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REPLIES

TO EXPERT OPINION ON PRELIMINARY REPORT ON EIA OF THE BELARUSIAN NUCLEAR POWER PLANT BEING CARRIED OUT ON REQUEST OF THE FEDERAL MINISTRY OF AGRICULTURE, FORESTRY, ECOLOGY AND WATER RESOURCES

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1 INTRODUCTION

The authors of the Report on EIA of the Belarusian Nuclear Power Plant express gratitude to Antonia Wenish, Helmut Hirsh, Andrea Walner who have prepared the expert opinion on EIA of the Belarusian Nuclear Power Plant by request of the Federal Ministry of Agriculture, Forestry, Ecology and Water Resources of Austria.

EIA of the Belarusian Nuclear Power Plant has been developed, in particular, on the basis of the following standard documents of the Republic of Belarus:

1. The Law of the Republic of Belarus dated November 26, 1992 «On Environment Protection»;
2. The Law of the Republic of Belarus dated June18, 1993 «On the State Ecological Assessment»;
3. Instruction № 30 on the order of carrying out of environment impact assessment of the planned economic and other activity in the Republic of Belarus confirmed by the Decision of the Ministry of Natural Resources and Environment Protection of the Republic of Belarus dated June 17, 2005;
4. Technical Code of the Standard Practice 099-2007 (02120/02300) "Location of Nuclear Power Plants. Manual on Development and Content of a Substantiation of Ecological Safety of Nuclear Power Plants" with regard to Annex 2 to «Convention on Environment Impact Assessment in Transboundary Context».

According to the standard documents EIA is being developed on the ground of the materials of the objects-analogues, therefore the replies to the questions concerning the technology of the concrete project of the Belarusian Nuclear Power Plant will be received at the stage of design works.

EIA of the Belarusian Nuclear Power Plant which has been finished taking into account the remarks received during carrying out of public discussions has been placed in the Global Network on the site of the Nuclear Power Plant Construction Directorate State Enterprise - dsae.by.

2 REPLIES TO THE QUESTIONS

1. Can you give more detailed explanations of the reasons of a choice of water-moderated water-cooled power reactors-1200 with a view to the available operational experience with the components and the systems, or, probably, there were other reasons?

In the world market the following projects of the nuclear power plants with PWR reactors are being offered:

- AP-600, AP-1000, the projects have not been implemented anywhere. There are serious claims to the project on the part of the regulating bodies of the United Kingdom of Great Britain and Northern Ireland;
- Project EPWR - France carries out construction of the first nuclear power plants for the last 15 years in Finland and in France, construction is being executed with serious backlog from the schedule;
- The Nuclear Power Plant-2006 Project. The Russian Federation is the only country which actively conducts construction of the Nuclear Power Plants with PWR-1000 reactors abroad within the last 10 years: China, India, Iran, and Bulgaria. Nuclear blocks on the Rostov Nuclear Power Plant have been put in operation in 2001 and on the Kalinin Nuclear Power Plant in 2005, "Temelin" Nuclear Power Plant in 2001and in 2002, the Tianwan Nuclear Power Plant in 2007. The closest prototype of the Nuclear Power Plant-2006 project has been commissioned in 2007 in the People's Republic of China (2 power blocks). Two power blocks in India are being completed as per the Russian projects of the third generation. Construction of two power blocks in Bulgaria and four power blocks in the Russian Federation began. In September of 2009 the Report on Termination of guarantee operation of the second power block of the Tianwan Nuclear Power Plant has been signed. Both power blocks operate stably at the level of capacity of 1060 MW, have high technical and economic and operational indicators.

2. What are the reasons of a choice of variant V-491 instead of V-392 M, does it mean that you prefer active but not passive safety systems?

«Nuclear Power Plant-2006 Project» concept as a basis makes use of two projects: Nuclear Power Plant-92 Project developed by Atomenergoproject Public Corporation , city of Moscow (RP V-392M) and Nuclear Power Plants-91/99 Project developed by St.-Petersburg Atomenergoproject Public Corporation, city of St. - Petersburg (RP V-491).

The choice of the type of a nuclear reactor and, accordingly, the general designer, has been carried out by the special State Commission by the results of estimation of a complex of indicators, the major of which were safety and reliability characterized by a set of parametres and factors. In fact, Nuclear Power Plant-92

Project developed by Atomenergoproject Public Corporation initially contains more systems of passive safety (which also has been considered by us at estimation).

Also in the course of estimation of the projects we considered the following indicators and criteria: referency of the project; technical data; ecological characteristics; economic characteristics; radioactive waste and spent fuel disposal; discharges and emissions from the Nuclear Power Plant; the general characteristic of the general layout and the basic structures; the extended characteristics of materials consumption of the project.

Taking into account all the criteria, the Project of development of St.-Petersburg Atomenergoproject Public Corporation, city of St. - Petersburg (RP V-491) has been chosen for implementation of construction of the Nuclear Power Plant in the territory of the Republic of Belarus.

3. The EFFICIENCY factor specified in the Report (more than 96 %) is very high. What was the basis for the given assumption?

