leak of the sodium; sodium combustion takes place in a pool. This approach is traditional for Russian fast reactors, and justified with long-term operation practice and experience of real sodium leaks.

For the Power unit No. 4 of the Beloyarsk NPP, State Scientific Center of the Russian Federation – Institute for Physics and Power Engineering named after A.I. Leypunsky performed numerous computational and analytical study where maximum values of the following parameters of sodium fires have been identified for each sodium premises:
- the pressure and temperature of the gaseous medium;
- the temperature on the concrete surface of the ceiling, the floor and walls of the premise;

- the integrated mass emission in fire emergency exhaust ventilation system (only for design basis accidents), in a working exhaust ventilation, in adjacent rooms;
- the integral radioactivity emission for radionuclides (for circuit I premises) in emergency fire exhaust ventilation system (only for design basis accidents), in a working exhaust ventilation, in adjacent premises.

The analysis concluded that the construction fence of process facilities with sodium withstand the thermal and dynamic effects that occur in emergency situations. The passive extinguishing media adopted in the project (trays with water seal, drain systems) significantly limit the effects of combustion. The use of nitrogen fire extinguishing for sodium premises of the reactor BN-800 is considered inappropriate.

The results of the analysis are reflected in the "Special Technical Standards in Terms of Fire Safety of Capital Construction "Beloyarsk NPP, power unit No. 4 with the reactor BN-800", 2014, developed by Research and Technical Centre for Industrial and Fire Safety Ltd. and agreed in the prescribed manner.

The following was adopted as criteria for fire protection of sodium premises in case of design basis accident:

1) Excluding the possibility of occurrence of the high pressures and the strong dilution in the sodium premises in case of fire.

2) In case of fire the temperature of the gaseous medium in the room should not be significantly greater than ~ 270 ˚C or 370 ºC (for the first and second circuits, respectively).

3) In case of fire the temperature of thermal insulation of walls and ceiling should not exceed 200 ºC, for concrete floor, ceiling and walls – 100 ºC, wall and ceiling covering - 600–650 ºC (for circuit I premises) and 400 ºC (for circuit II premises).

4) Exclusion of moisture in the gas environment of the premises.

5) Emissions of radioactive substances into the adjacent rooms and the atmosphere should be as low as possible. The values of emissions caused by these doses should not exceed the prescribed limits.

6) During all switching in the ventilation systems, including those associated with the combustion of sodium, the operating point must be within the working area of the fan pressure characteristics.

For premises with sodium equipment the following measures are provided:

- measures to reduce the likelihood of hot sodium in contact with oxygen in the air;
- measures to prevent sodium from the contact with water;
- measures to prevent the formation of explosive concentrations of hydrogen-air mixtures in the event if the sodium contact with water or steam will take place;
- systems for detection of sodium fires;

- fire localization tools (self-extinguishing trays, sealed penetrations, thermal insulation and lining of ceilings and walls of premises).
- the use of fire emergency exhaust ventilation systems, as well as forced feeding of extinguishing media and materials to the fire seat of the sodium.

All sodium systems are located in some relatively sealed spaces which are an independent fire zone. The project provides the protection of concrete structures from the direct exposure to sodium and thermal effects by covering them with insulation and steel cladding.

In the design of a fast reactor BN-1200, the experience of the design and construction of BN-800 was fully taken into account. In the years 2012 -2015, Research and Technical Centre for Industrial and Fire Safety Ltd. carried out research on the topic of "R&D in Support of NPP Project with the Reactor BN-1200", in which the concept of fire protection of premises with sodium coolant was developed, which included:

- the analyzes of fire danger of the reactors with sodium coolant;
- the justification of design solutions for fire protection of the reactor BN-1200;
- the draft document "Regulations for Technological Design of Fire Protection for Premises with Sodium Coolant of NPPs with the BN reactor".

The project "Regulations for Technological Design of Fire Protection for Premises with Sodium Coolant of NPPs with the BN reactor" was widely discussed and revised with the participation of experts from leading organizations on the subject of BN reactors: Project Office BN-K OKBM, State Scientific Center of the Russian Federation – Institute for Physics and Power Engineering named after A.I. Leypunsky, Atomproekt JSC, JSC OKB "GIDROPRESS".

These Regulations for Technological Design are aimed to provide the regulatory bodies, project developers and licensees of nuclear power plants with fast neutron reactors with sodium coolant with the recommendations and guidelines on concepts, methods and design principles used to protect nuclear power plants from the hazards associated with leaks and burning of sodium coolant.

It is expected that these Regulations should complement the requirements for fire safety, stipulated in SP 13.13130, in respect of the buildings, premises, equipment and pipelines with liquid sodium coolant.

LIFETIME WARRANTY FOR FIRE PROTECTION OF LOAD-BEARING METAL STRUCTURING USING THE METHOD OF ACCELERATED CLIMATIC TESTS

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Fire safety of NPPs depends on the actual fire resistance of load-bearing metal structures which is provided by fire protection of the required thickness. The warranty periods of its exploitation are among the factors that determine the quality of fire protection. In economically developed foreign countries, there are regulations according to which the firms producing fire protection means justify the warranty period, during which they undertake to replace or repair the fire protection which lost its functional qualities. In the construction industry of Russia, such rules in relation to the means of fire protection are not available. That's why the problem of *establishing, justifying and validating the warranty periods* for maintenance of fire protection properties of load-bearing metal structures of NPPs is highly relevant.

To solve this problem, it is advisable to use the experience gained in the development and operation of critical samples of rocket and space technology.

The report examines the main provisions of feasibility demonstration and confirmation of the warranty periods for preservation of operational properties of products made of polymer composite materials developed by OJSC "TSNIISM".

The technique involves conducting accelerated tests according to GOST R 51372-99 in special climate chambers to *determine* and *predict* changes in the properties of materials and structures fragments in climatic aging under the conditions of products use. The determination and prediction of changes in the material properties and design is carried out by one or more characteristic parameters of aging.

In agreement with the customer the test results are included in the standards or technical specifications for the material and (or) structure and are used to determine and predict the persistence of properties of the products under the influence of these factors.

It is advisable to apply the available results for fire protection of load-bearing metal structures of NPPs in order to assess changes of the fire retardant effectiveness of used coatings over time, the definition of warranty periods of their operation, as well as evaluating the possibility of extending these periods. The object of research can be relatively small sample sizes or pieces of metal structures with flame-retardant coating applied to them, according to the accepted technology. In climate chambers available at OJSC "TSNIISM" accelerated heat and humidity aging of fire protection of metal structures can be carried out, which have considerable size (up to 3 m), for their subsequent tests in fired furnaces of the accredited testing centers (e.g., in the VNIIPO of EMERCOM of Russia), or on special benches used in OJSC "TSNIISM" to determine the basic characteristics of fire retardant coatings.

Thus, the methodological and logistical support existing in OJSC "TSNIISM" should be used (with appropriate refinement) to *establish, justify and validate the warranty periods* for maintenance of fire protection properties of the load-bearing metal structures of the NPPs facilities.

In this connection, it is proposed to develop industry guidance, regulating the works on the *establishment, justification and validation* of warranty periods for maintenance of fire protection properties of metal structures of the *operating and designed* NPPs.

DEVELOPMENT OF THE RATIONING SYSTEM OF NPP FIRE SAFETY

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FGU VNIIPO of EMERCOM of Russia

The relatively high probability of occurrence and development of the fire is caused by the technological features of the process of electricity production at nuclear power plants. On the one hand, the main industries use large amounts of combustible substances and materials. Thus, the fire load per unit area in cable premises of the reactors can be around 20,000 MJ/m^2, tens and hundreds of tons of fuel oil may be in the same system of circulation pumps and turbo generators. On the other hand, the extremely high energy saturation of the production causes the large number of high-energy ignition sources. In terms of fire risk, fast neutrons reactors of BN type have a considerable specificity. In such reactors, metal sodium is used as a coolant. When operating the existing power unit at Beloyarsk NPP, more than 100 tons of fuel sodium coolant is used in the 1st circuit heated to a temperature higher than the self-ignition temperature. Under these conditions, a leak of sodium Sin case of depressurization of the equipment automatically causes a fire.

Despite the significant fire danger of the sodium coolant, technological difficulties in the establishment and operation of liquid metal cooling loops of the reactor, the Program of the development of nuclear power in the Russian Federation provides the serial construction of power units with the reactors of BN type. This is due to a number of significant advantages of such reactors: the use of cheap Uranus 238, as fuel, and development of nuclear materials in the operation of the reactor, applicable for future use.

To date, the power units with BN reactors were considered unique, and security issues, including fire safety of such units were considered on an individual basis, with reference to a particular facility. To create a single fire safety system for the first time, in accordance with their program of mass construction, in 2015 a fire safety system of valuation of such facilities was developed. Regulatory requirements and approaches are defined based on a broad analysis of the results of Russian and foreign research and development works. On this basis, VNIIPO developed Amendment No. 2 to the set of SP 13.13130.2009 "Nuclear power plants. Fire safety requirements".

The Amendment No. 2 identifies the specific requirements for fire safety of neutron reactors. These requirements are the result of the technological features of the fast neutron rectors, characteristics of burning and extinguishing of sodium in the event of a fire.

The main specificity of fire danger of this type of reactors is the following factors:

- the availability of large amounts of pressurized and heated sodium coolant, under the operation of the reactor, above the auto-ignition temperature;
- the danger of chemical and thermal effects of burning sodium on unprotected concrete building and walling structures;
- the danger of sodium interaction with water of the reactor cooling circuit or with water supplied when using fire extinguishing with water;
- the complexity of physical and chemical processes on the surface of the coolant leaks in view of the specificity of interaction between metal-oxide-extinguishing agents in the application of extinguishing on the surface.

Amendment No. 2 to the set of rules implies the possibility of a combination of different methods of active and passive fire protection in order to achieve secure conditions for the fast neutron reactors.

Currently the Amendment was discussed after posting on the Internet, and is being corrected by the results of its review. Putting the document into execution is planned in the 2nd quarter of 2016 after the presentation at the expert council of Ministry for Emergency Situations and after harmonization with legal services of the Ministry.

MODERN EQUIPMENT TO EXTINGUISH POSSIBLE FIRES IN THE NUCLEAR INDUSTRY

Introduction:

- Use of hydraulic guns in liquidation of the accident at the Chernobyl NPP on April 26, 1986.
- The first deluge guns (hydraulic guns) for NPP machine halls.
- The purpose of the project. Name and purpose of the device.
- Selection of design of the deluge gun (hydraulic gun) for NPPs.

Main extinguishing agents:

Water - physical properties of the liquid

Foam - foam properties

Powder - powder properties

<u>Manual fire hose nozzles:</u>

- Main types of manual fire hose nozzles.
- Main specifications
- Combined, shut-off, universal fire hose nozzles with adjustable flow rate and spray geometry.
- Manual fire hose nozzles for extinguishing fires in electrical systems with voltage up to 10,000 V.

Deluge guns (hydraulic guns):

- Main types of deluge guns (hydraulic guns).
- Classification of deluge guns (hydraulic guns).
- Deluge guns (hydraulic guns) and their specifications.
- Portable deluge guns (hydraulic guns).
- Stationary deluge guns (hydraulic guns).
- Vehicle-portable deluge guns (hydraulic guns).

<u>Examples of completed projects.</u>

SMART FIRE EXTINGUISHING MODULE “ZARYA”

S.V. Lektorovich
LLC “Innovative firefighting systems”, Togliatti

Increasing the level of fire safety at nuclear power facilities

Increasing the level of fire safety at nuclear power facilities is the most important task in the framework of nuclear and radiation safety, as fire and its consequences can lead to more severe emergencies at these facilities.

Comprehensive improvement of the safety in the operating units

Measures for comprehensive improvement the safety of the operating units include:

- Modernization of control, monitoring and protection systems of the reactor with the introduction of a second shutdown system of the reactor;
- Modernization of safety systems (emergency cooling system of the reactor, accident localization system, emergency power supply systems);
- The introduction of gas extinguishing systems in buildings for control and protection systems of the power unit, etc.

Fire Protection Technologies introduced or being introduced at nuclear power facilities

1. Aspiration fire alarm - Beloyarsk NPP (power unit 4);
2. Structural fire protection (box) - Beloyarsk NPP (power unit 4)
3. Modules on water mist - Rostov NPP (power unit 1, 2)
4. The use of slow-burning oil (OMTI type) - Leningrad NPP-2 (power unit 1, 2).

Smart gas fire extinguishing modules "ZARYA"

Russia's first gas fire extinguishing modules integrated with the technology of the Internet of things:

1. Automatic pressure control in the module;
2. Automatic notification of the need for maintenance and routine operation;
3. Message to the operator when triggering the module or pressure reduction.

Gas fire extinguishing modules "ZARYA"

Advantages:

1. No pipework;
2. Ability to work in an automated and/or autonomous mode;
3. No special room for cylinders;
4. Costs reduction and ease of maintenance;
5. Savings: 30-40%.

Modular fire extinguishing unit intended to protect the premises for different purposes and categories of fire risk, and technological equipment, etc.

Modes of module operation:

- manual;
- automatic;
- autonomous.

Modifications:

1. Zarya - 3 (cylinder capacity, l - 3, the protected volume, m^3 - up to 4).
2. Zarya - 10 (cylinder capacity, l - 10, the protected volume, m^3 – up to 14).
3. Zarya - 22 (cylinder capacity, l - 22.5, the protected volume, m^3 – up to 30).

GFES Halocarbon 125, Halocarbon 227ea.

Today fire extinguishing modules "ZARYA" are successfully applied at the facilities of large public and private companies such as Sberbank of Russia, GAZPROM Mezhregiontransgaz, Russian Federal Security Service, Bank "GLOBEX", «ForteBank» JSC, VEB group, and so on.

FEATURES OF COMPLIANCE WITH LEGAL ACTS AND NORMATIVE DOCUMENTS OF THE RUSSIAN FEDERATION ON FIRE SAFETY DURING SURVEILLANCE ACTIVITIES

V.I. Sazykin

Fire Security Service of Rosenergoatom Concern OJSC

Nuclear power plants are high-risk facilities, and compliance with fire safety rules during their operation is a prerequisite for ensuring nuclear and radiation safety of the country.

In order to reduce administrative barriers for doing business, Federal Law dated 26.12.2008 No. 294-FZ “On the Protection of the rights of legal entities and individual entrepreneurs upon implementation of the state control (supervision) and municipal control” was developed, which establishes the rights and obligations of federal executive authorities and the audited legal entities in the organization and carrying out the supervision, including fire safety. It should be noted that the provisions of this law do not apply to the investigation of emergencies, such as inquiry for fires and ignition.

In accordance with the RF Government Regulation dated 05.05.2012 No. 455, the territorial authorities of the state fire supervision of EMERCOM of Russia are not empowered to establish a regime of permanent state supervision at NPP. In carrying out the state supervision, the verification of compliance with mandatory requirements is not permitted, if such requirements do not apply to the powers of the supervisory authority.

During the organization of supervision, the federal executive authorities are obliged to carry out the supervision only for legitimate purposes and to prevent the audited organizations in advance (except for unscheduled inspections, which must also have legitimate grounds). It is important to remember that the audited organization has the right to refuse the supervisory authority in the inspection (even unscheduled one) if there is no ground for it, or in case of irregularities in completion of necessary documents.

During inspections, when establishing violations, the federal executive authorities often refer to the Federal Law dated 28.08.2008 No. 123-FZ “Technical Regulations of Fire Safety Requirements”. It should be remembered that the requirements of this law do not apply to the hazard location that have been put into operation or the project documentation which was sent for examination before the date of entry into force of the law - that is, before 01.05.2009. The requirements directly referred to in technical regulations are only mandatory, and the requirements of voluntary application of standards (particularly sets of rules), are not required to fulfill.

When revealing fire safety violations during the inspection, the federal executive authorities prepare an order to eliminate violations, following to which it is possible to prosecute persons who committed the violation, in particular cases - administrative suspension of activity of the enterprise; in any case, the federal executive authorities shall monitor in the future the elimination of violations. It should be remembered that the administrative responsibility may also occur as a result of violations carried out during the audit, such as, for example, obstruction of the lawful activities of federal executive authorities’ officials.

Territorial authorities of the state fire supervision commit a number of typical violations in the organization, carrying out and registration of results of fire safety checks at NPPs. The officials of Rosenergoatom Concern OJSC, who are responsible for the maintenance of the federal executive authorities checks should study the practice of typical violations committed during inspections, with the aim of reducing unreasonable demands on their results and reducing the costs related of their elimination.

THE SPECIFICITY OF USE OF TOOLS AND MEANS FOR FIRE PROTECTION AT NPPS IN TERMS OF OCCURRENCE, DEVELOPMENT AND ERADICATION OF BEYOND DESIGN BASIS (SEVERE) AND CHEMICAL ACCIDENTS

In accordance with the Federal Law “On Fire Safety” dated 21.12.1994 No. 69-FZ, the main tasks of fire protection, including on-site divisions of Federal Fire-Fighting Service for NPPs protection, are fire-fighting and rescue operations in hazard locations.

However, for potentially hazardous locations, which include nuclear power plants (hereinafter - NPPs), the failure (or termination) of functioning of the safety systems (components) of nuclear facility, as a result of emergency situations of natural, man-made and terrorist nature, can lead to loss of control and its destruction followed by nuclear and radiation accident. As a consequence, - enormous damage, irreversible negative changes in the ecology of the subject or of the administrative-territorial unit, a significant deterioration of health and safety of people living in these territories over a long period of time.

In these circumstances, one of the most important factors in ensuring the safety of operating NPPs, their staff and the population of the areas is stable operation of nuclear facilities of power units with their complete control and management over the entire period of time, localization and liquidation of beyond design basis (severe) and chemical accidents, including those followed by a fire.

Exposure of hazards of fires and their consequences on the NPP staff and, above all, on the operational personnel involved in monitoring and control functions of nuclear facilities can actually lead to loss of its control, and as a consequence, to possible violations of the established limits of power unit security.

In this context, the improvement of system of emergency and fire preparedness and protection of power units of NPPs is an absolute condition for ensuring the security of nuclear-hazardous facilities.

Analyzing the conditions, the specificity and results of the origin, development and elimination of a number of fires and accidents that have occurred at the Beloyarsk NPP (1978), the Chernobyl NPP (1986), Fukushima NPP (2010) and other NPPs, as well as the results of the integrated emergency response exercise on scenarios of occurrence and course of beyond design basis (severe) accidents at the operating NPPs in 2010-2013 we can conclude:

- about the need to bring the forces and means of fire protection of NPPs to participate in special operations related to localization and liquidation of possible design basis (severe) and chemical accidents at NPPs and their consequences;
- about the advisability of individual design and manufacture of primary and special-purpose fire vehicles for NPPs, according to its most effective use both in terms of fighting fires and for special works to prevent (localization) of negative scenarios for beyond design basis (severe) and chemical accidents at NPPs.

The analysis of the development and fire suppression (accidents) at NPPs shows that in the shortest time possible, the fire protection can provide a set of special work on the localization of emergencies and protection of operational NPP personnel carrying out technical operations (works) for shutdown, cooldown of the reactor and keeping it in subcritical condition.