It is not a matter of efficiency factor of the Nuclear Power Plant (EFFICIENCY) – (approximately 34 %), but a matter of the rated capacity duty factor (RCDF) - : design number of operation hours - 8400, the general annual number of operation hours - 8760, RCDF = 8400/8760 =95,8 %.

4. Can you give the description of a passive system of injection of high-pressure boron (project, drawing, operating characteristics)?

The passive part of the system of emergency cooling of a zone is intended for delivery in a reactor of boric acid solution with concentration of at least 16 g/dm^3 and temperature not less than 20ºC at a pressure in the first contour less than 5,9 MPa in a quantity sufficient for cooling of the active zone of a reactor before connection of the pumps of emergency injection of boric acid of low pressure in design-basis loss-of-coolant accidents.

The system consists of four independent channels with productivity of 50 % of each of them. In each channel one accumulator is being placed. Each accumulator is connected with the reactor by separate pipeline: two accumulators - with the front-end compartment of the reactor and two others - with the rear-end compartment of the reactor.

All the equipment of the system is located inside of the protective cover. Operation of the system is based on passive use of the energy of the compressed nitrogen, and for performance of safety functions (reflooding of the active zone) functioning of other systems is not required.

The drawings and operating characteristics will be submitted in the project.

5. What is the thickness of the walls (cylinder and dome) of the double protective cover of PWR-1200 reactors?

The Project provides for the constructive decision of the system of a sealed enclosure in the form of a double ferroconcrete cover. The space between the covers is connected to the ventilation system which provides for discharge and clearing of environment.

The thickness of the internal cover: a cylindrical part - 1200 mm, a spherical part - 1000 mm; the thickness of the external cover: a cylindrical part - 800 mm, a spherical part - 600 mm; a gap between covers - 1800 mm.

6. What are the characteristics of an air crash of the maximum force (weight of the plane, speed) which the reactor cover can sustain?

The weight of the plane - 5,7 tons, speed - 100 km/s.

7. Concerning external explosions. According to the Report, the maximum shock wave which the reactor cover can sustain appears to be low enough (10 kPa). On the other hand, in the literature higher figures have been specified. Which of these figures are true? What is specified in the specifications in the given concrete case?

The maximum shock wave which the cover can sustain: pressure 30 kPa, duration of impact- 1 second.

In TCP 170-2009 (02300) «General Provisions of Ensuring of Safety of Nuclear Power Plants» it is specified: «The systems and the elements important for safety should be capable to execute their functions in the volume established by the project taking into account influence of the natural phenomena (earthquakes, hurricanes, flooding possible around the Nuclear Power Plant site), the external technogenic events peculiar to the site chosen for construction of the Nuclear Power Plant, and/or possible mechanical, thermal, chemical and other impacts resulting in case of design-basis accidents» (point 7.6.1.).

8. How have the figures been received for the maximum loading at earthquake (numerical score, ground acceleration)?

The values have been received by means of calculation. Structural units of the buildings and facility are being designed with regard to maximum rated earthquake 0,12g - the maximum horizontal acceleration on a free ground surface (7 earthquake intensity as per scale MSK-64).

The equipment and the systems are being developed with regard to maximum rated earthquake 0,25g - the maximum horizontal acceleration on a free ground surface (8 earthquake intensity as per scale MSK-64).

9. Can you present the description of the device of localization of the fusion? Whether the tests of this device took place and if yes, what sort of tests? For example, what are the guarantees of possibility to avoid steam explosion?

The device of localization of fusion is intended for reduction of radiation consequences of serious accidents in which destruction of the active zone is being caused by its long drainage at low pressure in the first contour with the subsequent melting of the case of a reactor to safe level. Safety increase is being achieved at the expense of exception of discharge of liquid and solid radioactive materials outside the device of fusion localization which provides for avoidance of the damage of the system of the sealed enclosure of the zone of localization of accidents. The process of serious accident can be accompanied by not only destruction of the active zone and its fusion, but also by destruction of the case of a reactor. In these conditions a paramount task is preservation of integrity (strength and density) of the leak-tight cover which can be solved by means of the devices and the procedures being specially developed for control of serious accidents.

The basic functions which are carried out by the device of localization of melt:

- Holding of the bottom of the reactor vessel with corium at its separation or plastic deformation till the moment of escape of corium from the reactor vessel;
- Protection of the elements of a concrete mine design and leak-tight cover against thermomechanical influences of corium;
- Reception and placing in the internal volume of the liquid and solid components of corium of the fragments of the active zone and structural materials of a reactor;
- Steady heat transfer from corium to cooling water and the guaranteed cooling of corium melt;
- Prevention of corium escape outside the established boundaries of a zone of localization;
- Keeping of subcriticality of corium in a concrete mine;
- Minimization of carrying-over of radioactive substances in the space of a leak-tight cover;
- Minimization of hydrogen outlet;
- Non-excess of the maximum pressure in the structures located in the premises of a concrete mine at thermal actions in the course of out-of-design-basis accident, as well as at possible static and dynamic loadings;
- Ensuring of protection against destruction of the basic supporting structures of a reactor and dry protection at a stage of long-term cooling of corium.