Subsection 1.3
RADIATION SAFETY, NPP ECOLOGY, EMERGENCY PREPAREDNESS

Topical area
RADIATION SAFETY

ACCOUNTING AND CONTROL OF NPP RADIOACTIVE SUBSTANCES DISCHARGE WITH BACKGROUND ACTIVITY

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Monitoring of radionuclides content in surface waters within NPP control areas is an integral part of environmental monitoring. Presence of technogenic radionuclides in NPP water supply and sanitation systems is caused by global fallout due to nuclear weapons testing, the Chernobyl NPP and Fukushima NPP accidents. They also appear in NPP dump systems. Currently, normative legal acts and methodical documents in the field of radiation safety do not contain direct instructions on the accounting of contents of natural and technogenic radionuclides in water bodies at justification of allowable discharge level. In this regard there arises a discrepancy of approaches to rationing of discharges of the enterprises of various industries related to significant amount of water consumption. The clearest way to demonstrate these disrepancies is to compare the inflow of long-living radionuclides into water bodies due to operation of Russian NPPs nuclear facilities with estimation of inflow of the same radionuclides due to electricity generation at TPPs and HPPs. The table shows that the annual activity of 3H, 90Sr and 137Cs in the discharge of NPPs and nuclear power generating facilities is comparable.

T a b l e . Comparison of controlled discharge of radionuclides by various power plants, Bq / year

Power plant	^{3}H	^{137}Cs	^{90}Sr
Kursk NPP	Up to $2.5 \cdot 10^{13}$	Up to $4.2 \cdot 10^{7}$	Up to $5.4 \cdot 10^{4}$
Kola NPP	Up to $1.2 \cdot 10^{13}$	Up to $3.7 \cdot 10^{5}$	-
Balakovo NPP	-	Up to $2,7 \cdot 10^{7}$	Up to $4.7 \cdot 10^{7}$

Argayashsky CHP	Up to $1.3 \cdot 10^{12}$	Up to $6.8 \cdot 10^{8}$	Up to $5.7 \cdot 10^{9}$
Sayano-Shushenskaya HPP	$(9{,}4 \div 24) \cdot 10^{13}$	$(4{,}7 \div 7{,}1) \cdot 10^{10}$	Up to $1.2 \cdot 10^{11}$

NPPs contribution to the inflow of radionuclides into surface waters can be determined on the basis of their circulation monitoring in the system water body-NPP taking into account the balance scheme of water supply and water disposal. The main sources of radionuclides in NPP water discharge are: water intake points, atmospheric atmospheric fallout, forming the surface water drainage of industrial and storm sewage and discharges. In the proposed model, all the vessels of NPP balance scheme of water supply and water disposal are presented in the form of individual cells, which characterize the main processes influencing the change in the activity in the system of water supply and water disposal. At the same time, the results of field measurement serve for identification of deviation of the measured values of NPP liquid discharge activity from background values in order to verify the observance of federal norms and rules requirements and confirmation of safe NPP operation as a source of waste water inflow.

MODERN REQUIREMENTS OF INTERNATIONAL AND RUSSIAN DOCUMENTS FOR RADIATION SAFETY OF NPP PERSONNEL

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The recent history of radiation safety began forty years ago, when the International Commission on Radiological Protection (ICRP) issued Recommendations of 1977 based on the Linear No-Threshold Hypothesis (LNT) of dependence between the probability of radiation stochastic effects development and the radiation dose. After this revolutionary event, ICRP has continued improvement of radiation safety framework. So far, the latest recommendations of ICRP were issued in 2007 [1], and they are also based on LNT.

Recommendations of the ICRP as of 2007 became the basis for the new edition of the International Basic Safety Standards (IBSS), issued in 2014 under the auspices of the International Atomic Energy Agency (IAEA) as the third part of the general safety requirements [2]. The release of this fundamental document has marked a new stage in the development of radiation safety systems in IAEA Member States. With the help of the IAEA, IBSS requirements have been already implemented in Belarus, and the similar process is taking place in Romania. In the near future, all EU countries are expected to bring their regulations in compliance with **Directive 2013/59/ Euratom [3], which is the European analogue** of IBSS.

Current radiation safety standards of the Russian Federation NRB-99/2009 are based on the International Basic Safety Standards for Protection Against Ionizing Radiation and for the Safety of Radiation Sources, issued under the auspices of the IAEA in 1996. In the next years, work is to be undertaken to make a new edition of RSS corresponding to IBSS [2]. In relation to ensuring radiation safety of personnel of the enterprises of nuclear power industry, this work has already begun under the auspices of Council for methodological support of radiation safety on the enterprises of Rosatom State Corporation.

The report examines the main provisions of the ICRP Recommendations [1], highlights the IAEA Safety Standards [2] and the expected changes in Bulgaria, as well as the main results of the work started in 2014, work on the improvement of radiation monitoring at the enterprises of the State Corporation "Rosatom" in view of IBSS requirements [4].

CASCADE AS APDMS ELEMENT

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INTRODUCTION. RATIONALE

Monitoring of NPP and external organizations personnel radiation dose status is a major challenge to ensure radiation safety of personnel.

Description of the situation in the subject area

The current realization of APDMS allows to solve the problems completely it is facing currently. However, taking into account the experience gained on operation of APDMS and the level of technology development, there is a number of promising developments allowing to improve quality, efficiency and convenience of the system use.

FACTORS DEFINING EFFICIENCY OF RADIATION DOSES MONITORING

- observance of radiation safety regulations by personnel;
- use of modern software and hardware tools for collecting, storing, processing and analysis of radiation doses of the personnel;
- comprehensiveness of the interaction with other automated systems for radiation safety.

PRINCIPLES NECESSARY TO BUILD A MODERN APDMS SOFTWARE AND HARDWARE COMPLEX TO IMPROVE ITS OPERATIONAL AND TECHNICAL CHARACTERISTICS

- openness and unification;
- safety;
- modularity (adaptability);
- resiliency.

ENHANCEMENT OF APDMS FUNCTIONALITY AIMED AT IMPROVEMENT OF RADIATION SAFETY OF NPP

- openness of data exchange with ACS. providing software interfaces for interaction;
- interface to the control equipment (film holder cabinet, key holder) for the accounting and regulated and authorized access for personnel to dosimeters and thermoluminescent dating;
- automated filing and submission of data on employees via the online form;
- collection and analysis of external doses parameters of personnel from pollution control equipment;
- control of personnel location in the areas of controlled access areas;
- process of registration and coordination of radiological work permit in electronic form;
- use of analytical tools for evaluation of data on personnel exposure doses.

FORECAST OF PERSONNEL RADIATION RISK FOR CONDITIONS OF PLANNED INCREASED EXPOSURE

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One of the basic principles of the document ""State Policy Principles On Nuclear and Radiation Safety in the Russian Federation until 2025" (No. PR-539 as of March 1, 2012) is "harmonization of legislation of the Russian Federation with the international legislation on the basis of law-enforcement practice" (Section IV, 9b) and "realization of the principle of socially acceptable risk" (Section IV, 9d). IAEA Safety Fundamentals of 2007 (No. SF-1) and the International Basic Safety Standards of the IAEA in 2011. (№ GSR-3) require that employers shall provide all employees with adequate information on health risks associated with their occupational exposure. According to to the current Russian Standards of Radiation Safety (NRB-99/2009), Paragraph 3.2.1, the planned increased radiation (PIR) is allowed for men over 30 (when approved) only after their informing on possible doses of radiation and risk for health. Standards NRB-99/2009 define the allowable PIR modes in terms of effective and equivalent doses. In particular, at employee exposure, the effective dose from 0.1 to 0.2 Sv per year, further annual doses must not exceed 0.02 Sv. In this work we obtain forecasts of the radiation risks to the health of the Russian nuclear industry personnel with acceptable modes of external PIR. Since Standards NRB-99/2009 were developed taking into account the recent recommendations of International Commission on Radiological Protection in 2007, (ICRP Publication 103), for the calculation of the radiation risk there were used model, recommended by ICRP Publication 103. The maximum lifetime risk of radiation-caused malignant tumors (MT) corresponds to the exposure at age 30 with a dose of 0.2 Sv and higher until the age of 60 years inclusive - 0.02 Sv per year (total of 0.8 Sv). At further radiation, after 60 years, the lifetime risk decreases because of reduction of the remained life expectancy of the person. The maximum lifetime risk of cumulative dose with an acceptable (as per NRB-99/2009) external PIR for the Russian personnel is 1.8x10-2. This risk is deemed acceptable, as it makes 36% of the risk limit for dose limit for personnel, accumulated for the entire period of employment under normal conditions of use of radiation sources: (1.0x10-3/year) x (50 years) = 5.0x10-2 (para. 2.3 NRB-99/2009). The maximum potential damage is expected at the age of 30; it is equal to loss of 16 years of life per one radiation caused event of MT and practically does not depend on PIR dose. Thus, the aim of personnel protection optimization the in the PIR conditions is to reduce, as far as practicable, the lifetime radiation risk, both by reducing the doses and by increasing age at exposure.

CURRENT STATE OF PDM TECHNICAL MEANS AND THEIR DEVELOPMENT

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For successful development of nuclear technologies, one of the promising directions in modernization and technological development Russian economy, the organization of modern radiation control is necessary both for ensuring radiation and nuclear safety of facilities, and for achievement of a compromise in society around this direction. Besides, by the results of of radiation accidents analysis carried out by "Rosatom" in Japan in 2011, the justification of the specified direction development importance is a serious lag of domestic nuclear instrumentation from world level. Thus, the enterprises of "Rosatom" still use outdated domestic thermoluminescent (TL) complexes of personal dosimatric monitoring (PDM) of DVG-02T type, the FTC-02 and HSC-01 with manual single-piece loading of the detectors. More advanced o systems such as AKIDK-201, AKIDK-301, AKIDK-401 type, Sapphire-001 are not widely used at NPPs because of their lack of performance.

One of the most expected consumer properties of the PDM systems is the increased reliability and productivity of information reading that leads to, on the one hand, use at the enterprises with completely automated complexes which are still not produced in Russia. Therefore, more and more NPP dosimetry services purchase expensive automatic foreign TL PDM systems of RE-2000 and Harshaw 6600 types. Their main disadvantages are not only their price, but expensive maintenance and operation. On the other hand, the increase in productivity can be provided by changing the stimulation process: from rather inertial thermal one to express optical (method of optically stimulated luminescence, or OSL). The last one also significantly reduces the cost and lowers energy consumption. Similar OSL systems with automatic and semi-automatic distribution of dosimeters developed by the designers in cooperation with the specialists of Ural Electromechanical Plant. Their release has also begun in 2010 by Landauer Company. In both systems, the authors used a patented detector material based on anion defective corundum (TLD-500 or α-Al_2O_3:C detectors).

The work also discusses the issues of correctness of the determination of the individual equivalent doses for the personnel exposed to mixed gamma-neutron irradiation of the whole body and beta radiation of skin and crystalline lens. An important factor of special attention to these issues are the ICRP recommendations, including the reduction of the dose limit of beta-irradiation for the crystalline lens from 150 to 20 mSv, significantly tightens the requirements for the selection of technical means of such doses measurement.

Topical area

NPP ECOLOGY

BIOFOULING PREVENTION AND CONTROL IN SERVICE WATER SUPPLY SYSTEMS AS A PREREQUISITE ON SAFE NPP OPERATION: STEP 1 – UPGRADING BIOCHEMICAL MONITORING (BCM) SYSTEM

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The phenomenon of organisms conglomerations at the interface between liquid and solid (inert natural and artificial solid substrates) phases leads to fouling. This is a serious issue at process facilities (inner and outer vessel surfaces, all types of hydro technical and navigational structures and equipment in oceans, seas and inland waters). The conglomerations are primarily formed by special organisms (plants, animals and microorganisms) that exist temporarily or permanently attached to some surface. Construction and operation of power facilities with further transformation of naturally impounded reservoir into an impoundment or a heat think creates favorable conditions for development of such organisms – resident (local) species and colonizers from all over the Earth. Foulers are able to form colonies all over the length of the service water system (SWSS) – in heat sinks, make up reservoirs, hydraulic works, and inner sections of the SWSS. Almost all surfaces in contact with the water passing through plant equipment without any physical, chemical, or physical & chemical protection means are suitable for their development.

Anti-fouling screening was conducted for SWSSs (including heat sinks) of four Russian NPPs in 2014-2015. This screening was based on a standard pattern and later introduced into Method MT 1.2.1.15.1043-2015 as Attachment D. As a result, several dominant species of thermophile invertebrates were found, which are potential (and in some cases are actual) foulers of outer hydraulic works and SWSS equipment. Among these are: 3 species of mussels (*D. polymoprha, D. bugensis, M. leucophaeata*); unknown* number of moss animal species of *Plumatella* genus *(P. fungosa*, *P. emarginata, P. repens, P. geimermassardi, P bombayensis**); 1 species of colonial hydropolyp (*Cordylophora caspia*); 1 species of acorn shells (*Amphibalanus improvisus*); 1 species of entoproctes (*Urnatella gracilis*), 1 species of sponge (*Epydathia fluviatilis)*. Over the half of these species are colonizers. Their active reproduction in the surface area of water reservoirs and in the internals of the SWSS is caused by the dissemination phases of their life cycle, since larva and vegetative buds are spread all over the system with running water, and by their ability to feed from water (filter-feeding organisms, predaceous zooplanktonphages).

Under non-monitored reproduction, fouling leads to biological disturbances, e.g. operation disturbance, including biological damage to materials, movable parts stuck, including in NPP safety systems, reduced heat transfer, etc. It leads to economic losses and creates environmental hazard, and threatens the operational safety of NPP itself.

Target systems of foulers are intake facilities, SWSSs and cooling water outer circuit if it is non-closed and forms a common system with the water supply reservoir that contains disseminating and dormant phases of foulers and related organisms. In NPPs equipped with mist cooling systems (cooling towers) in addition to the cooling water systems, water temperature in the heat sink is sufficiently higher than that of the environment, and the development of one of *Legionella pneumophila* pathogenic microorganisms is then possible. Its individual serotypes cannot influence the NPP safety, but they may cause disease.

Gradual BCM adjustment for of the screening of foulers as the source of biodisturbances, and the use of BCM as the method for preventive protection planning helped identify another important aspect in the fouling formation. In the period 2007-2015, due to the long-term dynamics of thermal conditions registered by the environmental in-process monitoring, response displacement (pH) to alkaline area, and the fall in the water level in Udomlya and Pesvo lakes (caused by combined impact of the reactor scale-up at Kalinin NPP and natural causes), after apparent extinction in the lakes (and other parts of the SWSS), rapid and essential adaptation of *D. polymorpha* dominant fouler under new conditions happened. The adaptation phenomena included (1) sufficient displacement of reproduction period, (2) change of lifetime, and (3) colonization of new natural and technogenic ecotopes (shallows and cooling tower reservoirs).

Under the existing Contract, an experiment on the assessment of the efficiency of the standard NPP approach to plants for zebra moussel control was conducted. The use of sodium hypochlorite for population under current response displacement to alkaline area has provided deplorable result. The observed phenomenology of changes in some biological features of this species, as well as some other species**, appearance of new species of aquatic organisms, including foulers, and the reduction in the efficiency of one of the most popular means of control due to a number of reasons have proved the need for switching from the standard BCM approach to Method MT 1.2.1.15.1043-2015. In particular, this Method recommends some basic modes of monitoring. One of the modes introduced by this document is regular and is intended for screening for adaptive changes, and obtaining current data for preventive measures**** of protection against fouling, and the assessment of the efficiency of such measures.

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*the taxonomy and geography (probably, a colonizer from South-East Asia) are to be determined

** for example, *P. fungosa,* moss, which creates biological disturbances in the SWSS of Kursk NPP

*** in 2014 – 2015, there were found three new species of foulers (*M. leucophaeata* from Central America, *P. geimermassardi* moss (both species were found on Koporye Bay used as the reservoir of service water for Leningrad NPP) U. *gracilis* (heat sink of Rostov NPP)

**** planning of an advanced ***preventive*** **program** (with consideration of the experience) for fouling monitoring can be considered as the only alternative to the response measures, and as one of the basic prerequisites of NPP safety. Planning shall be executed with implementation of several basic principles: 1) ***fouling prevention strategies (proactive)*** shall take preference against the response measures, and shall also solve (2) ***other related issues (***metal corrosion and deposits formation); (3) choice of the preventive strategy which shall be ***separate*** for each power plant and for individual water consumers within its structure , and it shall be based on the water supply schedule and system type; (4) the strategy applied shall be ***environmentally safe,*** especially if small lake or multi-purpose lake system are used as the heat sink. Specific measures within the strategy shall be planned with consideration of (5) the knowledge of the way of fouling formation as the combination of physical and biological processes, composition and dynamics for fouling formation on the weakest parts of the system for each individual case, and (6) expected efficiency of the chosen measure with consideration of safety, and (7) its planed cost; (8) fouling ***monitoring*** shall be used as the basic method of the preventive program.

ENVIRONMENTAL ACTIVITIES OF ROSENERGOATOM CONCERN OJSC: PROBLEMS AND REMEDIES

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Environmental protection and resource management are the overriding priorities of Rosenergoatom Concern OJSC (hereinafter the Concern). For compliance with environmental regulations, NPP ecological services perform ecological monitoring and environmental safety estimation for prompt and efficient response for the minimization of the environmental impact. The principal tasks of the environmental services are control over the compliance with the regulations for environmental recourses management.

The Concern's Environmental Policy updated in 2014 shows its principles and commitments in terms of its environmental activities. This Policy is aimed at ensuring the NPP safety level sufficient for the management of the short- and long-term impact on the environment, personnel and citizens in order to preserve natural ecosystems, their integrity and survival. The Concern's environmental safety system has developed and improved each year. This fact is evidenced by the annual improvement of the environmental impact factors and strengthening of the NPP image.

To achieve the goals and implement the basic principles of the Environmental Policy, the Concern is committed to introducing and supporting the best international methods of environmental management.

All operating NPPs and the Concern's Central Office (CO), acting as supervision body, are certified for compliance with ISO 14001:2004 international standard and with GOST R ISO14001-2007 national standard. In the course of the certification, the auditors noted the high quality of the organization of the works related to formation and development of the environmental management system for the Concern's CO and for each operating NPP.

All the efforts to implement the Environmental Policy of the Concern virtually prove Concern's commitment to compliance with environmental regulations, applicable regulations and international standards, as well as the principles of informational transparency and accessibility.

Over the last 14 years, the amount of discharges and emission of polluting substances into the environment has reduced almost twice, and continuous optimization of formation and temporary storage of production and consumption wastes, their placement at own landfills and transfer to special companies are conducted.

NPP costs for environmental protection amounted to 3,406 mln RUB in 2015. No sanctions for environmental violations were imposed.

The Concern has constantly improved and updated guidelines and standards for environmental safety assurance and environmental protection.

The following issues are the priorities of environmental protection: retaining the achieved level of NPP responsibility relating to compliance with the regulations, and the prolonged validation of permissions and licensing documentation relating to environmental protection; implementation of the best available purification techniques of industrial atmospheric and water emissions, handling of production and consumption wastes, efficient use of resources; optimization of programs and schedules to control discharges of pollutants to the environment, correspondence between technical and methodological background for in-process environmental monitoring, compliance with international and national standards.

Topical area

EMERGENCY PREPAREDNESS

IMMEDIATE EVALUATION OF NPP EMISSIONS UNDER DESIGN-BASIS ACCIDENT CONDITIONS

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Federal rules and regulations in the area of nuclear power use providing safety of nuclear plants (NP) stipulate creating radiation monitoring systems which should provide measurement of controllable parameters values describing radiation situation in rooms and at NP sites, sanitary-protection zone and control area in all NP operation modes, design-basis and beyond design-basis accidents.

Providing plant-wide radiation control NP site is supposed to be equipped with the system of spectrometric control stations which issue measurement data used to generate the scenario of fission products coming into the atmosphere and make a forecast of their further dispersal considering current weather conditions.

For correct interpretation of spectrometric measurements results form and space position of radiation source defined by weather conditions and ways of fission products dispersal into the atmosphere have to be known. The report covers the procedure of modeling atmospheric processes relating to fission products flow and dispersal in conditions of NP site development and using it as a base for evaluation of fission products structure and mass according to spectrometric measurement data.

Using the procedure of modeling and evaluation of emission in real-time mode is provided by compiling the library of prior fission products dispersal scenarios during the design-basis accidents and considering repeating weather conditions typical for the area NP is located. In NP malfunction conditions compiled library allows promptly to set up correspondence between the data received from spectrometry and emission scenario taking place. Forecast of fission products dispersal in the atmosphere is made basing on the generated scenario of fission products emission.

General issues relating to designing algorithmic and software support for the decision-oriented information system in emergency cases at NP are also covered by the report.

MODERN TECHNOLOGIES PROVIDING SCIENCE-RESEARCH SUPPORT OF EMERGENCY AID GROUP IN CASE OF NP ACCIDENTS WITHIN ROSENERGOATOM CONCERN EMERGENCY SYSTEM. DESIGN AND EXPERIENCE

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In terms of Federal Targeted Program "Providing nuclear and radiation safety for 2008 and the period until 2015" software and hardware complexes (SHC) intended to support decisions relating to safety measures in case of radiation accidents at nuclear radiation dangerous objects were designed.

SHCs supporting decisions made in case of radiation accidents at NPPs are an integral part of science-research support of Rosenergoatom Concern Emergency System.

Software utilities for assessment of radiation doses on population based on Gaussian models of pollutant ventilation in the atmosphere got widespread development and implementation after Chernobyl accident. These software utilities provided the basis for most SHCs used by science-research institutions, technical support centers (TSC) when making forecast of doses for population and working out recommendations on protection measures during trainings carried out by Rosenergoatom Concern Crisis Center.

IBRAE RAN designed:

- SHC "System of emergency forecast of radionuclide dispersal in the atmosphere for RF constituent entities located in 100 km zone around NPP". SHC is intended to model detailed flow of radionuclides considering peculiarities of the area and basing on actual and forecast space-time fields of meteorological data.
- SHC "Software and hardware complex calculating gas-airborne pollutant flow in three-dimensional geometry of industrial NP site development". SHC uses improved methods of on-sites doses calculation. Aerosol deposition on horizontal and vertical surfaces of buildings at the site is evidence-based and supported by mathematical models.
- 3-D SHC "Specialized 3D visualizer of gas-airborne emissions dispersal in the area NPP is located and emergency services actions in terms of accident progressing" is intended to be used during trainings.

All SHCs are based on the models verified by the data of natural experiments (ACURATE, Joint URBAN, etc.) and real accidents (accident at the Fukushima NPP in 2011 and radiation incident in Elektrostal in 2013).

SHC prototypes were successfully tested in terms of Complex emergency response training (CERT) at Leningrad NPP in 2015.

SHCs are constantly developed and improved in terms of actual directions related to providing science-research support, response to radiation accidents at NPPs.

INTERNATIONAL REQUIREMENTS TO THE STRATEGY OF POPULATION PROTECTION IN CASE OF AN ACCIDENT AT NPP

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In November 2015 under the aegis of International Atomic Energy Agency (IAEA) there was issued the last, seventh, part of General Safety Requirements named "Preparedness and Response for a Nuclear or Radiological Emergency" (GSRPRE) [#01] which replaced IAEA Safety Requirements No. GS-R-2 issued in 2002 [#02].

Publication [#03] shows that Russian system of population protection in case of an accident at NPP on the whole does not meet international safety requirements of GS-R-2. This system has changed slightly for the last years and it is safe to say [#04] that it does not meet international safety requirements of [#01].

The present paper studies basic principles of population protection strategy in case of beyond design-basis accident at NPP as applied to the accident at the Fukushima Daiichi NPP in Japan started on the 11th of March 2011 [#05]. Decisions related to protection of population being the resident in the neighbourhood of the Fukushima Daiichi NPP are made by the Government of Japan considering international requirements specified in GSRPRE and requirements of Russian Federation and Belarus legislative base. Belarus is a new-comer country which starts designing National nuclear program. IAEA provides comprehensive assistance to such countries thereby Belarus is so far the only country implemented GSRPRE general provisions to the national base regulating NPP engineering and operation [#06].

Using "Fukusima stress-test" allows to reveal those areas where basic requirements of regulating documents of Russian Federation relating to providing emergency preparedness and population protection in case of an accident at NPP have to be revised.

SOME EXAMPLES OF DESIGNED ANALYSIS OF BEYOND DESIGN-BASIS ACCIDENTS CONSIDERING MOBILE EQUIPMENT FOR VVER-1000 REACTORS

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For the last years according to ROSATOM program "Actualized measures for reducing the consequences of beyond design-basis accidents at NPPs" there have been taken measures to improve safety of operating components of VVER-type NPPs by introducing additional equipment. The present work reflects global processes aimed at improving safety at NPP in the light of the accident at the Fukushima in 2011.

The present report contains some results of this work related to the analysis of using mobile pump units and diesel generators for VVER-1000. Originally additional equipment was supposed to be used for initial events related to the loss of ultimate heat sink and major plant blackout. Introducing mobile equipment and revision of manuals on beyond design-basis accidents (BDA) control required carrying out analysis for also other initial events to prevent their transfer into severe accidents or for mitigation BDA. Such modes are: primary coolant leak, primary to secondary leak with unfitting intake unit at emergency steam generator.

Many ways of use of mobile equipment combined with the available equipment were described in terms of the analysis which gives an operator the opportunity to choose how to limit the consequences of beyond design-basis accident depending on the situation in a certain component. The advantages of each way are demonstrated considering actual characteristics of mobile equipment and connection diagrams for different components.

Analysis made allows to conclude that additional measures and mobile equipment introduced into NPP components improve significantly safety of VVER-1000 type NPP for initial events related to the loss of ultimate heat sink and major plant blackout, primary coolant leak combined with multiple failures.

MODERN APPROACHES TO INTERVENTION JUSTIFICATION IN CASE OF RADIATION ACCIDENT AT NPP

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In Rosenergoatom Concern OJSC there is an established procedure of applying criteria for making decisions regarding measures of personnel and population protection in case of radiation accident at NPP based on the principle of intervention optimization (MU 1.2.5.03.0035-2010). This procedure was tried-out during trainings in the period from 2010 till 2015. Gained experience, analysis of protection measures taken after the accident at the Fukushima-1 NPP (March 11, 2011) and also modern IAEA standards allow to define further direction of this procedure development.

It is supposed that the procedure of intervention justification in case of severe accident at NPP should include five stages. Its short description is given stage by stage in figure 1.

The report takes up actual condition of each element of the intervention justification procedure in case of radiation accident in terms of above-mentioned approach.

ZONES OF PROTECTION MEASURES PLANNING AS A KEY ELEMENT OF MODERN EMERGENCY RESPONSE SYSTEM. CHALLENGES OF DEVELOPING CONCEPTION OF EMERGENCY ZONING AND POSSIBLE SOLUTIONS

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NPP safety is provided by consequent implementation of defense in depth concept where the last level is emergency planning. Key element of emergency planning system in terms of population protection is defining special territories or zones of protection measures planning. Herewith safety level of population in the area NPP is located directly depends on how deeply the conception of territory zoning has been thought out, how accurately the responsibilities between the organizations involved in emergency response on these territories have been assigned and how the procedure of taking protection measures has been carried out and also how the correspondent plans are provided by (material, financial and human resources).

Emergency zoning system considering emergency experience at nuclear power plants in Three Mile Island (US) in 1979, Chernobyl (USSR) in 1986 and Fukushima (Japan) in 2011 is detailed in modern IAEA standards.

Predefined emergency zones around NPP are also stipulated by RF regulatory and legal base. At the same time it is still a challenge since there are several zoning systems being practised simultaneously (5 zoning systems). These zoning systems were designed separately by different authorities and also different work groups within one authority therefore they correlate with each other slightly.

At the present time there are no thought-out and discussed by the specialists of the authorities concerned (FMBA of Russia, EMERCOM of Russia, Rospotrebnadzor, Rostechnadzor, ROSATOM) conception of emergency territory zoning around NPP, criteria and principles of defining sizes of corresponding zones and also approaches to planning and taking protection measures. This impedes accurate assignation of responsibilities and tasks between the involved emergency response parties.

The report discusses contradictions in existing regulatory documents setting the requirements to territory zoning in case of emergency. Approaches to developing emergency zoning conception in case of radiation accident at NPP considering modern IAEA standards and experience of taking measures for population protection after the accident at the Fukushima-1 NPP (Japan) are proposed in the report.

Секция 2

NUCLEAR POWER DEVELOPMENT

Subsection 2.1
ADVANCED NUCLEAR POWER UNIT DESIGNS

DIFFICULTIES IN EMERGENCY AFTERHEAT REMOVAL FOR FAST-NEUTRON REACTORS WITH SODIUM COOLING

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Hereinafter technical solutions are presented and analyzed to bring to light actual difficulties in emergency afterheat removal from the active zone of fast-neutron reactors with primary sodium. Hereinafter the main variant of emergency heat removal system (EHRS) is presented based on standalone heat exchangers (SHE) submerged into primary circuit coolant with final heat removal into the environment via intermediate circuit according to DRACS variant approved in the projects of European fast reactor (EFR), China Experimental Fast Reactor (CEFR) and advanced high-power reactor with primary sodium. Advantages: The system is fully-autonomous being based on passive principles with natural coolant circulation in reactor tank (NC). The disadvantage of the system is the necessity of additional expenses for development, justification and operation. The aforementioned description of experimental stands and calculation codes used for EHRS characteristic researches. Test reports a presented denoting the possibility of emergency afterheat removal in fast-neutron reactors with the help of EHRS and SHE. Every analyzed variant of cool-down testifies to the fact that output coolant temperature from FA heads in the active zone descends as compared to forced circulation mode (FC). Coolant temperature in the upper chamber fringe region increases as compared to FC. Heat distribution between the output of active zone assembly imitators, mobile thermal probes radially and heightwise, thermal probe joints to SHE, and intermediate heat exchangers (IHE) shows that being in NE mode the hot coolant elevates to the top of coolant chamber along the central pole in the tank. Extensive hot zone is formed in the mixing chamber upper zone transiting the coolant to IHE (SHE) input. The results of calculated justifications confirm EHRS efficiency in cool-down modes.

DEVELOPMENT OF ADVANCED REACTOR TECHNOLOGIES OF THE 4TH GENERATION UNDER THE INTERNATIONAL FORUM «GENERATION-IV»

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Hereinafter the review of conducted works is presented under the International Forum «Generation-IV» (IFG) including six advanced reactor technologies chosen as 4th generation reactor technologies:

- supercritical gas-cooled reactor;
- sodium fast reactor;
- fast gas-cooled reactor;
- supercritical water cooled reactor;
- lead fast reactor;
- molten salt reactor;

IFG structure, global goals, approaches, and its schools description is provided. The description of all reactor technologies developed under IFG is provided and technologies developed by SC «Rosatom» in the first instance; action plans and their realization, and current state of every reactor aspect are described.

Additional information is given concerning the methodological workshops acting under IFG: concerning risks and safety, non-proliferation and physical protection, and economic model. IFG workshop activity is describe paying attention to development of project safety criteria for 4th generation sodium fast reactors.

VBER-600 REACTOR FOR MEDIUM-SIZE UNITS. MAIN SCHEMATIC AND DESIGN CHARACTERISTICS

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Medium-size nuclear power units (acc. to IAEA classification - 300-700 MW(e)) may be considered as an advanced autonomous field of nuclear power application in our country in addition to high-power NPP units.

Mostly unit capacity decrease is followed by capital expenses rising, therefore the reactor and unit project is to include innovation decisions making the project economically attractive and adapted for regional power systems.

According to OKBM Afrikantov OJSC in the aforementioned context traditional decisions of transport reactors will be more useful (465 transport units have been constructed with the total operating experience of 9500 reactor-years).

OKBM Afrikantov OJSC co-operating with NRC «Kurchatov Institute» has developed a 4-loop sectionalized reactor with VBER-600 reactor (Ne =615 MW) based on unified heat exchanger loop with a capacity of 415 MW (155 MWte) created to be used as a part of a medium-size unit. The project combines traditional transport nuclear steam supply systems (NSSS) and decisions proved to effective in civil power field.

Technical decisions concerning the systems and reactor equipment determine reduction of construction, installation time, capital and operation expenses of NPP units, improve reliability, safety (acc. to the requirements of 3+ generation), operation, and become adapted to regional works (by power and operating load).

«NiAEP» OJSC has compared technical and economical characteristics of a two-unit NPP with VBER-600 reactor and VVER TOI unit and concluded that VBER has become an advanced field of development.

According to safety level VBER-600 units meet the standards of advanced nuclear plants of 3+ generation and can be situated close to residential areas.

Unified heat exchanger loop with a capacity of 415 MW (155 MWte) is a technical platform for power line creation of VBER type with a capacity of 600MW (4 loops), 450 MW (3 loops) and 300 MW (2 loops) using sectional methods of industrial production and development.

RESULTS OF CONCEPT-PROJECT DEVELOPING CONCERNING MEDIUM-SIZE NPPS UNITS

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In 2014 «NiAEP" OJSC collaborating with NRC «Kurchatov Institute», EDO Gidropress OJSC, OKBM Afrikantov OJSC, and RDIPE OJSC finished conceprt projects for a two-unit medium-size NPP with VVER-600 and VBER-600.

The works were conducted according to an agreement with Design Affiliate of Rosenergoatom Concern OJSC upon a decision a joint meeting of STC №1 «Nuclear power units and power plants» SC «Rosatom» and STC Rosenergoatom Concern OJSC concerning «Comparative analysis of technical and economical characteristics of two-unit NPPs and nuclear power and district heating plant of medium size» dated 09.04.2013.

In operation the following has been accomplished:

1) Technical and economical requirements to the project of a medium-size unit with VVED;
2) Medium-size NPP safety concept with VVER-600, and VBER-600 reactors;

3) Optimization of technological and layout decisions, site plan, and building included into NPP units with VVER-600 and VBER-600 reactors;
4) Concept development for medium-size NPP with VVER-600, and VBER-600 reactors;
5) Technical Specification for medium-size NPP project by the example of two-unit NPP with VVER-600, and VBER-600 reactors;
6) Comparative analysis of medium-size two-unit NPP concepts;
7) Binding of two-unit NPP with VVER-600, and VBER-600 concept to the conditions of Kola NPP-2 site.

Main reference points in medium-size NPP concept developing are:
1) Net electrical power not less than 600 MW;
2) Installation time for the NPP unit from the first concrete until start-up:
   - not more than 48 months for the head block;
   - not more than 40 months for the serial block;
3) Estimated costs reduction for the serial block installation compared to VVER TOI project at least for 25%.

Medium-size two-unit NPP concept projects were developed according to the requirements of Russian Government Regulation N 87 dated 16.02.2008 «Regulation on Composition of Design Documentation Sections and Requirements to Their Contents», and consisting of the following:
- Design Notes;
- Equipment and materials specification;
- Installation cost calculation;
- 3D configuration and drawings.

The criteria mentioned in Technical and economical requirements to the project of a medium-size unit have been fulfilled, besides that,costs for the serial block installation compared to VVER TOI were reduced for 27%.

The fact confirms that MS NPPs have an important advantage of lower costs for project realization and lower risks for investors.

Besides, the prospect potential of medium-size two-unit NPP with VVER-600, and VBER-600 reactors may be fulfilled in later stages of the project due to the combination of innovative technologies and traditional solutions.

Potential application of MS NPPs:
- areas where traditional natural power sources are impossible and power lines are absent or restricting power generators;
- as an UPS for important objects of national standing;
- seawater desalination being a problem of current concern due to possibility of fresh water shortage.

According to the developed safety concepts MS NPP units with VVER-600 and VBER-600 reactors belong to III+ generation and have competitive characteristics for the international market, therefore the pilot NPP only needs a site to be constructed.

PROJECTS OF UNITS FOR SNPP BASED ON BOILING WATER REACTOR OF BNRAHPS TYPE

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RDIPE OJSC has developed a power line of boiling nuclear reactors of autonomous heat and power supply (BNRAHPS). BNRAHPS reactors are designed as energy sources for small nuclear power plants (SNPPs) that are able to function autonomously in remote and hard-to-reach areas including the Extreme North and Polar zone environment conditions of the Russian Federation.

BNRAHPS unit heat power is 100–360 MW. Base mode electrical power is 100 MW. Heating load range in co-generation mode is 80 – 120 Gcal/h.

BNRAHPS unit heat power is 45–180 MW. Base mode electrical power is 45 MW, electrical power in condensation mode is 60 MW. Heating load range in co-generation mode is 50–60 Gcal/h.

Boiling water reactors become the basis of power engineering development in the areas possessing no power system due to variety of advantages:

- single-loop heat removal and substantial reduction of capital building expenses as a consequence;
- low pressure in the circuit;
- high rates of inherent safety and self-regulation due to negative void and temperature coefficients;
- fully variable natural coolant circulation;
- extended service life up to 80 years;
- inter-load period (over 2 years).

Unit load following mode makes it possible to deliver the operating load of 100% – 0% – 100% N nom.

Much attention is given to safety issues. Five levels of defense-in-depth and six channels of reactor emergency cool-down are provided.

BNRAHPS based SNPP may become the basis for economical and social development of the Polar zone. Co-generation work of nuclear power and district heating plant will provide the necessary amount of heat and electricity for plants, producing companies and residents. The power line being developed instead of a single unit will make it possible to flexibly respond to the prospect customers’ requirements.

AUTONOMOUS THERMOSYPHON ARPS OF VVER PRIMARY CIRCUIT

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Autonomous afterheat removal passive system of VVER primary circuit active zone («ARPS R») removes heat by means of exchanger equipment based on close-loop vapor and condensing equipment — two-phase thermosiphons (TPTs). TPTs guarantees efficient, reliable and safe heat removal. Being an autonomous close-loop device for heat removal, TPTs create a system of additional barriers between the energy extraction source and ultimate heat sink. TPTs, removing the heat from the primary circuit to the ARPS intermediate circuit coolant become an additional barrier on the way of possible nuclear debris travel outside of containment, which boosts the reactor safety level in conditions of inter-circuit leakage.

ARPS R contains 4 loops removing the heat from the primary circuit and 2 loops of pressure compensator cool-down passive system (PC CDPS), providing the opportunity to stay ahead in coolant pressure reduction in relation to temperature reduction saving the necessary amount for boiling in the active zone. Every heat removing loop from the primary circuit and every PC CDPS loop includes a heat exchanger based on TPT, a two-phase intermediate circuit, and a heat exchanging condenser situated outside of the containment. Cool-down loops pipeline of the primary circuit are connected to ECCS Du 300 pipelines: 2 loops for every of the two upper and bottom ECCS pipelines.

After starting the ARPS R, due to compensator simultaneous cool-down by means of PC CDPS, conditions for an earlier ECCS accumulators connection are formed. During ARPS R thermosiphon afterheat removal, the positive reactivity input is prevented due to the aforementioned. The method obviates the need to increase the RPS control or to install additional systems affecting the reactivity. Therefore ARPS R with its input algorithm has various advantages compared to ARPS SG, especially when thermosiphon heat exchanging equipment is used making emergency heat removal safe and reliable. ARPS R operation does not depend on the main reactor equipment which is steam generators.

The schemes and operation peculiarities of ARPS R and PC CDPS are being analyzed, and performance predictions have been made. Stability and response tests results are presented, besides that, overall accuracy rates of thermosiphon heat exchangers design circuits and the accuracy of obtained results in relation to main undetermined parameters of the model: nodalization specification, correlation parameter used when calculating leakage modes and heat exchange factors.