Ensuring of execution of these functions is based on a principle of passivity without use of the active elements and regulating actions on the part of operating personnel within, at least, 72 hours from the beginning of a heavy phase of out-of-design-basis accident.

The minimum sufficient information of the system of melt localization is represented in EIA [1]. The tests of the system of melt localization have been held at the Tianwan Nuclear Power Plant in the People's Republic of China.

More detailed replies to the questions put by you will be submitted in the design documentation (architectural design) of the Belarusian Nuclear Power Plant.

10. Can you present the description and characteristics of a passive system of bleeding from steam-gas generators (design, drawing, operating characteristics)?

What role does the given system play in terms of long-term passive excess heat removal? What other systems exist for the given purpose? How has been proved reliability of their functioning?

At present the architectural design of the Belarusian Nuclear Power Plant is at the stage of development. The design will contain the drawings and operating characteristics of the system of passive heat removal from steam-gas generators. The project of technical requirements for the system of passive heat removal from steam-gas generators has been drawn up which will be without fail considered in the design of the Belarusian Nuclear Power Plant.

The system of passive heat removal from steam-gas generators is intended for active zone residual heat removal to a final absorber through the second contour at out-of-design-basis accidents.

The system carries out the following basic functions:

- residual heat removal and reactor shut-down cooling in the modes of complete de-energizing of the Nuclear Power Plant;
- residual heat removal and reactor shut-down cooling in the modes of complete loss of a feedwater;
- restriction of discharge of the radioactive coolant in the atmosphere through the fast reducing device (FRD-A) or steam-gas generator safety valves at the accidents with a leak of the coolant from the 1-st to the 2-nd contour at failure of design safety systems;
- Minimization of discharge of the radioactive coolant at the accidents with a leak from the 1-st to the 2-nd contour and steam line break in the non-cut part outside of a protective cover;
- ensuring of a reserve for the active systems of safety in case of their failure for emergency reactor shut-down cooling at the accidents with small and, partially, average leaks of the coolant of the first contour.

Productivity of the system has been chosen in terms of the conditions of the most probable scenarios of out-of-design-basis accidents being considered in the project and consists of four completely independent channels with productivity of 4×33,3 %.

The system consists of four independent channels connected to the vapour and water zones of the corresponding steam-gas generators.

Heat exchangers of the system of passive heat removal from steam-gas generators are intended for heat transfer from steam-gas generators to the tanks of emergency heat removal of the system which are located outside of a concrete cover of a reactor compartment in the circular rigging around its spherical part. The system heat exchangers are submerged under a water level in the tanks and are located above steam-gas generators which provides for natural circulation in a system contour.

Also there is a system of passive heat removal from a protective cover, which is intended for long-term (off-line operation – at least 24 hours) heat removal from a protective cover at out-of-design-basis accidents.

The system provides for decrease and keeping of pressure inside the protective cover within the limits set by the project and heat removal to a final absorber at out-of-design-basis accidents with serious damage of the active zone.

Productivity of the system has been chosen in terms of the conditions of the most probable scenarios of out-of-design-basis accidents being considered in the project, and consists of four completely independent channels with productivity of 4×33,3 %.

System functioning is based on passive principles.

Heat-exchange surface of each of four independent channels amounts to 300 m^2. Condensation heat exchangers are located over gantry rails on the containment wall.

Heat from containment is being removed at the cost of steam condensation on the internal condensation heat exchanger from which it is being transferred to the tanks of emergency heat removal by means of natural circulation of the coolant. The water volume of the tanks of emergency heat removal of each of four independent channels amounts to 405 m^3. Heat removal to a final absorber from the tanks of emergency heat removal is being carried out by water evaporation in the tanks within the first 24 hours from the beginning of the accident and their further feed at the cost of reserve water resources located on the site.

The system of passive heat removal from a protective cover enables to keep pressure under a cover in the whole spectrum of out-of-design-basis accidents connected with exit of mass and energy under a protective cover at a level below the rated one.

The data on reliability of functioning of the systems will be represented in the project.

11. Do the figures on probability of serious damages of the active zone and probability of maximum permissible discharge presented in the Report on water-moderated water-cooled power reactor-1200 cover all operating conditions of the nuclear power plant (full capacity loading, low power operation and shutdown), as well as all initiating factors (internal and external)?

The target probable indicators established for the power unit of the Nuclear Power Plant-2006 [2]:

- Decrease of probabilities of the accidents on the power unit with serious damage of the active zone of a reactor to the level of 10^{-6} 1/year.reactor and great discharges outside the territory of the site for which fast counter-measures outside the site are necessary with a level of 10^{-7} 1/year.reactor;
- Restriction of the maximum permissible discharge of the basic dose-forming nuclides to the environment at the serious out-of-design-basis accidents with probability of 10^{-7} 1/year.reactor with a level of 100 TBq of caesium-137.
- Decrease of maximum permissible discharge of the basic dose-forming nuclides to the environment at the serious out-of-design-basis accidents with probability of 10^{-7} 1/year.reactor, to the level at which:
    - Necessity of introduction of the immediate measures including both obligatory evacuation as well as long-term evacuation of the population outside the territory of the site; the nominal radius of a zone of planning of obligatory evacuation of the population does not exceed 800 m from the reactor compartment;

- Obligatory introduction of protective measures for the population (shelter, iodine prevention) is limited by a zone with a radius of maximum 3 km from the unit.

The given target probability indicators cover all the operating conditions of the Nuclear Power Plant as well as all the initiating factors. The specified indicators of the technical requirements to the project of the Belarusian Nuclear Power Plant are defined as the obligatory ones.

12. Unclear aspect is connected with probability of events. In particular, whether 95% quantile of probability of serious damages of the active zone and probability of maximum permissible discharge can be provided for?

The dose limits established for the Nuclear Power Plant-2006 power unit and target probability indicators completely meet the requirements of the valid Russian normative documents, the recommendations and safety norms of the International Atomic Energy Agency, the International Advisory Group on Nuclear Safety (INSAG1 - INSAG12) and to the requirements of the European exploiting organisations to the projects of the nuclear power plants of the new generation with reactors of the type PWR [3]. The Table represents for comparison the target indicators of radiation and nuclear safety of the power units with increased safety for various projects of the nuclear power plants and the requirement to them.

Table 1 – Indices of Nuclear and Radiation safety of the NPP

Criterion	EUR [1] INSAG-3 [7]	ND of RF [4,5]	Project of NPP-2006 [2]	Project USA-APWR [6]
Quotas of population irradiation from discharge at normal operation of the NPP, μSv/year	Is not being regulated	50(50)	10(10)	-
Quotas of population irradiation from discharge at normal operation with regard to breaks of normal operation of the NPP, μSv/year	100	Is not being regulated	100	100
Effective dose for the population at design-basis accidents, μSv/event - with a frequency of more than 10^{-4} 1/year - with a frequency of less than 10^{-4} 1/year	 1 5	Is not being regulated	 1 5	 1 5
Effective dose for the population at design-basis accidents, μSv/year	-	5	5	-
Probability of serious damage of the active zone, 1/year.reactor	1E-5	1E-5	1E-6	1E-6
Probability of serious discharge for which fast countermeasures outside the site are necessary, 1/year.reactor	1E-6	1E-7	1E-7	1E-7

The probabilistic analysis within the scope of the requirements [2-7] will be carried out in the course of development of the project of the Belarus Nuclear Power Plant and represented in the corresponding section of the design documentation.

13. The Report affirms that the Nuclear Power Plant-2006 corresponds to the requirements of EUR. Can you submit the additional information on the given problem? In particular, on the source of discharge which, how it is supposed, meets the requirements of « Criteria on the Limited Impact»?

The verification procedure for blocks PWR of the increased safety offered by EUR enables to connect the predicted emergency ground and high-altitude discharges of the certain list of radiation-significant nuclides with the necessity of introduction of protective measures outside of the industrial site irrespective of the conditions of localization of the site. The results of the verification procedures for out-of-design-basis accident with maximum permissible discharge at the Baltic Nuclear Power Plant (is the object-analogue) are presented in Table 2. Consideration has been carried out for the rated emergency discharges; the calculations cover the radionuclides which form by more than 90% a predicted dose of irradiation.

Table 2 – Results of Verification Procedure Recommended by EUR for NPP-2006

Name of Criterion	Maximum value [EUR]	Design value for NPP-2006
Out-of-design-basis accidents (frequency less than 10^{-6} 1/year.reactor)		
Criterion B1 – restriction on introduction of emergency protective measures at distances from the reactor of more than 800 m	$< 5\cdot10^{-2}$	$1{,}2\cdot10^{-2}$
Criterion B2 – restriction on introduction of delayed protective measures at distances from the reactor of more than 3 km	$< 3\cdot10^{-2}$	$1\cdot10^{-3}$
Criterion B3 – restriction on introduction of long-term protective measures at distances from the reactor of more than 800 m	$< 1\cdot10^{-1}$	$1\cdot10^{-2}$

It follows from the Table 2 data that the maximum permissible discharge of the Nuclear Power Plant-2006 accepted for radiationt-significant nuclides reliably meets the requirements of acceptance criteria of verification procedure which additionally confirms observance by the Baltic Nuclear Power Plant (is the object-analogue) of the following purposes:

- To exclude necessity of introduction of emergency evacuation and long-term evacuation of the population outside of the territory of the Nuclear Power Plant site;
- To limit a zone of planning of obligatory protective measures (population shelter, iodine prevention) for the population to the radius 3 km maximum.

The estimation of the limited impact on the economy has been carried out by comparison of the sum of discharge at ground level and high-altitude discharges during the accident with criteria as per EUR. The initial data for such comparison are presented in the Table.