FA REACTOR THERMOHYDRAULIC SPECIFICATION OF NPP-2006 PROJECT BY APPLYING FULL-SIZE CFD-MODEL.

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FA reactor full-size CFD-model of NPP-2006 project has been developed in STAR-CCM 10.04 software package. The model represents the coolant leak path in FA with no need of porous body proximity.

The model includes input and output flow stabilization, a shank with bottom support grid, fuel beam and FA head with a fragment of protective tube unit bottom plate. Fuel beam consists of 312 fuels, 13 spacer grids, 4 mixing grids, 18 guide channels and one instrument channel. Hexahedral grid model is used with interfaces on separate parts. Grid model overall size by solid and liquid parts is approx. 700 mln control volumes.

Coolant flow is prescribed at the inlet. Heat is supplied via massive fuel column energy release changing by fuel height. At fuel beam sides limiting condition is applied as the sliding wall symmetry. Thermal and physical characteristics of coolant and FA components material depend on the temperature. They include density, viscosity, thermal conductivity, and heat capacity. All calculations are performed by means of k-w SST turbulence model and second-order sample circuit

Due to developed CFD-model has defined the pressure drop coefficients of spacer and mixing grids in fuel beams, and in the whole FA as a Reynolds number range of 50000 to 450000 in the mode of minimum controllable power level. FA life-size mock-up design and experiment data have been compared. FA reactor thermohydraulic specification has been made at the end and at the beginning of the campaign. The results presenting the highest temperature of fuel cladding external surface do not exceed the value allowable for for normal reactor operation in NPP-2006 project.

At later stages FA CFD-model is planned to be improved by means of adding individual power and deposition function throughout the height of every fuel.

Subsection 2.2
DEVELOPMENT OF NEW NUCLEAR POWER UNITS

RESULTS OF DEVELOPMENT OF CONCEPT DESIGNS OF MEDIUM POWER NPP UNITS

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In 2014, NIAEP JSC jointly with NRC Kurchatov Institute, EDO Gidropress JSC, OKBM Afrikantov JSC, and VNIIAES JSC prepared materials of concept designs of a two-unit medium-power NPP with VVER-600 and VBER-600.

The works were performed under the agreement with Rosenergoatom Concern OJSC Design and Engineering Branch based on the Decision of the Science & Technology Council No. 1 Nuclear Power Facilities and Nuclear Power Plants of the SC Rosatom and the Science & Technology Council of Rosenergoatom Concern concerning the Comparative Analysis of Performance Characteristics of two-unit medium-power NPPs and NCPs as of 09.04.2013.

The following was done in the course of the works:

1) Technical and economic requirements for a water-cooled medium-power unit design;
2) Safety concept for a medium-power NPP with VVER-600 and VBER-600 reactors;
3) Optimization of the main process engineering solutions and layout arrangements, general layout plan for the buildings and structures of NPP power units with VVER-600 and VBER-600 reactors;
4) Development of concept designs for a medium-power two-unit NPP with VVER-600 and VBER-600 reactors;
5) Terms of Reference for a medium-power NPP design using the example of a two-unit NPP with VVER-600 and VBER-600 reactors;
6) Comparative analysis of concept designs of a two-unit medium-power NPP;
7) Referencing the concept design of a two-unit NPP with VVER-600 to the conditions of Kola NPP-2.

The basic reference points in the development of the materials of medium-power NPP concept designs were as follows:

1) Net electrical power - at least 600 MW;

2) The period of construction of an NPP power unit from the first concrete to the physical start-up:
    - for the pioneer unit - no more than 48 months;
    - for the commercial unit - no more than 40 months;
3) Reduction in the estimated cost of the construction of the commercial power unit as compared to the VVER-TOI design is no less than by 25%.

Medium-power two-unit NPP concept designs have been developed with due consideration of the requirements of the Government Regulation of the Russian Federation No. 87 "Concerning the Sections of the Design Documentation and Requirements for Their Contents" as of February 16, 2008, with the following having been developed as part of such documents:

- Explanatory notes;
- Specifications for equipment and materials;
- Construction cost calculation;
- 3D arrangements and drawings.

During the works, the criteria of the Technical and economic requirements for a water-cooled medium-power unit design have been met, with the achieved value concerning the estimated cost of construction of a commercial power unit as compared with the VVER-TOI designs being at least 27%.

This indicator once more confirms an important advantage of a medium-power NPP, viz. less expenses for the project implementation and the risk reduction for investors.

Besides, the potential for further improvement of the concept designs of a two-unit medium-power NPP with VVER-600 and VBER-600 reactors, thanks to the combination of innovative technology and the use of well-proven traditional solutions, can be unlocked at the subsequent stages of engineering.

Possible applicable scope of medium-power NPPs:

- for districts where the use of traditional organic energy sources is impossible or hindered, and powerlines are unavailable or impose constraints on the capacity of power generating equipment;
- as the source of uninterruptable power supply for important facilities of the national importance;
- for desalination of sea water, which seems important due to the possibility of a shortage in fresh water.

According to the developed Safety Concepts, national designs of medium-power NPP power units with VVER-600 and VBER-600 are of Generation III+ and have enough competitive strength to timely enter the international market, which requires finding the site for the construction of a pilot NPP.

ABSTRACT OF THE REPORT «WAYS TO IMPROVE MAIN BUILDINGS OF NPPS WITH VVER-TYPE REACTORS»

Speaker O.V. Koltun

It is well-known that Russian Generation II NPP designs implemented currently in Russia and abroad are the evolution of Generation II power units with V-320 reactors. Notably, the main reason for their modernization was the need for making adjustments to the design in line with the increased safety requirements. As a result, mechanical development of new process systems caused significant increase in the design complexity and the share of the equipment not involved directly in power generation.

The dimensioning specifications have also become more complicated since new components and volumes have been added to allocate the equipment. Layout and design solutions for buildings have also been changed due to their more complicated structure and many new details.

All of the above impacts such important parameters as CAPEX and project deadline. It is evident that extension of NPP construction period makes such nuclear power industry projects less attractive for investors and creates risks for project implementation.

All of the above require a thorough analysis of new NPP projects to assess their constructability and capability to adhere to the scheduled pace of construction. The actions aimed at eliminating the drawbacks of the designs in order to ensure competitive performance of NPPs as compared to the traditional power industry shall be developed.

The report discusses the following lines of improvement of main building designs performance indicators:

- options to improve layout arrangements of the general layout facilities;
- changes to space-planning decisions of the reactor building;
- the need for revising strength calculations of pressurized structures and the principles of their reinforcement;

Assessment of the proposed actions is given from the point of view of construction schedule reduction and reduction of CAPEX for NPP construction.

EXPERIMENTAL RESEARCH OF NOVOVORONEZH NPP-2 PASSIVE SAFETY SYSTEMS JOINT ACTION AT "REACTOR-CONTAINMENT" LARGE SCALE TEST FACILITY

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The report contains results of diurnal integral experiments, held in SSC RF IPP at «Reactor-Containment» test facility, erected for research of joint action of passive safety systems within VVER-1200, being erected at the site of Novovoronezh NPP-2 (NV NPP-2).

Construction of equipment within the «Reactor-Containment» test facility provides modeling of interconnected mass-exchanging processes between the reactor and containment in case of a beyond design-basis accident with extensive coolant leak from the main coolant pipeline and operation of passive safety systems – second stage hydroaccumulators and passive system of heat removal from reactor to the air.

Experiments have been held to determine impact (against heat removal efficiency from the reactor via the steam generator (SG) and the passive heat removal system in case of beyond design-basis accidents) of noncondensing gases, generated in the primary circuit, and gases, transferred due to mass-exchanging processes from the containment via the failure cross section.

Modeling of accidents «Full cross-section fracture of Main coolant pipeline upstream and downstream of the reactor» has been held at «Reactor-Containment» test facility in conditions of active safety systems failure.

As a result of beyond design-basis accidents modeling (duration – 24 hours) it has been found, that experiment with modeling of cold pipeline fracture has shown a ~27 % reduction of steam generator model condensing capacity. The experiment with modeling of hot pipeline fracture has shown ~30 % reduction of steam generator model condensing capacity at the end of the experiment compared to capacity at emergence of condensing mode in the SG. Reduction of the steam generator condensing capacity is conditioned by presence of gases in steam, generated in the reactor, and by delivery of steam-air mixture from the containment model to the reactor during mass exchange between the reactor and the containment during 24 hours of the beyond design-basis accident.

Experimental data, obtained at the «Reactor-Containment» test facility, have been used by Atomenergoproekt JSC to justify efficiency of passive safety systems joint action. Besides, the experimental data have been used for additional verification of calculation programs, which perform modelling of interconnected thermohydraulic processes within the reactor and the containment during beyond design-basis accidents with leaks from the reactor.

Quantitative analysis of experimental data concerning condensing capacity of the steam generator chilled by passive heat removal system, shows that joint action of hydroaccumulators-2 and Passive heat removal systems provide conditions to keep coolant level in the reactor, needed for sound chilling of the zone at least for 25 hours in case of continuous upper design temperature of atmospheric air + 38 °C, accepted for the project of NV NPP-2.

NPP construction monitoring in Rosenergoatom Concern OJSC

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In order to provide the personnel involved in NPP construction projects with accurate and timely information that is necessary to identify the risks, make efficient managerial decisions, and enhance planning and implement construction processes, Rosenergoatom Concern OJSC (hereinafter the Concern) has developed and implemented the Automated Construction Monitoring System (hereinafter the ACMS-Uchet) and the Project Portfolio Management Information System (hereinafter the REAC PPMIS).

The ACMS-Uchet consists of two components, viz. a top-level analytical system and a local system of record.

The top-level system (ACMS-Uchet) provides performance of the following functions:

- collection of information from various systems from all the people involved in the implementation of an NPP construction project;
- project analytical reporting in various dimensions, with the option of viewing the data up till the source documents (drill-down technique).

The local system (LCMS) performs the following functions:

- automation of detail documentation (hereinafter the DD) reviewing processes;
- automation of issuance and control over implementation of corrections with regard to the DD;
- automation of the processes for reporting and checking local cost estimates and KS-2 Certificates (interoperability with Atomsmeta PC). Generating KS-3 Reports;
- cost accounting, and control over the estimated limit and use of capital investments;
- control over performance of agreements and equipment supplies;

- identification and forecasting deviations from plans with regard to deadlines and costs;
- in case of joint use with the REAC PPMIS, ensuring compliance with the construction work schedule.

Integration of the LCMS with the accounting and finance system allows more accurate tracking of the use of capital investments and company-wide cost management.

Project analytics would not be complete without deadline management database. This function has been implemented in the REAC PPMIS. The system help develop and manage the 3rd phase NPP construction schedule. The schedule is generated in compliance with the requirements of the method that has been used in the construction project for Rostov NPP Unit 3. All the works included in the schedule are connected to the source records in the LCMS, which allows fast updating of the project implementation schedule based on the data of the source accounting documents and tracking the key process events achievement.

The availability of detailed information on the scheduled works allows generating immediate work-in-progress reporting directly within the system, ensures topical planning and present necessary source data for external systems when generating top-level reports (long-term investment programs, summary investment programs, reports for the Ministry of Energy, etc.).

It is planned to develop systems of record for capital projects in the branches with the NIAEP JSC system of records within the framework of establishing a common informational space of the Customer and the General Contractor, in order to improve the efficiency of getting approvals for capital project documentation, and to implement forecast models and simulation modeling (what if) in an analytical system with the purpose of efficient management of the costs and deadlines of construction projects.

The process design solutions used in projects will help use the implemented tools not only in the implementation of NPP construction projects, but also in NPP overhaul projects.

ISSUES OF STRENGTH AND SERVICE LIFE OF VVER INTERNALS IN LONG-TERM OPERATION

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One of the most critical safety-related components of VVER-1000-type reactors are the in-vessel internals which form the reactor core, organize the coolant flow in the reactor, hold and protect the controls and in-core monitoring devices.

The main component of the internals that conditions and limits the service life of VVER-1000 reactors with the extended service life to 60 years, is the core baffle that operates in the conditions of high gradients of neutron exposure and temperatures. According to EDO GIDROPRESS JSC's estimates, during 60 years of operation, the maximum damaging dose of neutrons for the baffle of VVER-1000 reactors is 100÷120 dpa, and for the baffle of VVER-TOI reactors - 160 dpa.

The calculations made by EDO GIDROPRESS JSC according to RD EO 1.1.2.99.0944-2013 Calculation Procedure for Strength and Remaining Service Life of VVER-1000 Internals at Service Life Extension for 60 Years or More, show that at the conservative forecast of radiation-induced swelling, the service life of VVER-1000 baffle is limited by the impact of radiation-induced swelling and radiation-induced creep on its deformation, and is less than 60 years of operation.

One of the most realistic ways to guarantee the extended service life of 60+ years is the development of new radiation-resistant steel for the internals.

Manufacturing and using such new radiation-resistant steel for the internals of new VVER-type reactors will guarantee the 60+ years design service life of the baffles of new power units with VVER-type reactors, and make Russian NPPs more commercially viable at the global scale.

SOME PROSPECTS OF THE NUCLEAR POWER INDUSTRY DEVELOPMENT BASED ON THE USE OF WATER-COOLED REACTORS

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Today, the main type of reactor units in Russia and abroad are power units with VVER-type reactors (PWR) with two circuits. From among about 500 power units in the world, around 300 are of this type. Rosatom's success is evidenced by many construction projects for power units with VVER-1000 (1200) both in Russia and abroad. However, the prospects of the nuclear power industry based on water-cooled reactors shall be considered more broadly.

The author thinks that complete rejection of channel-type reactor development is unjustified. After Chernobyl accident, RBMK reactors have operated for more than 25 years at all Russian NPPs. MKER design does not have the drawbacks characteristic of RBMK reactors related to nuclear safety. There are studies that demonstrate that it is possible to construct channel-type reactors with ordinary water acting as the moderator.

One important task is designing middle power reactors. Unfortunately, VVER-640 design was never put into practice. Due to the fact life circumstances require participation of power units in load schedules it is reasonable to use the combination of the negative temperature coefficient of reactivity of the VVER reactor core with universal-pressure generators which ensures high flexibility of the power facilities. Such combination has been well proved in marine NPPs. In this regard, VVER-600 (OKBM Afrikantov) with a modular arrangement of reactor equipment is more preferable than VVER-600 (EDO Gidropress) based on conservative technical solutions.

The one-circuit reactor designs are not used in Russia at all. The positive experience in the use of pressure vessel boiling water reactors (there are about 100 such power units around the world) supported by the Russian experience in operation of VK-50 reactor at NIIAR and VAU-6S at Aleksandrov Research Institute of Technology (NITI) has, nevertheless, not been taken into account. This line of development, in particular, could be used as replacement power for Bilibino Nuclear Co-Generation Plant (NCP), where EGP-6 power units have depleted their service life. The Fukushima Daiichi NPP accident has nothing to do with the type of the reactors used and in no way endangers the reputation of VVERK (BWR). The proposal to meet the needs of Chukotka by means of a floating NPP (KLT-40) seems not adequate enough.

The most important issue of putting the supercritical-pressure (SCP) water coolant in reactors in operation is not supported by the required scope of research and development work, where the physicists should have the leading role in creating a safe reactor with no prompt criticality. The authors have experience in participation in the testing and development of a pressure-vessel reactor VAU-6S, which is why they can confirm that VVER and SCD are incompatible. The negative "void reactivity effect" does not guarantee safety, since in the wide range of modes, there may be increase in water content in the reactor core followed by unacceptable surge in positive reactivity. As early as in the 60s, an outstanding physicist S.M. Feinberg at the Kurchatov Nuclear Institute И.В. focused on neutrons of intermediate energy (VPN-705) rather than on slow neutrons in his design of a ship reactor.

The issue can be positively solved only through clear, highly professional and centralized organization of the whole work package, both in terms of scientific research and design work.

MATHEMATICAL MODELING OF SEVERE ACCIDENTS IN FAST NEUTRONS REACTORS

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A comprehensive mathematical model for the design analysis of severe beyond design basis accidents in sodium-cooled fast neutrons reactors has been developed for the first time.

The domain of computation is a manifold one. Mathematical modeling of subdomains as porous bodies has been conducted with the use of the principles of conservation of mass, impulse and energy described as equations of mathematical physics.

The task related to the formation of a heat-generating layer at the lower end shield has been solved. Heat-generating layer zones have been modeled. In particular, the melting of steel particles and, later, fuel particles has been taken into account through modeling of the heat sink capability in the heat-generating layer. A formula to determine the melting effective heat in case of in-reactor structures rupture has been proposed.

The developed mathematical model has been implemented in the BRUT code.

The results of the calculations made by means of the BRUT code have been compared to the results of the reactor experiments. The calculation results appeared to be largely in line with the experimental data.

The calculation block of the BRUT code natural convection has been verified. The calculation results appeared to be largely in line with the experimental data.

The functioning of the calculation block for in-reactor structures rupture has been checked.

In order to check the correctness of the calculations for the temperature pattern in the heat-generating layer, there has been used M.V. Kashcheev's accurate solution of the task of unsteady-state conduction of a limited cylinder of R radius and *l* length with uninterrupted heat sources placed in the media with time-variant temperature, with boundary conditions of the third kind at three boundaries.

By means of the BRUT code, it has been shown that in case of a severe accident associated with major plant blackout and failure of all reactivity controls in high-power BN-type reactors, partially destroyed reactor core is not ruptured.

Also, the calculations of an accident involving destruction of 18 fuel assemblies of the first and second rows of the small-power BN-type reactor core have been made by means of the BRUT code. The destruction of FAs results in the formation of a heat generating layer located at the lower end shield. First, the melting of the lower end shield starts and then stops.

The BRUT code has been used to calculate a hypothesized severe accident involving full destruction of fuel assemblies across the small-power BN-type reactor core. According to the results, the corium falls into the containment area. The heat generating layer cools down in the containment area. The coating and the steel under the coating do not melt in the containment area.

In order to rapidly assess the parameters and, first of all, the time of the structures rupture, a mathematical model that solves the task in one-dimensional approximation has been developed. Based on the model, the BRUT-O code has been created. The results show that the time the corium needs to reach the pressure chamber plate calculated by means of the BRUT-O code, is by 10% less than the time calculated by means of the BRUT code.

OPTIMIZATION OF LOCAL HEAT TREATMENT OF STEAM GENERATOR CLOSING CIRCUMFERENTIAL WELD

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The report describes the way of ensuring high-quality local heat treatment of closing circumferential welds PGV-1000M and PGV-1000MKP steam generators due to uniform heating of the whole area of the weld and the controlled heating zone of complex surfaces of steam generators, and significant cutting of the time for the heat treatment operation and the subsequent operations when using flexible ceramic heating pads (mats) and a special testing facility developed by the company.

Key words: local heat treatment, testing facility, flexible ceramic heating pads (mats), closing weld, steam generator.

IMPROVEMENT OF BEARING CAPACITY OF STRUCTURAL CONCRETE COMPONENTS BY MEANS OF ANISOGRID COMPOSITE STRUCTURES

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Traditionally used schemes for reinforcement of concretes with steel bars, which prevent fracture of concrete under tension and bending, have practically exhausted their ability for improvement of bearing capacity of built up structural components. To a large extent, this is the case of essential structures erection at nuclear power plants – increase of concrete structures cross-sectional area is limited by incremental mass of concrete, which may become a critical factor for a construction failure.

Issues for development of new structural elements are being considered for building of various essential high-strength structures. Concrete structures, reinforced with anisogrid lattice composite structures (developed by TSNIISM JSC as applied to rocket, space and aviation technology and at present being used commercially for manufacturing of spacecraft and launch vehicle hulls), are offered as main structural elements. Reinforcement of concretes with anisogrid composite framings will eventually allow to either considerably improve their bearing capacity without cross-section increase, or considerably reduce mass and dimensions of such structures.

Anisogrid composite lattice structures, being a complex of spiral, ring and, if needed, longitudinal ribs from monodirectional composite material are made by method of continuous automated winding, which allows for rigid and durable integral carcass systems. Fiberglass, fiberbasalt or carbon rod-shaped and shell anisogrid structures are characterized by low weight. In comparison to traditional steel reinforcement carcass, such structures are characterized with considerable bearing capacity and increased rustproofness.

To confirm functionality of the concept offered, we provide results of experimental research and computed finite-element modeling of composite concrete structures, reinforced with anisogrid lattice structures. Composite concrete elements test results being compared with test results for reinforced concrete elements reinforced with conventional steel reinforcement bars. The research reveals increase in bearing capacity for composite concrete structural elements in comparison to concrete elements reinforced with steel bars at comparable rigidity of compared structures.

Potential for doubling or tripling bearing capacity of concrete structural elements is shown by application of anisogrid lattice composite structures used as reinforcement elements in comparison to conventional schemes of reinforcement by means of metal carcass.

Strength analysis and computational-experimental verification of reinforced concrete computational models beyond elastic deformations used in BREST OD-300 reactor and safety-related components.

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When analyzing the strength and seismic stability of structures, such as the BREST-OD-300 reactor block (RB) hull, and when calculating safety-related components for extreme conditions, there is a need for consideration of non-linear mechanical properties of concrete components of the structure: non-linearity of the stress-deformation diagram; the phenomenon of cracking and destruction caused both by compressive and pulling stresses; creep and shrinkage; radiation effects; non-linear law of grip of concrete and steel; and the dependence of the said properties and phenomena on temperature.

In order to verify and finely adjust the parameters of computational mathematical models of concrete deformations and strength created by means of the Abaqus software, the authors, in cooperation with KBSM JSC and ATOMENERGOPROJECT OJSC, have experimentally performed and, subsequently, solved and verified a number of full-scale tasks to analyze the concrete and reinforced concrete destruction caused by thermal and extreme loads. Based on the full-scale testing and the created algorithm for determining the material deformation diagram for the types of concrete in question, computational relationship between the stresses and deformations under compression and pulling have been obtained. The research included various types of destruction of reinforced concrete beams caused by dynamic and static loading.

With the experimental data and based on the published papers on destruction of reinforced concrete samples, the concrete mathematical models have been verified. In particular, for the types of concrete used for the RB hull, parameter values in the non-linear concrete model «Concrete Damage Plasticity» by the Abaqus software have been determined. The said concrete model was used to calculate the strength of the RB hull during heating and under seismic impact in a non-linear arrangement. The calculations provided a detailed stress-stress state of the mass concrete and steelwork of the RB hull. The research included a study on the formation and growth of cracks in the RB hull mass concrete. Based on the Concrete Damage Plasticity model, the impact of a plane on the reactor building has been simulated. The response spectrum in the distinguished mounting points of the plant critical equipment of the reactor building under a heavy airplane for various foundations and the ground damping coefficient has been determined.

The scheduled works for 2016 include further verification of the results and experimental work for the purpose of the verification report for the use of this model for computational-experimental works, including under temperature exposure up to 600 °C.

Research on thermohydraulic characteristics of high-voltage and high-temperature transformers with the use of a full-scale CFD model of a winding segment

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For NPP equipment, there are special requirements for maintaining autonomous functioning in emergencies. That is why the decision to cool high-voltage transformers winding only through the natural circulation loop (NCL) of a dielectirc liquid, which simultaneously functions as the coolant, is justified. The winding design of existing transformers is not efficient enough in cooling down the winding wire. This results in electric current density reduction; moreover, a transformer becomes unserviceable at all in a hot climate.

An efficient tool for the analysis of the flow-dynamics and heat-exchange processes in the winding is the computational fluid dynamics, or the CFD method, which is based on the numerical solution of Navier-Stokes equations. For the thermotechnical calculations, the winding design that provides for the maximum heat removal from the surface of a single wire has been used.

The report presents the results of the calculations by means of the CFD model of a segment of the primary winding that consists of round wires, and a cooling unit at the transformer wall. The CFD gridded model made up about 30 mln. control volumes. A liquid with the thermal and physical characteristics that match the typical transformer oil parameters was chosen as the coolant. The established boundary conditions were the heat release in winding wires and the temperature of the cold wall of the transformer within the range from 20 °C to 100 °C.

The calculations showed reliable cooling of the conical coil at the coolant temperature up to 100 °C, which corresponds to the temperature of the air around the transformer up to 80 °C. The maximum value of the dielectric liquid heatup was about 20 °C. Notably, the coolant heatup value drops with the cold wall temperature increase. This will contribute to better thermomechanical and strain-stress state of the transformer's primary winding.

Such methods will be used to create high-voltage high-temperature transformers at NPPs operating in a hot climate without forced cooling.

MANUFACTURING, INSTALLATION AND OPERATION OF POLAR CRANES

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Polar crane is one of the main technological devices for reactor compartment maintenance during main lift-and-carry operations at the stage of installation and repairs of power units, and during operation. It may handle fresh and spent nuclear fuel. Such equipment was introduced in seventies of the previous century. Although until the present day many issues, connected with manufacturing, installation and operation of such equipment, haven't been resolved.

One of the problems is accomplishment of the crane running wheels geometry. As of today, running wheels are mostly conical. Although, it is not wheels geometry which is critical for the crane circular motion. The most important characteristic here is mutual disposition of wheels within minor balance beams, then disposition of the minor balance beams themselves, then main balance beams and, finally – geometry of wheels. Geometric shape of wheels has impact on sliding conditions within wheel and rail contact area.

Based on experience of polar cranes installation, performed at Rostov, Leningrad NPPs it was offered that in cases of bench erection at a preinstallation site it is necessary to control standard geometrical parameters, e. g., bridge diagonals by KK and FF points, camber of longitudinal beams, etc. It is also necessary to control geometry of chassis with measuring sliding parameters (tangential and longitudinal), slip coefficients (cross-sectional) and directional stability parameters. In this context, it should be emphasized that diagnostics of crane chassis technical condition should take place at the factory of origin (otherwise, any revealed deformations or declines from nominal parameter will not be amended). Manufacturer «Tyazhmash» in Syzran responds with full understanding of this problem. Besides that, works, connected with control of crane rail sections, may be performed at the site of installation contractor, since its geometric parameters in AES-2006 project shall be provided not only by sections geometry, but also by conditions of their laying according to radial line. In previous projects of NPPs with VVER-1000 reactor, with even number of sections, should this happen, project circle of railways could not be provided without standard geometry. And project 2006 had an uneven number of sections. Consequently, center of joint falls to center of the opposite section. In such a way, relocation of joint in any direction may correct railway geometry parameters up the conditions needed.

Operation of this equipment shall be supported with monitoring and evaluation of polar crane influence on hermetic containment. Monitoring shall be started from the date of the crane installation on the railway. In this case, special attention shall be given to external side of the prestressed containment at the level of crane beams consoles of railway during completion of containment up until its testing.

THESES OF THE REPORT MAIN COMMISSIONING WORKS PERFORMED AT BN-800 REACTOR OF POWER UNIT 4 OF BELOYARSK NPP

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Commissioning of BN-800 Reactor at Power Unit 4 of Beloyarsk NPP at the 'idle' stage – prior to start of its own power generation, was carried out by a number of enterprises as per mutually agree work program. JSC "Afrikantov OKBM", being the Chief Engineer and Single-Source Manufacturer of reactor equipment, provided development of the particular specialized commissioning programs, adjustment supervision, author's supervision and setup of particular equipment of sodium process part of reactor compartment.

During preparation for commissioning the following programs were developed:

- Commissioning program of Power Unit 4 of Beloyarsk NPP
- Stage programs of sodium acceptance at NPP, gas heating up, filling up the reactor and secondary circuit with sodium, carrying out commissioning at the new fuel storage, at the systems and equipment of turbine hall, physical startup, and pilot operation.

The programs considered differences of the equipment, reactor core, processes during commissioning of BN-800 Reactor from VVER-type Reactor (liquid sodium coolant, structure of reactor core, physical peculiarities of startup, specifics of the equipment of the reactor core components handling system, etc.).

The following commissioning works were specific for BN-type Reactor during commissioning:

1) Gas heating up of the reactor.
2) Filling up the reactor and the secondary circuit loops with sodium.
3) Setup of the pumps control systems of the primary and secondary circuits.
4) Setup of the equipment and control system of transport process train.
5) Setup of the Reactor emergency cooldown system through air heat exchangers.
6) Adjustment of the sodium technology instruments.

On November 2015 the works on study of the reactor core characteristics and commissioning at 'idle' stage were completed. The Power Unit was commissioned (turbine generator was energized at the reactor power equal to 25% of the rated one) on December 10, 2015.

THESES TO REPORT ON THE SUBJECT: «COMISSIONING OF NA INSTRUMENTATION ON BELOYARSK NPP»

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The Report presents information on Na system instruments and controls (Na Instrumentation) of Power Unit 4 of Beloyarsk NPP.

Unique character of Na Instrumentation consists in sensors conformity with severe conditions of measuring the parameters of high temperature liquid Na of BN-800 Reactor.

Design and operation principle of the sensors is described to monitor the following process parameters:

1. Liquid Na pressure
2. Liquid Na flow rate
3. Liquid Na level
4. Availability and leakage of liquid Na
5. Gas presence in liquid Na flow.

Short description of Na Instrumentation Secondary Converter Package (NISCP) combining secondary converters of Na Instrumentation is given.

The Report describes specific problems, occurred during adjustment of this equipment, and presents their analysis, depicts disadvantages and comments to Na Instrumentation, identified during commissioning, as well as their troubleshooting methods.

Conclusion summarizes the operations performed on putting into operation NA Instrumentation of Power Unit 4 of Beloyarsk NPP.

ACCOUNTING SYSTEM OF COMMISSIONING AND REPORTING DOCUMENTATION DURING COMMISSIONING OF POWER UNIT – ADVANTAGES AND DISADVANTAGES

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In accordance with clause 8.8: STO 1.1.1.01.0678-2007 "Principal rules for assurance of nuclear power plants operation":

"Every NPP shall be provided with ... commissioning documentation.

The documents specified above shall be registered and stored as established at NPP".

In accordance with STO 1.1.1.01.003.0667-2011 "Classification of technical documentation of Rosenergoatom Concern OJSC, Section 7, Table 1, clause 1.7.2 "Commissioning documentation". Commissioning documentation includes the following, in particular:

- programs, certificates, minutes, reports (totally more than 40 items).

In accordance with STO 1.1.1.03.003.0881-2012 "Commissioning of NPP units. Terms and definitions", clause 4.81 reporting documentation:

Documents being formalized during construction and installation works and commissioning for submission to the builder (minutes, certificates, and reports). Thus, RD is a part of commissioning documents collection.

STO 1.1.1.03.003.0907-2012 (with rev. 1) "Commissioning of NPP units. Reporting documentation" is a document which governs the work with reporting documentation. Work with RD in its sense assumes the work with original documents only.

In order to provide automated accounting of RD being created during commissioning of the unit, database of reporting documentation was generated by means of Oracle Database Management System with the following capabilities:

- Automation of documents control and accounting processes.
- Processing, integration, and storage of originals.
- Storage of electronic documents and online access to them.
- Generation of reporting forms and data export.

Availability of flexible connections, modular architecture, and other specifics of base construction defines its good potential with regard to further increasing of capabilities.

The developed database was implemented at Beloyarsk NPP as a pilot project.

During base implementation the documents, governing acceptance procedure for documents accounting and registration, storage of originals and introduction of amendments to them, were developed. RD blanks with formalized attributes, which were required for automatic generation of the document registration number, were developed.

Base of the registered documents was the main tool, confirming performance of particular commissioning works for their payment, as well as activated and facilitated the work of many committees (Rosatom State Corporation, Rosenergoatom Concern, Rostechnadzor of the Russian Federation), which carried out that or another supervisory roles.

Conclusions:

1) Task of creation of efficient reporting documentation database was performed within a short time.
2) The required Organizational and Executive Documentation (OED) was developed.

3) Functional group of Industry Diagnostic System (IDS) and required infrastructure were created.
4) Technology of IDS accounting and processing of original documents for temporary archive storage was developed.
5) Work of the interested users in IDS DB was organized. Today about 300 users are registered. All of them can work in DB simultaneously.
6) Disadvantages in IDS preparation and cooperation with management and executives were defined.
7) Discrepancies in KREA STO with regard to RD nomenclature, as well as drawbacks of the reporting documentation terminology, were identified.
8) Created DB allows to organize work with RD by several sites at once with control of design flow and commissioning documentation approval.
9) Created database has a great potential for its expansion and creation of full-featured commissioning support instrument on its base at the sites of new NPP units by adding other modules, such as: coordination plan, estimates, process systems (with pump list, motor-operated valves, measurement channels, etc.).
10) STO 1.1.1.03.003.0907-2012 "Commissioning of NPP units. Reporting documentation" contains a great number of templates. However, the document does not record automated data processing, that is, in our opinion, an essential disadvantage. In general all the analyzed STOs on commissioning works have similar disadvantage.
11) Full range of the unified templates of RD documents, meeting the requirements of regulatory base of Rosenergoatom Concern OJSC with regard to commissioning of NPP power units, can only be developed after unification of registration number structure and based on the platform of the Unified Database Management System.
12) In spite of NP-001-15 requirements of permanent storage of commissioning documentation only for safety classes 1 and 2, such a storage, in our opinion, is reasonable to be arranged for a whole collection of reporting documentation. Reporting documentation shall also include commissioning documentation in accordance with which that or another commissioning works were carried out.

TECHNICAL MANAGEMENT OF COMMISSIONING BY VALVES. CHANGED APPROACH ON POWER UNIT 3 OF ROSTOV NPP.

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Brief introduction

New approach to carry out commissioning of motor-operated valves with respect to arrangement of electrical installation and commissioning, cooperation with the Customer, assembly organizations and design unit, handing over valves to the Customer, was developed and implemented at Power Unit 3 of Rostov NPP. The new approach is based on: one technical leader (on electrified valves) for process part, electrical part, and APCS, formation of flexible commissioning teams (valves adjustment together with diagnostics), arrangement of cooperation between assembly organizations, electrical installation organizations, commissioning organizations at all stages of commissioning works, development and management of actual base with comprehensive information on every unit of valves with daily updating. Such an approach let us provide the following: control of equipment installation, detection, and elimination of impacts and defects at an early stage, technical management of electrical installation works (determination of priority areas, daily targets, online solution of technical and organizational issues), development of stepwise procedure of commissioning and handing over the valves to the Customer. Flexible teams formation (valves adjustment together with diagnostics) let us decrease the number of team personnel and exclude failure of the valves during commissioning, reduce time for commissioning, exclude additional testing of the valves to carry out diagnostics and provide high quality of works (zero defects during commissioning).

Work objectives and tasks

An objective of technical management of commissioning of motor-operated valves at Power Unit 3 was to provide preparedness of process systems to due time and inadmissibility of testing disruption due to unpreparedness of motor-operated valves, ensuring high quality of commissioning of motor-operated valves and zero defects at adjustment stage.

To achieve the stated objective, the following particular tasks were specified:

1. Monitoring of valves installation (detection of impacts, irregularities at an early stage (before electric wiring), involvement into settlement of design impacts (operational) and monitoring of irregularities elimination). Determination of installation sequence.

2. Monitoring of the valves maintenance platforms installation (detection of impacts, irregularities at an early stage (before electric wiring), involvement into settlement of design impacts (operational) and monitoring of irregularities elimination).

3. Development and implementation of procedure of electric wiring, adjustment, and handing over the valves to the Customer prior to start the works (submission of the valves after electric wiring and defect elimination, form and contents of work completion certificates, valves acceptance procedure in workshops).

4. Planning and detection of motor-operated valves preparedness sequence to preparedness of process systems. There are no outages because of motor-operated valves unpreparedness to carry out testing of process systems (management of installation and commissioning).

5. Optimization of personnel number (formation of flexible commissioning teams together with diagnostics) with increase of work performance and quality.

6. Development and maintenance of actual base with comprehensive information on every unit of valves with daily updating (nameplate data (of the valves, electric drive, electric motor), data on MMZ and MMO adjustment, numbers of designs (engineering design, installation diagrams, schematic diagrams), rooms for motor-operated valves installation, diagnostic data measured (travel time, currents, voltage, etc.), comprehensive information on cable connections from drive to control post (including assemblies and cabinets of Typical Software and Hardware Facilities (TSHF), etc.), numbers of certificates and dates of testing motor-operated valves, numbers of motor-operated valves adjustment and diagnostics, etc.).

Results

Applying the new approach let us significantly increase labor efficiency: achieved are the volumes of valves testing equal to from 112 to 130 units a week (at the similar Units of RoNPP 2 – at most 30-40 units a week, KNPP 4 – at most 25-40 units a week).

Implementation of new arrangement of commissioning of motor-operated valves, including cooperation with the Customer, assembly and engineering organizations, allowed to increase labor efficiency by 70% as per our estimation, make arrangements on elimination of the process equipment defects, excluded failure of the valves during commissioning, as well as let us create conditions for premature startup of Power Unit 3 of RoNPP. At that we managed to hand over the Power Unit for operation with 100% preparedness of the valves that we failed to achieve at the previously commissioned Unit 2 of Rostov NPP and Unit 4 of Kalinin NPP.

The approach, implemented at Power Unit 3 of RoNPP, is offered to be applied for the subsequent power units under construction that will let us significantly reduce time and improve quality of commissioning of motor-operated valves.

Perspectives of commissioning works optimization on account of application of "Virtual NPP Power Unit" Software and Hardware Complex (SHC)

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"Virtual NPP Power Unit" Software and Hardware Complex (VPU SHC) is a set of software tools and accounting codes for modeling of thermohydraulic, neural physical, electric power, and other processes at NPP power unit, initially of VVER-type. Integrated calculation model, built on VPU SHC tools for particular power unit, is called “Virtual NPP Power Unit” (VPU). Application of VPU SHC allows to provide high scientific and technical level of documentation, its compliance with European requirements to NPP projects (EUR Rev.D), Customer requirements and IAEA recommendations. To this end the Complex allows to perform the following tasks at all the stages of NPP power unit lifecycle from designing to decommissioning:

- Integrated modeling of startup, transient, and stationary operation modes of power unit in a single scale
- Modeling of complex end-to-end scenarios of accidents development with a view to determine measures for their prevention and localization, including performance of accounting for safety analysis and justification
- Design-basis justification of commissioning works plan (procedure)
- Verification of operational documentation (operating instructions, test programs, etc.)
- Check of operatorship on the virtual main control room
- Check of technical solutions during replacement of equipment or planning of works on updating
- And other tasks.

In 2012 ATOMPROEKT successfully conducted acceptance tests of pilot version of virtual power unit of VVER-type NPP (AES-2006 design). On completion of testing ATOMPROEKT accepted VPU SHC and AES-2006 VPU into service. Today the Complex is actively reprocessed and used to check design solutions, as well the works on VPU SHC integration into the process of commissioning works with view to reduce the time and cost of NPP commissioning, are carried out. Models for VPU are being developed together with Atomtechenergo JSC.