Table 3 – Observance of Criteria of Limited Impact on Economics for the Baltic NPP

Radionuclide	Criterion as per EUR, TBq	Values of MPD for the Baltic NPP, TBq
Out-of-design-basis accidents (frequency less than 10^{-6} 1/year.reactor)		
^{131}I	4000	100
^{137}Cs	30	10
^{90}Sr	400	0,12

From consideration of the data presented above the additional confirmation follows that the criteria of ecological safety of EUR for the Baltic Nucler Power Plant (is the object-analogue) are being observed. Thus it is possible to make a conclusion that the set of the active and passive systems of safety being applied in the project of the Baltic Nucler Power Plant completely provides for observance of the requirements of the ecological safety of EUR.

Since the verification procedure of EUR is the comparison of the criteria received as a result of multiplication of the value of the maximum permissible discharge of nine reference isotope groups by the standardized coefficients with the criteria accepted by EUR, the resulted conclusions are completely applicable also for the Belarusian Nuclear Power Plant.

14. Can you tell in more details about the requirements which are being lodged to the nuclear installation (besides EUR)?

The concrete requirements to the nuclear installation are listed in the Technical Codes of the Standard Practice of the Republic of Belarus 170-2009 (02300) «General Provisions of Ensuring Safety of Nuclear Power Plants» and 171-2009 (02300) «Rules of Nuclear Safety of Reactor Installations of Nuclear Power Plants».

The above-mentioned documents establish that safety of the Nucler Power Plant should should be provided for at the cost of consecutive implementation of the concept of deep-echelon protection based on use of the system of physical barriers on the way of distribution of ionizing radiation and radioactive substances in the environment and the systems of technical and organizational measures on protection of the barriers and preservation of their efficiency, as well as on protection of the personnel, the population and the environment.

The Nucler Power Plant project should provide for technique and the organizational measures directed at prevention of the design-basis accidents and restriction of their consequences and providing for safety at any of the initial event being considered by the project with application according to the principle of a single failure of one failure independent of the initial event of the following elements of the systems of safety: of an active element or the passive element which have mechanical moving parts, or one error of the personnel independent of the initial event.

According to the concept of a deep-echelon protection, the Nucler Power Plant should have the systems of safety intended for execution of the following basic functions of safety: emergency shutdown of a reactor and its keeping in subcritical state; emergency heat removal from a reactor; keeping of radioactive substances in the established boundaries.

The Project of the Nucler Power Plant, the work paper of the systems and the elements important for safety should define, and for the safety systems and elements and the elements important for safety related to classes of safety 1 and 2, should be ready and checked prior to the beginning of physical start-up, adaptations and devices, as well as the programs and techniques designated for check up: of serviceability of the systems and the elements (including the devices located in a reactor), replacement of the equipment which has worked out its resource; tests of the systems for conformity to the design indicators; check of sequence of passage of signals and switching on of the equipment (including transfer to the emergency power sources); control of a state of metal and welded connections of the equipment and pipelines; check of metrological characteristics of the measuring channels for conformity to the design requirements.

The Nucler Power Plant project should provide for the means which help to exclude individual errors of the personnel or to decrease their consequences, including those in the course of maintenance.

The safety systems should function so that their action will be performed till complete execution of their function. Returning of the system of safety to the initial condition should demand consecutive actions of the operator.

The active zone and other systems which define the operating conditions of the Nucler Power Plant should be designed so that to exclude excess of the established limits of safe operation of fuel elements damage throughout the term of use established for them. Excess of the specified limits also is not supposed at any of the following preliminary situations (taking into account action of the protective systems): any single failures in the control systems of a reactor installation; loss of power supply of the main circulating pumps; switching-off of turbogenerators and heat consumers; loss of all the sources of power supply of the normal operation; leaks of a contour of the reactor coolant being compensated by the charge circuits of the normal operation; a malfunction of one of the safety valves.

The active zone together with all its elements which influence on reactivity should be designed so that any changes of reactivity by means of the regulating units and the effects of reactivity in the operational conditions and at design-basis and out-of-design-basis accidents will not cause uncontrollable growth of energy release in the active zone which leads to the fuel elements damage beyond the established design limits.

All the equipment and pipelines of a reactor coolant contour should sustain without damage any static and dynamic loadings and thermal effects arising in any of its units and components, at all the initial events being considered, including indeliberate energy release to the coolant caused by: sudden introduction of positive reactivity at discharge of impact element on peak efficiency reactivity with the maximum speed if such discharge is not prevented by a design; input of the "cold" coolant to the active zone (at negative temperature factor of reactivity on the coolant) or by any other possible positive effect of reactivity connected with the coolant.

The Nucler Power Plant block should provide for the following systems of safety:

1. Control safety systems (CSS). CCS should carry out their functions automatically at occurrence of the conditions stipulated by the project. CSS should be designed so that at automatic start possibility of their switching-off by the operating personnel will be blocked within 10 - 30 minutes. CSS should be designed so that the started action will be performed till complete execution of their functions. Returning of the system of safety in its initial condition should demand consecutive actions of the operator.