Modeling of different modes of power unit operation during NPP commissioning allows to:

- Develop and debug various solutions on thermotechnical modes, electrical engineering solutions, and automation of NPP management
- Evaluate accuracy of operation modes, design algorithms actions
- Facilitate balancing of particular systems and complex testing (regulators, valves, washers, etc.)
- Carry out comparative analysis of calculations and tests to determine deviations and generate offers on improvement of control modes and algorithms
- Try out accuracy of personnel operations specified in test programs
- Improve quality of commissioning and operational documentation being developed (including such important documents as Emergency Operating Procedures, Beyond Design Basis Accidents Management Guideline,

Systems and Equipment Testing Program), try them out on the models prior to start commissioning and carry out verification of the specified documentation at the totally new level

- Train graduates and provide superior training of personnel for newly-built NPPs
- Determine efficiency of updating project or correction of solutions and evaluate effects after updating or replacement of the equipment
- Justify taking new progressive and innovation solutions on commissioning of power unit.

Thus we expect the following effects after acceptance of VPU SHC during preparation and performance of commissioning works:

- Improvement of quality of commissioning documentation and operating instructions, which means the quality and efficiency of commissioning works performance
- Reduction of time for making decisions, time of preparation at real power unit and number of tests conducted during NPP commissioning, as well as the number of failed and repeated tests
- Commissioning works optimization at all the subsequent units of series (after trying out at the main one), as well as reduction of number of test (loading) cycles and risk of damage of expensive equipment
- And as the result, reduction of time for commissioning of power units.

EXPERIENCE OF DEVELOPMENT OF STAGE PROGRAMS ON COMMISSIONING OF UNIT 6 OF NVNPP WITH REGARD TO THE LISTS OF NUCLEAR HAZARDOUS OPERATIONS

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There is a list of nuclear hazardous operations, which is specified in cl. 4.16 of Nuclear Safety Rules (PBYa) of NPP Reactor "***Operator shall develop the list of nuclear hazardous operations of NPP Unit based on design documentation, design list of nuclear hazardous operations, and its operation experience***", provided for majority of operating and administrative NPP personnel, inspectors of Rostechnadzor at NPP, both in central office and in local branches. Such lists are developed, approved, and applied at every NPP.

In virtue of specifics of their work, operating personnel of operative NPPs, and officers, monitoring their work, do not consider clause 5.2.2 of OPB-88/97 (ref. to Section 5.2 "Commissioning"), where the 4th abstract presents the following: "***The documents, governing commissioning, physical and power startups (PhS and PS), as well as pilot operation (PO), shall contain the list of nuclear hazardous operations…***".

“Construction” lifecycle stage at its final phase “Commissioning” is crossed over with “Operation” lifecycle stage.

It means that during approximately (in fact more than) 270 days, when two lifecycle stages go on simultaneously at the Unit, two lists of nuclear hazardous operations are effective. One is “operational” based on cl. 4.16 of Nuclear Safety Rules of NPP Reactor, the other one is “commissioning” based on cl. 5.2.2 of General Safety Regulations (OPB).

If nuclear hazardous operation is included into one of stage programs (PhS, PS, PO), that is, as a rule, a testing, it shall be specified in commissioning list (lists).

In case nuclear hazardous operation (NHO) is required based on operating documentation: it is required by Process Regulation, Maintenance Regulation, Preventive Maintenance or Unplanned Repair Schedule, that is, as a rule, repair or maintenance operations, it shall be specified in operational list.

The Report specifies interaction of operational and commissioning lists of NHOs, the stages which the development of commissioning list of NHOs passes, the questions on commissioning list which were required to be solved during development and expertise of stage programs of commissioning Unit 6 of NVNPP. Specific examples of problems solution are given.

Section 3

NUCLEAR POWER ECONOMICS

COMPETITIVENESS OF NUCLEAR POWER SOURCES, ESPECIALLY IN CONDITIONS OF FAR NORTH OF RUSSIA LOW POWER AREAS

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DPP of the same electrical power level is defined as the main competitor of low power nuclear power sources under conditions of Far North Russia regions.

Effectiveness variant evaluation of nuclear power source of the specified electrical capacity of 100 kW within the capital costs range from 100 to 900 mln. rub. is carried out for comparison in terms of competitiveness of the same capacity DPP.

Limits the competitiveness of capital expenditure and specified capacity are determined on the basis of variant evaluation of the nuclear power source project effectiveness and the information on DPP electricity tariffs in Kamchatka, the Far East and the Republic of Sakha (Yakutia).

It is demonstrated, that low power nuclear power source can compete with DPP in the Far North:

- as a technological power source, if the cost price of the produced electricity is lower than the cost of DPP-produced electricity;
- as a commercial power source, if the selling tariff for produced electricity is lower than the electricity selling tariff tariff of DPP.

It is also demonstrated, that a nuclear power source with the specified electrical power of 100 kW can compete with DPP as a technological power source in the territories of the Russian Far North, where the cost of electricity produced by DPP is not less than 50 rubles for Kwh, the capital costs for the construction of a nuclear power source do not exceed app. 340 mln. rub., and the annual volume of operating costs, including staff costs, but without the fuel costs, is about 25.8 million. rubles, or app. 56% of the cost.

The accuracy the results obtained depends on a number of parameters, such as input data for assessing the effectiveness of the project, but is mainly determined by the accuracy of information about the cost and the selling tariff for electricity, produced by DPP in these areas.

The lower limit of nuclear power sources commercial competitiveness in terms of electric power against the specific capital costs (SCC) is determined for DPP selling tariffs of 30 and 60 rubles for Kwh on the basis of the above-mentioned variant evaluation. The boundaries of the specified limits in terms of the specified electrical capacity is described by the following equations:

$$N_{30} = 0.0143 \text{ x } SCC + 147.06$$
$$N_{60} = 0.0145 \text{ x } SCC + 120.5$$

It is also demonstrated, that the less is the selling tariff for electricity produced by DPP in the territory of the Far North of Russia, the greater is the competitive electric power of nuclear power source.

Staff salary is the largest item of expenditures in the expense amount for both competing power sources electricity production. From the perspective of staff cost reductions, it seems urgent to develop single control center monitoring and management systems for a group of low power nuclear power sources.

Another characteristic feature of low power nuclear power sources is a higher share of allocations to reserves in the overall cost. This implies the need for revision of these regulations in relation to low power nuclear power sources.

FORMATION AND CALCULATION OF COSTS ASSOCIATED WITH LIQUID RADIOACTIVE WASTE TREATMENT IN THE BRANCH OF CONCERN ROSENERGOATOM OJSC "KOLA NUCLEAR POWER PLANT"

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1. The production of radioactive waste (RW), as an inevitable component of the process of electricity production in nuclear power plants.

2. Legal aspects of the financing of radioactive waste management activities.

- The division by the right of ownership;
- Financial maintenance of radioactive waste management activities.

3. Liquid radioactive waste (LRW) treatment at Kola NPP.

- The unique Russian product - a complex for processing liquid radioactive waste (LRW CP);
- LRW CP structure;
- LRW CP technostructure;
- LRW CP performance;
- The average annual percentage of LRW under the federal ownership, processed by LRW CP.

4. Features of formation and calculation of LRW processing costs at Kola NPP.

- The list of items of expenditure;
- Features of cost allocation, in case of impossibility of direct attribution;
- Distribution of general production costs;
- LRW processing costs formation by property type.

5. The structure of LRW CP costs at Kola NPP.

- Processing costs of LRW under the ownership of Rosenergoatom Concern OJSC;
- Processing costs of LRW under the federal ownership;

6. Comparative characteristics of evaporator sludge processing methods:

- The traditional approach;
- Kola NPP approach.

For both the development of nuclear power, and for the proofness and security of citizens, as well as for the protection of the environment, the approach of the Kola NPP to the process of LRW processing, has, in the long run, a distinct advantage of reducing the volume of radioactive waste by more than 100 times and, consequently, lowering the costs of the subsequent storage and final disposal.

VALUATION OF NON-STANDARD EQUIPMENT FOR THE FACILITIES OF NUCLEAR POWER AT THE PRESENT STAGE

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Determining the investment in the construction of new nuclear power objects brings up the question of non-standard equipment valuation. Nuclear power facilities at the design stage are characterized by novelty and lack of manufacturing experience. The uniqueness of the designed facilities makes it impossible to use the estimate price books. Determining the cost of such equipment requires different approach.

Methods of assessing the cost of future products, specified in literary sources, can be divided into two main groups: the methods of assessment by peers and cost estimation method. Parametric methods and techniques of corrections can be associated as auxiliary methods.

Methods and approaches for determining the value of the designed facility shall vary at different stages of work, allowing to obtain a more accurate appraised value on approaching the final stage. It is necessary to take into account the features of any given method application according to the stage of design work.

The specific features of the application of specific weights method, specific indicators method, scaling method and cost method are reviewed along with the provision of practical examples.

Indicators, used within a basically-index method, are compared to each other, and their specifity is reflected.

During the production of the developed products the effect of seriation can be observed. The cost price of the first samples include the costs of production development. Commercially produced kits appear to be much cheaper with the cost reduction of 30% and lower.

To assess the accuracy of economic evaluations it is proposed to use the experience of the International Development Association of the value engineering.

Valuation of the developed equipment for nuclear power facilities at different stages of the project should be done by several methods, taking into account the advantages and disadvantages of a particular approach and depending on the available information. Obtaining similar estimated quantities on the basis of different methods may indicate the reliability of the calculations.

The presence of an industry-wide technique, developed on the basis of nuclear industry experience would help to unify, organize, and justify equipment assessment process, as well as to improve nuclear facilities construction cost control in order to achieve economic competitiveness for the projects.

FEATURES OF DISCOUNT METHOD APPLICATION WITHIN THE IMPLEMENTATION OF NUCLEAR POWER PROJECTS ECONOMIC EVALUATIONS

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The report presents the approaches for economic analysis and practical techniques based on the use of the discount method. The short historical and theoretical insight is performed; the modern application and the trends in the development of the method are pointed out.

The discount rate is emphasized as an important economic parameter, which is applied within the discounting technique and methods. Interpretations, contradictions and approaches to determining the value of the discount rate are analyzed. The economic criteria outcome sensitivity to the discount rate parameter when performing economic assessments is pointed out.

Features of discount method application within the implementation of nuclear power projects economic evaluations are reviewed. The nuances of the practical application of the method for different stages of the project life cycle and the economic analysis of the nuclear power system are marked out. The justification of the discount rate values with allowances made for current international approach to this parameter is additionally emphasized.

The recommendations for improving the analytical discounting techniques in the nuclear industry are proposed for consideration.

THE MAIN GOALS AND OBJECTIVES OF THE FINANCIAL AND ECONOMIC BLOC OF CONCERN ROSENERGOATOM OJSC

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The report describes the State Corporation Rosatom member, Electric Power Division (hereinafter - Division), defines its role in the overall structure of the State Corporation and in the power sector of the country as a whole. It also briefly presents the structure of the financial and economic Concern bloc, its responsibilities and area of its operations influence on the overall Division performance.

The report presents the main long term objectives and tasks set by the State Corporation Rosatom for the Division, directed at the efficient provision and an increase in plant-produced electricity in the country, ensuring the safety of production and operation of a nuclear power, increasing the scale of operations, including the development of new business streams, as well as improving operational efficiency and competitiveness of products and services in the markets.

One of the main tasks of the financial and economic bloc is the creation of a modern information and analytical system. In this regard the prospects of development of Concern information analytical system are highlighted, as well as its scope and the main results which are to be achieved on its incorporation.

METHODOLOGY FOR ESTIMATING ENERGY COSTS: SCIENTIFIC AND PRACTICAL ASPECTS

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Cost valuation of various generating technologies electric energy is performed by calculation or simulation, and is usually required in two situations:

1. Comparative analysis of generating technologies or power stations' projects within a single generating technology (performed in order to select one of the options or to determine the optimal structure of the energy system as a combination of several options).
2. Government regulation of tariff setting and allocation of benefits for a particular generating technology.

The following periodization of the methodological approaches to the assessment of energy costs development can be defined:

- the stage of local solutions, based on commercial payments basic principles (till 1950);
- the stage of penetration and systematization of fundamental economic and mathematical methods into energy costs assessing field (1950-1980);
- the stage of development of energy costs assessing applied methods (1980-2005);
- the current stage of implementation of the methodology for specific situations (from 2005 to present).

The evolution of the estimated cost of a kilowatt-hour concepts analysis suggests a close relationship between the scientific and academic research and practical necessity to make decisions on various aspects of the electricity markets development and their separate generating objects. The context of the origin and formation of various theoretical approaches is very important for the understanding of both the approaches and possibilities for their use at the present stage.

Different methodological approaches to energy cost assesment have been emerging since the inception of the electricity market (a little over 100 years): long- and short-run marginal cost (LRMC, SRMC), total life cycle costs (TLCC), levelised cost od electricity (LCOE), systemic levelised cost of electricity (sLCOE), levelized avoided cost of energy (LACE). Some concepts do exist at the present time, other performed a historic role, and have been integrated into a more general theory.

The whole scope of subsequent development of methodological approaches to the electricity cost assessment can be predicted as a response to new requests against the backdrop of the need to make decisions on new technologies and / or projects.

State Corporation "Rosatom" has developed electricity cost assessment concepts, which are used at different stages of nuclear power plants life cycle and is carrying on with an improvement of tools for modeling the cost of the kilowatt-hour, NPP-produced under Russian production technique (resource model and dynamic simulation system).

ROSENERGOATOM CONCERN OJSC OPERATION IN THE WHOLESALE ELECTRICITY AND POWER MARKET. RESULTS AND PROSPECTS

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The two-part model of the wholesale market for electricity and power (hereinafter - wholesale market) operates in Russia at the present time. Related variable costs of power plants are covered by the electricity trading, relatively fixed are covered at the expense of power trading. Electricity trade takes place every day (the next day), power trade is carried out by means of long-term competitive capacity auction (hereinafter - CCA). Special power trade conditions are determined for new facilities to guarantee the payback of capital costs.

The report presents the specifics of the wholesale market NPP rules, a detailed analysis of the Concern Rosenergoatom OJSC revenue mix structure in various market sectors subsequent to the results of 2015 operation and a comparison with other major generating companies. In addition, it will present the analysis of the most significant events that have affected both the functioning of the wholesale market in 2015, and the results of the NPPs operation on the market: present power consumption situation in the Russian Federation, the adoption of long-term competitive selection rules for 4 years in advance, payment discipline situation, new NPPs units market implementation and the certification of their capacity, investment sourcing for integration of new NPPs to the Russian UNPG, rescheduling the obligation to deliver a new generation power to the market.

IMPROVING THE COMPETITIVENESS OF NPPS BY REDUCING ELECTRICITY PRODUCTION SEMI-VARIABLE COSTS

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Under the decrease in world power prices, particularly of natural gas, the level of competition in the wholesale market of energy and power is increasing. Considering the objectively higher cost of NPP construction and, consequently, the higher proportion of rate depreciation, as well as a significant reduction of thermal generation fuel component, in order to preserve the competitive position, it is necessary to work systematically on the reduction of nuclear power costs.

Variable NPP costs consist of the fuel component. Except for the price they are largely determined by technological factors; for that reason, the main task of Concern Rosenergoatom OJSC (hereinafter - Concern) is the reduction of semi-variable costs (hereinafter - SVC).

Work on cost reduction is carried out at the Concern from the date of approval of the Federal Target Program "Development of Nuclear Power Industry", however, in the last two years, it is entering a qualitatively new level. For 2016 the task is to reduce the specific SVC in reliance on installed capacity by 25% in comparison with 2013 comparable prices.

The report reflects the SVC structure and dynamics over the past years, as well as the approaches to reducing them in the context of the structure:

- personnel-associated costs;
- maintenance-associated costs;
- other expenses, which include security services, telecommunications, rent, consulting, information and technical support, advertising, travel expences, entertainment expenses, etc.

EVALUATION OF FUEL UTILIZATION EFFICIENCY AT UKRAINIAN NPPS

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The analysis of the fuel use efficiency was carried out at Ukraine NPPs on the basis of the fuel component of the supplied electricity cost. This value is determined by the accounting method as the quotient of the cost of NF, spent (decommissioned) for the production of electricity released during any period, and the volume of electricity, released during the same period. At that, the cost of refueling lot is decommissioned during the single campaign. Thus, the dependence of the values was determined by refuelling lot cost values, rather than by the active core arrangement and the operation

For a more accurate assessment of the efficiency of fuel use in stationary feed of VVER-1000 reactor power plant units at Ukrainian NPPs, a "Methodology of calculating of VVER-1000 reactors power plant units' fuel use at NPP separate division" (hereinafter - the Methodology) was introduced in early 2015. The methodology establishes the procedure for the calculation of these values, as well as the form and the procedure of the results presentation.

Evaluation of fuel use efficiency is based on the following estimated values: utilization of the energy potential, unloaded fuel assemblies average burnup and the specific consumption of natural uranium (hereafter - fuel use indicators).

The fuel use indicators are calculated by the finished fuel campaigns and fuel cycles, as well as for fuel campaigns, which are planned with and without consideration of the reactor operation, featuring power reactivity effect.

With the introduction of new types of fuel assemblies and advanced fuel cycles to the NPP separate division, the Methodology might be a subject to change, which will allow to calculate fuel use indicators with a different output data.

The report presents the fuel use indicators, received during 2015. On the basis of this data, the analysis of the fuel use efficiency is carried out, and the basic dependencies of the results are calculated.

URANIUM RESOURCES SAVING DURING THE CONVERSION OF THE VVER-1000 REACTOR TO RECYCLED FUEL

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Currently, one of the strategic objectives of the State Corporation Rosatom is a global expansion of VVER technology platform. It is aimed at strengthening the position of the State Corporation on the global market of nuclear technologies abroad, accompanied by the growth of products and services supply by the State Corporation throughout the life cycle of NPPs, as well as the services for the development of nuclear infrastructure of Russian nuclear power technologies recipients countries.

One of the criteria for sustainable development of nuclear energy in accordance with the phased approach to nuclear infrastructure development for the purpose of construction of the first NPP is a reliable and uninterrupted fuel supply. To this end, the State Corporation, as a responsible vendor, provides services, aimed at the supply of fresh nuclear fuel (FNF) and the return of spent nuclear fuel (SNF) in addition to reactor technologies. One of the additional options provided by the State Corporation Rosatom is the possibility of the fabrication and supply of reprocessed uranium (REPU) based FNF. For example, according to the contract on the construction of Hanhikivi NPP, the first core inventory is planned on the basis of REPU spent fuel from VVER-440 reactors, operating in Finland.

This paper discusses various strategies of conversion of VVER-1000 reactors to REPU-based fuel depending on the existing regenerator resource base: single reactor (in the case of newcomers), six reactors (in the case of countries with well-developed nuclear power of Bulgaria scale), six reactors together with regenerator, formed from the accumulated spent fuel in on-site SNF store (838 FA) (a country with an advanced nuclear power and accumulated SNF).

NPP ECONOMIC INDICATORS SIMULATION, PERFORMED IN THE SYSTEM MODEL OF NUCLEAR COMPLEX "SMAK"

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System model of nuclear complex computational code ("SMAK") is designed to support management decision-making in the course of construction projects analysis of NPP with fast reactor and fusion reactor.

The purposes of creation:

- Improvement of management tools quality
- Improvement of the decision-making efficiency and validity
- Speed and flexibility in the course of projects competitiveness analysis.
- Functionality:
- Building a nuclear fuel cycle enterprises resource model and an economic model of the project throughout the life cycle
- Simulation of capital and operating costs
- Analysis of the results in on-line mode
- Comparison of projects under equal conditions
- Research, comparison of technologies, solution of inverse problems.

Object-oriented programming and modeling, implemented in SMAK computational code, allows to compose a desired structure grid from various functional modules (power plant unit, fabrication/re-fabrication module, processing module, RW management module), depending on the type of power plant unit and the technology of nuclear fuel cycle, for the construction of economic and material balances.

Individual technical and economic characteristics are given to each object in the system; economic (commercial, fiscal, social) effectiveness of the project within different prices level (forecast, current, deflated) is determined. On the upper level indices are integrated to determine the economic efficiency of the energy complex as a whole. The present value of levelised cost of energy (LCOE) is an integral upper level indicator, which characterizes the competitiveness of the whole project and is defined in the model inclusive of discount.

"SMAK" computational code implements various capital cost settings, depending on the project development:

- Summary cost estimate
- Resource estimate
- IAEA accounts
- Calculation by peers
- Calculation by specific indicators
- Combined.

The structure of operating costs determination corresponds to the common structure of the Concern Rosenergoatom. Fuel costs are determined with allowances made for the structure of the nuclear fuel cycle and for the cost of all process stages. The "SMAK" computational code explicitly simulates material balance for the fissile isotopes during both the initial stage of the nuclear fuel cycle, and the final stage, subject to RW management.

SEM consumer properties:

- The possibility of complete alienation from the developer
- User-friendly ergonomic interface
- All labels and tips are performed in Russian
- Minimum requirements for computer resources, real time calculation
- All calculated data may be exported to MS EXCEL.

RATIONING AS THE BASIS OF THE FINAL PRODUCT COST MANAGEMENT

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Rationing of labour is the basis of successful development of the industry economics, for work/services correct price formation, and for healthy competition.

Rationing allows to identify the extent of each employee involvement in the final product implementation, and therefore to understand their contribution to the common cause and calculate their efficiency. As a result, the efficiency of individual employee is directly related to the efficiency of every enterprise activity, which is especially important in the current situation of economic downturn in the country as a whole.

Rationing is the basis of "pass-through cost" system, a model, which shows, for instance, how the increase of cost of uranium as a "primary redistribution product" or cheapening the services cost affects the price of electricity as the "final product" of the nuclear industry. This provides additional revenue and increases investment resources for further development.

RELEVANT ISSUES OF THE NEW NUCLEAR POWER PLANT CONSTRUCTION COST MANAGEMENT

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The purpose of the construction cost management is to ensure the formation of the construction cost in accordance with the pricing policy of the state in the construction in order to ensure the effectiveness of investment projects carried out in the form of capital investments.

The basic principle of the construction cost management is the methodological unity of approaches, the continuity and logical interconnection of control procedure, determination and monitoring the nuclear power plant construction cost during the process of management of capital investment and construction management at all stages of the project life cycle.

Cost management of NPPs under construction includes the following steps:

1. Capital planning and capital investments control.
2. Estimation of the project cost.
3. Estimation of capital investments efficiency.
4. Setting target values for the project implementation cost.
5. The formation and the examination of the estimated cost.
6. Project implementation cost control (monitoring and analysis of the actual costs, comparison with the previously planned values, monitoring of the scope of work, actually carried out, and monitoring of deliveries for compliance with the terms of the contract).
7. Developing measures of corrective and preventive nature.

The main problematic issues of pricing and cost management of construction:

1. Non-compliance of construction projects during the stages Project and DD.
2. The lack of developed and constantly updated full industry budget-normative base (IBNB), satisfactory in the context of modern technologies and material resources, used these days, taking into account peculiarities of the construction of NPPs, and corresponding to the approved project in order to avoid the alleged deficiency or estimated limit gain, as well as of industry standards and cost accounting rules, ambiguously included in the summary cost estimate, indirectly related to capital expenditures.
3. The risk of changes in the cost of equipment and building materials of foreign production.
4. The risk of failure of equipment suppliers and contracting construction organizations to carry out their obligations.

Section 4

INTERNATIONAL COOPERATION FOCUSED ON ENSURING NPPS SAFETY

AN ANALYSIS OF THE LEGISLATION OF THE RUSSIAN FEDERATION, THE EU AND THE UNITED STATES, APPLICABLE AT THE STAGE OF THE CONSTRUCTION OF NPP SITE SELECTION

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Legal regulation in the field of peaceful application of atomic energy, and supervision over the safety of its use is one of the most important elements in ensuring NP safety and nuclear energy development. There are various methods and approaches to the management of design, siting, construction and operation of the NP. Comparative analysis of the normative base of the Russian Federation, the EU and the United States allowed to match the requirements for the selection of potential sites for NP, to identify the main differences in the approach, and to outline future directions for the promotion of Russian NP projects in international nuclear power markets.

Russian normative documents, as well as in the United States regulatory documents contain quite comparable and relevant requirements. Minor differences are determined by the individual requirements wording and terminology, used. Unlike the IAEA standards, which are conceptual and declarative, requirements of Russian normative documents and regulatory documents of the United States are more specific. By comparing the results it can be concluded that, in general, the documents of the Russian Federation, the United States and IAEA content-wise coincide, but there are some differences:

- To assess the natural and man-made origin influences, IAEA standards provide a number of documents for each type of exposure. Unlike the IAEA documents, normative documents of the Russian Federation and regulatory documents of the United States use similar documents, which include all kinds of external influences;
- Russian standards define more specific volumes and requirements for study and research when selecting NP site;

Good US practices are represented in a number of documents:

- Detailed study of the site selection criteria for NP placement;
- Creation of working groups with the permanent basis involvement of highly qualified professionals for each for the selected criteria matter;
- Criteria weight evaluation and identification of appropriate service functions.
- There is a uniform criterial approach to the suitability of the site to accommodate the standard nuclear power plant projects.

In general, subsequent to the results of the comparative analysis, one can conclude that the projects of Russian NPPs meet all modern standards and requirements and contribute to increasing the attractiveness to foreign customers in the course of international tenders. The aim of future work is the application of good US and EU practices for the implementation of Russian projects abroad. To this end, some recommendations were made.

IBERDROLA NUCLEAR PROJECTS IN RUSSIA AND UKRAINE

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Iberdrola Engineering and Construction started its presence in Ukraine in 2004, when it was awarded with a project in Ukraine aimed at helping with the implementation of some safety measures required by the IAEA at **Khemelnitsky –II and Rovno –IV NPP**. The final objective was to help NAEC Energoatom to get loans from the European Commission, Euratom and the EBRD for the modernisation of the plant.

Following this relevant project, Iberdrola was awarded with the **On-Site Assistance** project at **Khmelnitsky NPP**. Since 2004 and during four years Iberdrola helped with the implementation of some *hard* projects (those related with the supply of new components or systems) and *soft* projects (those related with improvement of management, methods, procedures, skills or culture): emergency procedures, alarm sheets, outage optimization, etc.

In 2005 Iberdrola was awarded by EC with its first project in Russia. An **On-Site Assistance** project for **Kola NPP** were developed from this year till 2009 supporting Kola NPP with the implementation of some *hard* projects: Liquid Radioactive Waste Treatment System, Fire Protection and Leak detection systems. Some *soft* projects were also developed: improvement of Outage Planning, Maintenance Procedures, Safety Culture, Waste Management, Foreign Materials Exclusion, ... The project ended in 2011.

Between 2007 and 2012 Iberdrola developed a similar project at **Novovoronezh NPP,** also in the framework of EC Instrument for Nuclear Safety Cooperation. A very relevant *hard* project was implemented: Emergency feedwater to Steam Generators System was supplied and housed in a new safety building built up by Novovoronezh NPP. The *soft* activities were also related with the interchange of operation and maintenance best practices: Human Factors, support to OSART mission, Safety Culture, Radwaste management, CRUD exclusion, etc.

In parallel with these projects, and starting in 2005, two Collaboration Agreements (CA) were stablished yearly between Iberdrola and each one of the main nuclear utilities: Concern Rosenergoatom and NAEC Energoatom. These CA were aimed to technical and scientifical interchanges looking for best practices in maintenance, operation, radwaste management, core dynamics, emergency preparedness and support, plant modernization.... These CA were fruitful in terms of knowledge interchange and promoted the development of some different projects.

From 2006 till 2010 Iberdrola developed another INSC project in Russia, **Influence of neutron irradiation on material of reactor internals**, bringing a complete set of validated experimental data and scientific/technical/safety methodology, which are required to address the issue of structural integrity of the core internals, and which would include the latest international experience.

In the period 2007-2010 Iberdrola performed the **implementation of Risk Monitoring technology at Balakovo NPP,** providing technical assistance to develop the Risk-Monitoring tool in VVER-1000 type Russian Nuclear Power Plants, and transferring the needed know-how.

Another relevant project was the development in 2009-2010 of an **Efficiency Study at Novovoronezh 5** in order to search for solutions to improve the Energy production of the Plant (electric Megawatts, MWe). Following this analysis, from 2012 till 2014 a project started willing to implement **Ultrasonic Flow Meters** in the feed water pipes of Novovoronezh and Kalinin NPP that would have provided a cheap 2% increase in the electrical output. Metallurgical and metrological certifications were successfully achieved before the project stopped.

In South Ukraine NPP, from 2006 till 2008 Iberdrola developed the **Replacement of the thermal insulation** for sections of pipes and equipment of the primary circuit inside the containment building.

From 2007 to 2009 a technical support to special Energoatom sub-division (SE) called 'Scientific and Technical Centre' (SE-STC) were provided: transfer of best EU know-how on Quality and Safety Management, and of specific technical skills, in order to enable SE-STC to provide an effective and sustainable scientific and engineering support to NAEC Energoatom.

In Ukraine (Zaporozhye NPP) , from 2009 till 2015, a supply of a **Radwaste Incinerator facility** project has been successfully developed in a turnkey model inside the EC cooperation program.

Today, three INSC projects are in their final steps of development in Ukraine: **Safety Culture** best practices transfer, **Risk-Informed In-Service Inspection** of safety pipes optimization and **Living PSA** implementation. All of them are expected to be concluded in 2016.

JUSTIFICATION OF THE "LEAK BEFORE BREAK" CONCEPT APPLICABILITY FOR MOCHOVCE NPP, ENERGY UNITS 3 AND 4

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VNIIAES

The work has been undertaken to identify safety conditions of main circulation piping respiratory pipelines of power units No. 3 and No. 4 with reactor plant VVER-440, based on the concept of "leak before break" (LBB).

In order to ensure the applicability of the LBB concept it is necessary to calculate the stress-strain state (SSS), to determine the critical crack length, crack opening, the stability of cracks in emergency conditions, or the maximum-design earthquake (MDE) mode, as well as to calculate the coolant flow rate and choose a leak monitoring system. All these tasks were carried out under contract with the Mochovce NPP (Slovakia) for energy units 3 and 4: LBB Activity for MO 3, 4 No. 4600005433/820/10 from 16.06.2010.

The results of the calculations, carried out to determine the conditions for the safe operation of the main circulation pipelines DN500 and DN200, and respiratory pipeline in accordance with "leak before break" concept, demonstrated the possibility of use of the LBB concept in full for all the project weld and adjacent zones. Calculation modes included the following modes: normal service conditions, hydraulic testing, normal service conditions + maximum design earthquake, and stratification mode for respiratory pipeline.

IAEA OSART MISSION AS A TOOL OF IMPROVENT OF NOVOVORONEZH NPP OPERATIONAL SAFETY

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In November of 2015. Novovoronezh NPP held OSART IAEA mission organized by invitation of the Russian government and aimed at addressing issues that are important for operational safety. Activities before, during and after the mission has united the staff in order to achieve the objectives of improving the operational safety and reliability of its stations.

Strategy formulation for measures, needed to be taken to eliminate possible inconsistencies began with in-depth self-examination, according to the results of which NPP activities, demanding enhancement, were identified: radiation protection systems, documentation, integrated management system, laboratory equipment, archives and storage facilities, outdated machine-tool mechanics shop fleet, imperfect work practice and so on.

In the course of preparation about 20 seminars were conducted, highlighting OSART methodology, clarifying IAEA regulatory requirements and checking compliance processes at NV NPP with the requirements of these documents. The total number of similar events from 2012 to 2015 is 680.

The scope of testing included power plant unit No. 5 with auxiliary systems (power distribution system, auxiliaries, backup power, steam supply, water treatment system, storage, and systems for handling and storage of radioactive substances and waste, repair shops and warehouses).

During the mission, the experts have carefully studied the number of station programs and procedures, station performance values; they have also monitored the implementation of the works and comprehensively interviewed the station staff. The mission at Novovoronezh NPP was the 186th in a row in the OSART program, launched in 1982. The total experience of the team members in the nuclear industry amounted to about 350 years.

IAEA experts confirmed the operational safety and reliability of the Novovoronezh NPP, noted seven good practices, which can be used at other enterprises of the nuclear industry. Furthermore, representatives of the International Agency and NPP exchanged technical knowledge and experience in order to discuss the possibilities of achieving the common goal - to further improve the operational safety condition.

IAEA standard practice is to conduct subsequent mission after completion of the OSART program at NPP. Novoronezhskoy NPP Management has expressed the desire to take action in the areas identified for improvement and declared its readiness to receive a return visit eighteen months later to assess the level of performance and effectiveness of corrective actions, taken in accordancew with the recommendations and proposals, developed by an international team. At the moment, the specialists of Novovoronezh NPP carry out systematic work in order to eliminate inconsistencies in all areas of activity in the run-up to the cutoff control, as the main benefit of the OSART missions is the preparatory work that precedes them.

ON THE ISSUE OF HRD FOR NUCLEAR INFRASTRUCTURE DEVELOPMENT IN NEWCOMER-COUNTRIES

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Implementation of a new nuclear power programme inevitably requires from the newcomer states a well-timed and resource-balanced approach to human resource development (HRD) that is to provide sustainable development of the nuclear power programme in compliance with fundamental safety principles. The present paper addresses the concept of HDR system arrangement in the country where a political decision has been taken to develop nuclear power. This concept is based on the IAEA “Milestone Approach”, on generalization of practical experience accumulated in the Russian Federation and other vendor countries in regarding their cooperation with embarking states.

The earliest stage of nuclear power programme development is of special concern. At this stage the NEPIO (Nuclear Power Programme Implementing Organization) plays a dominant leading role. In view of that, the basic prerequisites of the HRD system development are the legislative NEPIO available in the country, with clearly defined functions and responsibilities, and the HRD strategy developed by the NEPIO that covers all the19 issues of nuclear infrastructure. In its turn, it requires the availability of a pool of managers and specialists in a country, who are capable of assessing (1) the competences required to develop the national nuclear power programme and (2) the potential of national university education and vocational training systems in building up the required competences. These managers and specialists are expected also to be able to develop the HRD strategy, establish cooperation with the IAEA and potential vendor countries. With the aim to render assistance to embarking states at this stage of HRD system development the Russian Federation offers special training service as a part of integral solution to support national nuclear power programme.

A LONG TERM AND FRUITFUL COOPERATION BETWEEN ROSENERGOATOM AND EDF

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EDF and Rosenergoatom, the two world largest nuclear operators, develop a fruitful cooperation for more than twenty years. Initiated in the mid-nineties to support the Russian nuclear sector during the economic transition and with a clear focus on safety which still remain pre-dominant, this bilateral cooperation is now based for more than a dozen years on experience sharing. Through their key roles in their respective national nuclear sectors, EDF and REA have progressively developed their cooperation from operation matters to a wider field of exchanges such as engineering support, new projects and research and development.

This cooperation is illustrated by recent examples such as exchanges on post Fukushima action plans -also involving both the Russian and French nuclear regulatory agencies, long term operation concept and design options for new projects.

This long term cooperation has helped to develop a better common understanding in between the two partners at different levels and in many domains starting from safety. This cooperation also benefits the nuclear community at large through the World Nuclear Association of Operators and the IAEA.

IAEA EXPERIENCE WITH INTEGRATED NUCLEAR INFRASTRUCTURE REVIEW (INIR) MISSIONS

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The IAEA developed the "Milestones Approach" in 2007 to assist countries that are considering or planning their first nuclear power plant. The aim is to help Member States understand the commitments and obligations associated with developing a nuclear power programme.

The Milestones Approach facilitates the undertaking, at the request of an IAEA Member State, of a structured review of the country's nuclear power infrastructure - this is the Integrated Nuclear Infrastructure Review – or INIR. The INIR programme has a number of components – support to the Member State self-evaluation, pre-INIR mission, the full INIR mission and the Follow-up INIR mission. After an INIR mission, an Integrated Work Plan is developed between the Member State and the IAEA to coordinate IAEA assistance to address recommendations arising from the INIR mission. Since 2009 the IAEA has conducted 17 INIR missions in 13 countries, at their request. Looking back, an analysis of six years of INIR experience highlights which infrastructure issues require close attention during the initial phases of the development of a nuclear power programme.

THE IAEA'S PERSPECTIVE ON INTERNATIONAL COOPERATION FOR NPP SAFETY

Tarren. P

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In 2016 the International Atomic Energy Agency (IAEA) celebrates its sixtieth anniversary and its mission today is just as relevant as when it was established in 1956. The IAEA's statute authorizes it to 'establish or adopt... standards of safety for protection of health and minimization of danger to life and property... and to provide for the application of these standards...

The IAEA engages closely with member states to achieve consensus on what constitutes a high level of safety and formulates these views into agreed safety standards. This is achieved through the use of consultancy and technical meetings between agency personnel and experts nominated by a range of member states. By this means the agency calls on worldwide experience. Once agreed by the IAEA's board of governors and published these standards are made freely available to all member states to adopt. In some instances these are used primarily in organizations responsible for the operation on nuclear power plants (NPPs), in others they are used as the primary references for nuclear regulations in a member state.

The IAEA also assists member states in the application of these standards. This assistance takes the form of Safety Review Services. These are wide ranging across many of the IAEA's activities and again draw on the expertise of the global nuclear community as well as the skills of the agency staff to provide independent and transparent appraisals of how organizations in member states comply with the standards.

This paper reviews the current status of the IAEA's safety reviews in the area of operational NPPs and how these have evolved along with the industry itself and in response to major events. It also shows how the Russian Federation has engaged with and supported the IAEA with the aim of improving the safety performance of the Russian civil nuclear power reactor fleet.

ASSISTANCE TO NEWCOMER COUNTRIES` REGULATORS. RUSSIAN APPROACH

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Presentation indicates the main trends in providing by Rostechnadzor assistance to regulatory bodies of newcomer countries in developing their national system of nuclear and radiation safety regulation as a integral part of Nuclear Power Infrastructure.

Taking into consideration the fact that Rostechnadzor is an independent regulatory body subordinated directly to the Government of the Russian Federation and it is authorized to provide full scale assistance in the development of national systems of nuclear and radiation safety regulation in the countries–customers of the construction of Russian design NPP, Rostechnadzor carries out cooperation with counterparts by providing assistance in development of legal and regulatory framework in the field of nuclear energy use and by transferring practice of licensing, safety review and supervision.

Rostechnadzor, basing on the IAEA recommendations (e.g. SSG-16, NG-G-3.1 and others) assists its counterparts to bridge the gaps in their experience and competence, in legal and regulatory framework, insufficient staffing etc. by organizing workshops, technical visits and trainings as well as rendering of consulting and outsourcing services with support of its TSOs.

The legal basis for cooperation consists of intergovernmental and interagency agreements and memorandum that defines areas of cooperation such as experience exchange, licensing procedures and review of licensees` documents, inspections practices, training and consulting of regulatory personnel. For these purposes Rostechnadzor with its TSO elaborated complex modular programs and roadmaps of cooperation.