2. Protective systems of safety. The Nucler Power Plant project should provide for the protective systems of safety providing for reliable emergency shutdown of a reactor and its keeping in a subcritical condition at any modes of normal operation and infringements of normal operation, including design-basis accidents. The efficiency and speed of the systems of emergency shutdown of a reactor should be sufficient for restriction of energy release by the level which does not lead to the fuel elements damage beyond the established limits for normal operation or design-basis accidents and suppression of the positive reactivity which appears as a result of display of any effect of reactivity or a possible combination of the effects of reactivity at normal operation and design-basis accidents. Emergency shutdown of a reactor should be provided for irrespective of the fact wheter there is the energy source or it has been lost.

3. Localizing systems of safety. Localizing systems of safety for keeping of radioactive substances and ionizing radiation in the course of accidents within the limits stipulated by the project should be provided for. The reactor and the systems and the elements of the Nuclear Power Plant which contain radioactive substances should be placed in airtight premises entirely for localization of radioactive substances being discharged at design-basis accidents within their boundaries. Thus, and also in case of other localization, it is necessary that at normal operation and design-basis accidents the corresponding established doses of irradiation of the personnel and the population, as well as the standards on discharge and content of radioactive substances in the environment will not exceed the standard levels. The necessity and admissibility of the directed discharge of radioactive substances at out-of-design-basis accidents should be grounded by the project. The localizing systems of safety should be provided for each block of the Nuclera Power Plant.

4. Secure systems of safety. The Nucler Power Plant project should provide for the necessary secure safety systems which carry out the functions of supply of the safety systems with an operating environment, energy and creations of the necessary conditions of their functioning, including heat transfer to a final absorber. Secure safety systems should have the indicators of reliability of performance of the set functions sufficient for possibility to achieve the necessary reliability of functioning of the last being defined in the project together with the indicators of reliability of the safety systems which they provide for. Performance of the specified functions by the secure safety systems should have an unconditional priority over the action of internal protection elements of the secure safety systems if it does not lead to heavier consequences for safety; the list of the internal protections of the elements of the secure safety systems which are not subject to disconnection should be grounded in the Nucler Power Plant project. The Nucler Power Plant project should provide for necessary and sufficient means for fire protection of the Nucler Power Plant, including sensors and burning suppressions of the inhibitor and the coolant. The Nucler Power Plant project should provide for the automated operating mode of the systems of fire control from the moment of voltage supply on the equipment of the block of the Nucler Power Plant in the course of carrying out prestarting adjustment works. Automatic protection of a reactor should have at least two independent groups of actuators.

15. Whence the data on characteristics of a source of discharge presented in the Report have been taken? Why more considerable figures of discharge are not being analyzed?

The data on the characteristics of a source of discharge have been taken from the analysis of the following materials:

1. The Khmelnitskaya Nucler Power Plant, power unit 2. Estimation of Environmental Impact, Energoproject CIEP, 43-915.201.012. OB13.
2. The Report on EIA of the New Nucler Power Plant in Lithuania dated August 21, 2008, NNPP_EIAR_D2_Combined_Ru_200808_FINAL.
3. The Nizhniy Novgorod Nucler Power Plant. Power units № 1 and 2. A preliminary variant of the materials on environmental impact assessment. Concern Energoatom Production and Commercial Firm, 2009.
4. Nucler Power Plant-2006. Grounds for Investments into Construction of the Leningrad Nucler Power Plant-2. Volume 5. Environment Impact Assessment. St.PbAEP Public Corporation.
5. The Nucler Power Plant-2006. Grounds for Investment into Construction of the Baltic Nucler Power Plant. Volume 5. Environment Impact Assessment. St.PbAEP Public Corporation.
6. The Report on substantiation of safety of the Tianwan Nucler Power Plant -2, Chapter 15 . Analysis of Accidents, Book 4. St.PbAEP Public Corporation.
7. The Preliminary Report on Substantiation of Safety of the Balakovskaya Nucler Power Plant. Power unit 5, 29.11.04, Version 0.

8. The Novovoronezhskaya Nucler Power Plant-2 with power units № 1 and № 2. Section 4.8. Radiation Protection. Atomenergoproject Public Corporation. Amendment 2. 25.08.08.
9. Information on the Accident at the Chernobyl Nucler Power Plant and its Consequences Prepared for the International Atomic Energy Agency. Abagyan A.A., Asmolov V.G, Gus'kova A.K. etc. Atomic Energy. V.61, Issue 5, November of 1986.

The amount of discharge of the reference isotopes iodine-131 = 3,1 E+15 and caesium-137=3,5E+14 to the environment has been chosen on the following basis: at out-of-design-basis accidents the integrity of a protective cover is being retained for at least 24 hours, leakings through the containment - 0,2 % per 24 hours and discharge lapses in a 24 hours period. Thus, as a result of an out-of-design accident the following elements have been thrown to the containment:

Iodine - 131: 3,1 E+15: 0,002 = 1,55 E+18;
Caesium - 137: 3,5E+14: 0,002 = 1,75 E+17.

The given values of activity of the reference isotopes properly co-ordinate with the emergency discharge of the Chernobyl Nucler Power Plant (iodine 131 = 2,7 E+17 Bq, caesium 137 = 3,7E+16 Bq).