Section 5

HUMAN RESOURCE FOR NUCLEAR POWER

Subsection 5.1
TRAINING OF YOUNG SPECIALISTS FOR NPPS

SPECIALISTS TRAINING AND CONTINUING EDUCATION OF NUCLEAR POWER EMPLOYEES IN IATE SRNU MEPhI

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National Research Nuclear University MEPhI is a research group in the regions of the State Corporation Rosatom presence. Personnel training for the nuclear power complex is held in MEPhI and in several SRNU MEPhI branches as well. IATE (Obninsk),VITI (Volgodonsk), STI (Seversk) and others.

Among IATE SRNU MEPhI departments it is worth to mention, in the first place, the Department of Physics and Power Engineering that prepares specialists in the field of NPP operating, calculation and design of NPP reactors, etc., and also the Department of Continuing Education and Professional retraining providing a continuing professional education for specialists of the nuclear power complex, in particular, as per the course «Nuclear Plants: design, operation and engineering», track «NPP operation».

Proper training of young specialists, as well as continuing education and professional retraining of nuclear power employees are maintained by the high qualification of IATE SRNU MEPhI faculty and the well founded technical base including NPP nuclear reactors simulation units.

Within implementation of the Program of Creation and Development of Federal State-Funded Educational Institution of Higher Professional Education «National Research Nuclear University MEPhI» for 2009-2017, approved by the government of the Russian Federation in 2013-2015, the IATE SRNU MEPhI professors elaborated more than 20 professional development programs for industry employees and corresponding academic facilities for classes of different tracks, including «Safety and reliability of higher risk objects operation», «Violation influence analysis on NPP security», «Radiation safety assurance of nuclear industry personnel», «Environmental safety assurance during work in hazardous waste management area», etc. A similar work is planned for the following years as well.

IMPLEMENTATION OF FGOS-3+ STANDARDS' CAPABILITIES FOR GRADUATES PRACTICAL TRAINING IMPROVEMENT USING THE EXAMPLE OF THE APPLIED BACHELOR PROGRAM DEVELOPMENT «NUCLEAR POWER AND THERMAL PHYSICS»

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Based on the competency building approach an curriculum concept has been developed for the principal educational program (PEP) of the applied bachelor «Nuclear power and thermal physics» in ROSATOM- CICE&T (14.03.01). The main objective of the applied bachelor PEP development and installation is to reduce the training time period of the personnel (while the training quality is conserved), which is prepared to start an independent work as the engineering operation personnel in Russian and foreign nuclear power plants.

The reduction of the training time period is assumed to be reached by moving away from an excessive academicism, the practical and laboratory works' considerable increase, and the part of them will be held during practice at nuclear power plant basing on TCs resources.

The curriculum includes modules and disciplines paying special attention to engineering training, using of analytical simulators covering the questions of intercultural communication during the construction process and the NPP guarantee maintenance period, as well as fundamental legal principles of cooperation with subcontractors in the NPP location country.

Timetable scheduling as per the modular approach will increase students' occupancy time at NPP. The academic disciplines and practical training programs are compiled in such a manner that a student has a clear idea of his/her professional activity supported by practical experience by graduation, and passed a significant part of exams which are necessary for access to an independent work at NPP.

The comparison of education time with traditional curriculum of engineer education allows to talk about an «economy» of about 1.5 years and the clearer graduate orientation to certain problem solving of the nuclear power program's implementation in the country.

APPLICATION OF VIRTUAL REALITY TECHNOLOGY FOR PROCESS SUPPORT OF TRAINING AND CONTINUING PROFESSIONAL EDUCATION FOR OPERATION AND MAINTENANCE PERSONNEL

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An important work process of maintenance personnel at an industrial site is training and regular continuing education. For education and learning control purposes different simulator complexes are created and used, as well as computer learning systems (CLS), technical learning systems (TLS), and imitation complexes and analytical simulators.

Computer systems allow to practice production processes, for example, of maintenance, emergency procedures, equipment diagnostics, including equipment location search at an object site. This allows to reduce considerable personnel errors during technical operations at the real objects.

This report is dedicated to Neolant CJSC, following TLS:

The release modeling complex during radiation accidents and procedures in case of fire.

The Complex Virtual Reality Room of 6, 7 units of Novovoronezh NPP.

The release modeling complex during radiation accidents and procedures in case of fire (TLS DP PZPRA)

The main objective of TLS DP PZPRA creation and implementation is to increase the maintenance personnel training's quality by application of the high technology education including 3D imaging and interactive learning control. And it is intended for a solution of the following basic tasks:

1. Demonstration of quality and quantitative characteristics of radioactive release to the environment in case of an accident beyond the design basis (release direction and velocity, activity, radiation intensity) in 3D format and depending on the ambient conditions.
2. Presentation of release consequences for population (radiation doses, residential ares exposed to radiation pollution).
3. Providing recommendations for an elaboration of population protection measures (dose characteristics rating with critical and normal levels, comparison with criteria for urgent decisions making during the initial period of a radiation accident).
4. Information presentation about the emergency rescue service activity (optimization of escape routes, equipment delivery and forces and measures' distribution, etc.).

5. Demonstration of a situation around nuclear- and radiation-dangerous facilities to the appropriate authorities and community in case of an accident beyond the design basis.

6. Reflection of mechanisms, routes and order of informing all the services participants of emergency response, including local and federal authorities, during the radioactive release in an emergency in nuclear- and radiation-dangerous facilities.

7. Demonstration of the automated radiation monitoring system functioning during gas and aerosolised radioactive release in a sanitary protection zone and a radiation-control area in an emergency in nuclear- and radiation-dangerous facilities.

8. NPP personnel training in case of an accident beyond the design basis in accordance with the «Personnel security action plan».

Complex Virtual Reality Room of 6, 7 units of Novovoronezh NPP

The main objective of the virtual reality room (VRR) creation and operation is to increase reactor shop maintenance personnel training's quality by application of the innovative technology education including 3D imaging and interactive learning control.

The VRR is created for demonstration and learning control regarding a location of the controlled access zone equipment and rooms and the enclosure vessel of 6, 7 units of Novovoronezh NPP.

In the furtherance of this goal, the VRR assures the following tasks solution:

- Rooms and equipment indication in the controlled access zone of 6, 7 units of Novovoronezh NPP in 3D model.
- Information display regarding the equipment located in the controlled access zone and the enclosure vessel of 6, 7 units of Novovoronezh NPP, and its characteristics.
- Free navigation availability across the model in the walking simulation mode.
- Availability to guide a route, where the trainee specialists should walk with an indication of the equipment that must be supervised in a determinate order.
- Availability to edit the equipment data.
- Availability for an instructor to monitor the trainee's activities and to receive reports regarding routs passing.

ENGINEERING AND SCIENTIFIC PERSONNEL TRAINING SYSTEM FOR NUCLEAR POWER INDUSTRY

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The inform includes basic stages and characteristics of the engineering and scientific personnel training system. The starting point in the system development was an opening of the Department of Physics and Technology in Gorky Industrial Institute in 1961 upon an initiative of Africantov Igor Sergeevich. The key milestone in formation of the personnel training system's modern image was the «Cooperation Agreement (strategic partnership agreement) between NNSTU and OKBM» concluded in 2007. According to the Agreement for the target specialists training in the reactor facility design sphere the specialized department «Nuclear Power Plants Design» was created and started functioning in Nuclear Energy and Applied Physics Institute of NNSTU. During the period 2008-2015 about 450 persons graduated from this department, more than 140 persons passed the predegree practical training in the organizational unit and accomplished a graduation project regarding OKBM subject matter. 118 persons of them defended their graduation projects in the state certification commission of the organization.

«OKBM Africantov» OJSC received a license for educational process performance of the scientific personnel training in terms of a postgraduate training program with a degree in: 05.04.11 - Nuclear reactor engineering, machines, units and materials technology of nuclear power industry; 05.14.03 - Nuclear power plants, including design, operation and decommissioning, where nowadays 30 specialists take training course.

On the basis of «OKBM Africantov» OJSC in cooperation with NNSTU the United Dissertation Committee of Defense of Master's and Doctoral Dissertations was created, the Committee numbers 25 Doctors of Science from OKBM, NNSTU, RIAR, IPPE, UNN.

The current scientific research and development have tracks such as active core research, examination of equipment and its components as a security of an accounting codes verification, etc.

Every year the organizations holds a Best Young Engineer Contest. 60 young specialists took part in this contest in 2015, 25 of them passed to the second and final stage, 10 contest winners were awarded with a letter of recognition and money prize.

The Youth Policy was established and put into operation till 2020, new aims and tasks were set: young specialists' professional growth, connection between generations and transfer of experience. This experience is recommended for expansion on other plants of State Corporation Rosatom.

The work on the engineering and scientific personnel training by the specialists of «OKBM Africantov» OJSC and NNSTU culminated in Nizhny Novgorod City Prize in the sphere of education in 2012.

The engineering and scientific personnel training system for the nuclear power industry assimilated the long experience in creation, development and perfection of personnel training methods in «OKBM Africantov» OJSC.

Subsection 5.2
USE OF VETERANS' EXPERIENCE AND EXPERTISE. TUTORSHIP

SPIRITUAL AND MORAL, LABOUR, ENVIRONMENTAL AND PATRIOTIC EDUCATION OF YOUNG PEOPLE, CAREER GUIDANCE AND NUCLEAR POWER ACHIVIEMENTS PROMOTION

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The work on the spiritual and moral, labour, environmental and patriotic education of young people, career guidance and nuclear power achievements promotion are held by our veteran organization in cooperation with the specialists of the principal departments, Information Management and Public Relations, and the young nuclear experts' foundation of Rostov NPP.

Labour and environmental education

It is, first of all, by personal example.

The projects oriented to restore Volgodonsk Arboretum and "Golden Autumn" have started. The environmental festivals "Glorified Steppe", "Ecoworld" and two ecological cleanup days in the recreation camp and city residential areas celebrated.

64 contracts were concluded with veterans for their participation in the outage-2015, thus the second integration into profession is operated for the retired employees of the Concern. The main capital of Rosatom is a working man, his competence and professionalism. Knowledge and experience of veterans are in demand, and they acting as tutors share them with a greatest pleasure with young people.

Career guidance and nuclear power achievements promotion

We educate the rising generation in the Data Center of Rostov NPP ("round tables", thematic lectures, excursions) and in the city educational establishments.

With financial support from Rosenergoatom Concern four "nuclear" classes have been established, where also the career guidance is arranged.

The best school students are awarded with the target contracts for university studies with a right of employment in NPP after graduation in 2015.

The event "Meeting of three generations" was celebrated with the participation of veterans, NPP young specialists, students of VITI, SRNU MEPhI.

Spiritual, moral and patriotic education

It is, perhaps, the biggest and the most important part of work with young people. Considering that there is no future without past, we are trying to educate the rising generation with examples of heroic past of our nation. We hold events ("round tables", classes dedicated to remembrance and courage "To those who went through atom", meetings and floral tribute) confined to festivals and commemorative dates.

It became a tradition to run the Immortal Regiment march in cooperation with young people.

In cooperation with Rostov NPP veterans and young nuclear experts started the project called "Soldier's Tomb". By the 70th Victory anniversary 6 soldiers' burials were restored in Rostov NPP location region, which had been in a sad condition.

3-days youth bike ride took place in May covering the battle field in Rostov region, which was dedicated to the 70th Victory anniversary.

Our veterans with pleasure share their knowledge, experience through mass media, take participation in material preparation for the corporate newsletter "Atom Energy", MOOVK website, regional press (74 publications).

NUCLEAR FUEL MANAGEMENT AT SMOLENSK NPP

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The inform presents the nuclear fuel handling instructions at Smolensk NPP reflected in the Regulations for nuclear fuel management. The nuclear fuel handling instructions are stated from the moment of its delivery to the New Fuel Storage of Smolensk NPP and further operations with it at the generation units and to the Independent Spent Fuel Storage Installation (ISFSI), up to its preparation and storage operation before sending the packing storage set UPH-109 to the Regional Storage for custody and recycling. The inform describes that all the technological operations of nuclear fuel handling at Smolensk NPP are held taking fully into account the Project requirements and specifications documents valid at the present time.

NPP SAFETY CONTROL IN ACCORDANCE WITH FEDERAL RULES AND REGULATIONS

The inform specifies problems of safety control at nuclear power plant (NPP):

- safety principles;

- safety criteria;
- safety functions.

Nowadays there is a quite perfect safety conception at nuclear power plant and a real regulatory framework covering the whole life cycle of NPP.

The inform presents safety requirements at NpP and meeting these requirements at Smolensk NPP.

BUS DUCTS APPLICATION

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Nowadays Clients evaluate the whole project in an integrated manner obligatory taking into account not only the equipment cost, but also its installation, maintenance and possession. From this point of view the bus duct is beyond competition: it does not require any attention in operation, meanwhile the bus duct service life can be compared with the operating life of the whole unit.

The tightening of fire safety regulations also contributed to the consolidation of the bus duct position also. The cost of events of cable treatment with special incombustible compositions is high. The treatment should be renewed, and special incombustible cables of VVGng and VVVng1_B types can not be called cheap. And it is quite difficult to find them with sections of 240 mm^2 and higher!

All these years there is a constant growth of power consumption: the cable sections are getting bigger, and even copper cables by no means everywhere enable to transmit the required power to necessary distance. The problem is also in minimal cables bend radiuses, the cable requires more room in the extremely expensive meters our times. The magistral bus ducts even with aluminum buses have no this problem, as bus ducts components can be connected at any angle to each other. The copper cost increase has completely reversed the situation. As the cable precisely with copper core are used for interconnection wiring inside buildings in the vast majority of cases. Bus ducts with aluminum buses are not very much different per dimensions from the copper ones having the better rate up to 40%.

Nowadays the power distribution systems with high quality bust duct system gained widespread use allowing in significant way to lower the arising electrical power quality problems in operation at construction sites. The bus duct systems allows not only spare financial means at design and installation works, but are currently the only energy conservation equipment being convenient in operation and enable to replace cable systems.

The bus duct systems enable to transmit significant power to a long distance with far less line voltage drop. By virtue of the conductor rectangular section, the bus duct systems has the lowest inductive circuit impedance, thus lowering full power consumed by circuit, and consequently electric losses. Application of the distribution bus duct systems in a building lowers conduction field of heaviest current conductor under application of the rectangular conductors section. The electromagnetic field by virtue of the metal protective cover almost completely becomes localized inside the bus duct, and at the 5 cm distance from the bus duct casing will not cause any disturbances in the information networks. Besides, it is know that the harmonic current components die down, while the conductor rectangular section "damps" high harmonics currents directed by nonlinear loads the faster and the more effective way in comparison with the cable systems, thus lowering heat losses under power transmission and also allowing to "straighten" the current sine wave without applying a specialized expensive electronic equipment.

Software tool for generating active core loading and visualization maps, Visual Reactor Core Control (VRCC)

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During commissioning and further operation phases of VVER-1200 units of Leningrad NPP B-491 project the Nuclear Safety and Reliability Department personnel of Leningrad NPP-2 needs the information to be presented in the form of an active core map. For addition of operating software functions with possibility to display the maps (software package KASKAD, program "Reactor Simulator" (RS) forming part of the complex analysis system) the Software Tool (ST) VRCC has been developed, which is Visual Reactor Core Control.

The ST VRCC includes four independent program modules:

1 Simulation Area and Nuclear Fuel Modules:

- enable to modulate sequence of nuclear fuel loading and unloading (fuel assembly simulators) including all its types, in automatic mode (with 1 s time step), and in manual mode as well (after every loading step a message is generated with a request to continue or to finish the loading);
- enable to generate nuclear fuel loading/uploading maps (fuel assembly simulators), to change them with fuel assembly and adjuster road serial numbers in case of necessity;
- enable to display the fuel assembly different types (fuel assembly simulators) on a map, the fuel elements location in fuel assembly, the group distribution of reactor control and safety system regulating elements (RCSS RE).

Simulation Area and Nuclear Fuel Modules can be used for the generation of different active core maps, and the additional loading control of simulation area and nuclear fuel during the fuel change process at the workplace of an auditing physicist.

2 The NFCE-ICMS module consolidates information regarding NFCE and ICMS sensors location and displays in a representative style.

3 The Parameter module represents a VVER-1200 reactor type map, where it is possible to display any parameter accompanied by the fuel assembly graphic presentation coloring (or fuel assembly simulator) depending on its value (the color changes from the lesser value to the major), and visualization of any of 6 symmetry sectors, as the active core parameters' evenness in VVER reactors is frequently considered within symmetry sector limits.

During the Leningrad NPP units construction the ST VRCC already is a right hand for consideration and integration of the reactor simulation area loading programs, first loading of reactor operating active core, programs and testing methods of NFCE and ICMS, development of accompanying documents requiring maps domain, and can complete a task of the personnel efficient training taking into account AES(NPP)-2006 B-491 project specifications. In the medium term the developed software tool can be completed with a set of functions and modules as soon as necessary tasks are available.

The ST VRCC use completes operating software tools and enable to minimize the personnel error probability during work performance related to the active core loading and the measured and design parameters processing of the active core.

NPP SAFETY CONTROL IN ACCORDANCE WITH FEDERAL RULES AND REGULATIONS

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The inform specifies problems of safety control at nuclear power plant (NPP):

- - safety principles;
- - safety criteria;
- - safety functions.

Nowadays there is a quite perfect safety conception at nuclear power plant and a real regulatory framework covering the whole life cycle of NNP.

The inform presents safety requirements at NPP and meeting these requirements at Smolensk NPP.

POPULATION AND ENVIRONMENT PROTECTION AGAINST RADIATION DURING RUSSIAN NPPS OPERATION

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The personnel and population protection against radiation during NPP operation in accordance with the General Safety Regulations of Nuclear Power Plants NP-001-15 is assured by compliance with the radiation security legislation of the Russian Federation, as well as federal rules and regulation in the field of nuclear power use and other normative legal acts.

On the base of the radiation monitoring results' analysis the article reviews the radiation safety level of population and environment protection against gas and aerosol releases, liquid radioactive effluent releases, and radioactive waste management at NPPs in 2011-2015. The radiation monitoring results are compared with normative standards, federal rules and regulations in the field of nuclear power use.

RUSSIAN FEDERAL RULES AND REGULATIONS IN THE RADIATION SAFETY FIELD. COMPARISON WITH THE RECOMMENDATIONS, PROVIDED BY RADIATION SAFETY INTERNATIONAL DOCUMENTS

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The inform presents the Public Health Regulations' provisions NRB-99/2009, the basic document interpreting and presenting details of the Russian laws requirements for personnel and population health protection against the risks which are results of effect of ionizing radiation:

- Regarding nuclear power use;
- Regarding population safety against radiation;
- Regarding population sanitary and epidemiological safety.

Alike IBSS (International Basic Safety Standards for Protection Against Ionizing Radiation and for the Safety of Radiation Sources) the NRB-99/2009 is the high level document in the regulatory and procedural documents' hierarchy which are the basis of National (Russian) Infrastructure of Radiation Safety Regulation and Assurance.

The NRB-99/2009 settles the fundamental principles forming the basis of the modern radiation safety system (normalization principle, substantiation principle and optimization principle) and criteria of population safety assurance against radiation and risk management of late radiation effects appearance on human body or separate body parts taking into account its radiosensitivity, doses (values) of ionizing radiation effects under normal conditions of ionizing radiation source operation, planned personnel radiation increase over the set dose limits and in a radiation accident conditions.

RADIATION SAFETY LEVEL AT THE PRESENT STAGE

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The inform reflects the radiation safety system and modernization of the environmental radiation control and monitoring facilities at Smolensk NPP held after the accidents at Chernobyl NPP and Fukushima NPP.

The inform contains a large analysis of:

1. Process radiation monitoring system (PRM).
2. Reactor coolant leak detection automatic system (RCLDAS).
3. Radiation and contamination monitoring system (RCM).
4. Environmental radiation monitoring system (ERM).