16. What figures of discharge represent the most serious scenarios and what are the maximum permissible discharges?

The Nucler Power Plant-2006 project establishes the maximum permissible discharge with regard to the achieved level of safety for a class of serious accidents on the block [8]:

- For the early phase of the accident connected with leaks of radioactive substances through thinnesses of a double protective cover and bypass of the containment, in absence of power supply on the block: xenon-133 - 10^4 TBq; iodine-131 - 50 TBq; caesium-137 - 5 TBq.
- For the intermediate phase of the accident, after power supply restoration on the block, connected with discharge through a ventilation pipe: xenon-133 - 10^5 TBq; iodine-131-50 TBq; caesium-137 - 5 TBq.

For estimation of the maximum permissible discharge the analysis of radiation consequences of a reference scenario of the serious accidents connected with slow growth of pressure in the containment (total probability approximately 10^{-7} 1/year.reactor) according to the recommendations of the IAEA for the Nucler Power Plant with PWR [9] has been carried out. Within a framework of the Report the maximum permissible discharge has been used for preliminary estimation of the scope of protective measures for the population at serious accidents on the power unit.

Table 4 represents the rated values of the maximum permissible discharge and the requirement to them established in various countries and the projects for comparison. Implementation of the planned strategy in the projects has lowered the rated levels of the maximum permissible discharge grounded according to the requirements specified above.

Table 4 – Maximum Permissible Discharge and Requirements to them, TBq

Dose-forming nuclide	Requirements to location of the NPP, USSR, year 1987	Requirement of the Resolution of the Council of State of Finland 395/91	Tianwan NPP [10]	Project of NPP-2006 [8]	USA-APWR [6]
Xenon-133	Is not being regulated	Is not being regulated	10^6	10^5	3.10^5
Iodine-131	Maximum 1000	Is not being regulated	600	100	349
Cesium-137	Maximum 100	Maximum 100	50	10	5,6
Strontium-90	Is not being regulated	Is not being regulated	1	0,12	0,15

17. Are the authors of the Report on EIA aware of the results of preliminary reports on safety at the Leningradskaya Nuclear Power Plant-2 and the Novovoronezhskaya Nuclear Power Plant-2 (Nuclear Power Plant- (Water-moderated water-cooled power reactor-1200/491)) which are at a stage of construction?

Yes. In the course of preparation of the materials on EIA the following materials on the objects-analogues have been studied and used:

1. The Nuclear Power Plant-2006. Substantiation of the Investments into Construction of the Leningradskaya Nuclear Power Plant-2. Volume 5. Environment Impact Assessment. St.PbAEP Public Corporation.
2. The Novovoronezhskaya Nuclear Power Plant-2 with power units № 1 and № 2. Section 4.8. Radiation Protection. Atomenergoproject Public Corporation. Amendment 2. 25.08.08.
3. The Nizhniy Novgorod Nuclear Power Plant. Power units № 1 and 2. A Preliminary Variant of the Materials on Environment Impact Assessment. Concern Energoatom Production and Commercial Firm, 2009.

18. What scenarios on the maximum design-basis accidents and out-of-design-basis accidents have been analyzed by the designers of the Nuclear Power Plant?

For objectivity of the Report the consequences of the most serious out-of-design-basis accident have been considered. Among four types of out-of-design-basis accidents the most serious consequences, from the point of view of the radiation damage result in out-of-design-basis accidents of the third type. In this case due to complete de-energizing of the Nuclear Power Plant cooling of the active zone of a reactor stops. It leads to serious damages of the nuclear fuel, but the protective cover keeps its tightness. As per the 7-level scale accepted by the IAEA such accident has the fifth level of severity. Namely at such accident the maximum possible discharge of caesium-137 of all the types of out-of-design-basis accidents takes place, and the total intensity of discharge is approximately by 80 times more than that at the maximum design-basis accident. Discharge of radioactive substances at the accident would proceed about 24 hours [11].

19. Can you describe the measures on control of the nuclear reactor accidents and the corresponding measures which can provide for the least discharge in case of out-of-design-basis accident?

The analysis of the reference out-of-design-basis accident at Nuclear Power Plant-2006 (the Nuclear Power Plant-92 project) is presented in [12]. The basic purpose of ensuring safety of the Nuclear Power Plant at out-of-design-basis accident consists in achievement and maintenance of a safe state of the Nuclear Power Plant (Servere Accident Safe State) at serious accident not later than within 7 days in one week from the accident beginning. For this purpose it is necessary to carry out the following conditions:

- The fragments of an active zone are in a solid phase, and their temperature is stable or decreases;
- Heat release of the fragments of the active zone is being removed and transferred to a final absorber of heat, the configuration of the fragments is such that efficiency factor is much more lower than 1;
- Pressure in the zone of a protective cover is so low that in case of loss of sealing of the protective cover the criterion of restriction of radiation consequences for the population is being observed;
- The outlet of fission products in the zone of a protective cover has stopped.

For ensuring of integrity and tightness of a design of a protective cover at serious out-of-design-basis accidents the project provides for:

- Prevention of early damage of the internal protective cover;
- Prevention of late failure of the protective cover at the cost of the corresponding measures, such as:

- Ensuring of heat removal and localization of melt in a trap, exclusion of direct impact of a melt on a protective cover, the base, concrete of reactor mine;
- Prevention of accumulation of potentially dangerous concentration of hydrogen.

The initial events of the reference out-of-design-basis accident are as follows:

- Break of the basic circulating pipeline Du 850 in the input of the reactor with bilateral blowdown;
- Loss of the sources of an alternating current and, accordingly, nonserviceability of all the active safety systems for the long period of more than 24 hours, failure of start of all diesel- generator sets; emergency supply is being carried out from the storage batteries.

Dynamics of development of the serious out-of-design-basis accident is presented in Table 5.

Table 5 – Developmemt of a Serious Out-of-Design-basis Accident

Event	Time	Comment
Break of the reactor coolant pipe PD 850 on outlet of the reactor. Loss of all the sources of AC	0,0 s	Initial event
Deactivation of all the reactor coolant pipes . Deactivation of the system of infeed-blowdown. Prohibition on switching on of fast reducing devices of steam dumping FRD-C	0,0 s	Application of failure: loss of all the sources of AC of the NPP including all the diesel generators
Actuation of an emergency protection system	1,9 s	By the fact of de-energizing of the block with delay of 1,9 s
Start of work of the accumulator of the system of emergency cooling of the active zone	8,0 s	Decrease of pressure of the first contour below 5,9 MPa
Start of the system of passive heat removal	30,0 s	By the fact of de-energizing on the section of safe power supply with delay of 30 s
Loss of borated water supply from the accumulator of the system of emergency cooling of the active zone	144,0 s	Decrease of the level in the tanks of accumulator of the system of emergency cooling of the active zone till the mark of 1,2 m
Start of steam condensation in the pipe heater of the steam generator	3600,0s	Parameters of the second contour are lower than those of the first contour
Start of hydrogen generation in the active zone at the cost of the oxidation reaction	44,6 h	T of fuel elements > 1000 ^{0}C
Decay of the active zone and start of accumulation of the decayed materials of the active zone and vessel internals in the lower mixing chamber	47,7 h	
Melting of the support grid in the lower mixing chamber and accumulation of the parts of the active zone on the bottom of the reactor vessel.	51,0 h	T of the support grid > 1500 ^{0}C
Decay of the reactor vessel and start of escape of the melt in the melt localization device	52,0 h	T of the case > 1500 ^{0}C

For the purpose of minimization of the consequences of a serious out-of-design-basis accident the following systems are being applied:

- The system of heat removal from the hermetic casing (sprinkler system);
- The system of emergency and planned shut-down cooling of the first contour;
- The system of control of concentration and emergency removal of hydrogen;
- The system of catching and cooling of the fused active zone out of a reactor.

The purposes being achieved at operation of the given systems of safety are represented in Table 6.

Table 6 – Result of Operation of Safety Systems at Control of Out-of-Design-Basis Accident

Safety System	Period of Operation	Achievable Purpose
System of hydrogen emergency removal	Within the whole period of an accident	Ensuring of hydrogen nonexplosiveness
System of passive heat removal. System of accumulators of the second grade	Before transfer to the heavy stage	Prevention of the early damage of the protective cover. Ensuring of heat removal from the protective cover and fuel.
System of collection and cooling of the molten active zone	After decay of the reactor vessel and transfer of the accident to the out-of-vessel stage	Achievement of the safe state of the NPP (SASS). Provision of heat removal and localization of a melt in a trap. Termination of fission products outlet to the protective cover zone.
Sprinkler system. System of emergency and design shutdown cooling of the first contour	In three days after beginning of the accident	Achievement of the safe state of the NPP (SASS). Decrease of pressure in the zone of the protective cover. Provision of heat removal from the protective cover and fuel. Prevention of late failure of the protective cover.

Consideration of the list of out-of-design-basis accidents, the scenarios of development and their consequence serve for working out of the guidance on control of the out-of-design-basis accidents and drawing up of the plans of the measures on protection of the personnel and the population in case of these accidents. The final lists of out-of-design-basis accidents, their realistic analysis which contains estimation of probabilities of the ways of behaviour of out-of-design-basis accidents are being established in the project of the Nuclear Power Plant and in the Report on substantiation of safety of the Nuclear Power Plant. The given documents will be developed at the subsequent stages of designing of the Belarusian Nuclear Power Plant.

20. What levels of radioactivity do you use for classification of radioactive waste (high, average, low)?

Classification of solid and liquid radioactive waste by degree of their activity or radiation impact is being carried out according to criteria [13 - 15] which are represented in Table 7.

Table 7 – Classification of Solid and Liquid Radioactive Waste on Specific Activity

Category of Waste	Radiation level, mSv/h	Specific Activity, kBq/kg		
	Gamma-emitting	Beta-emitting	Alpha-emitting (without transurans)	Transuranium
Low-activity	from 10^{-3} to 0,3	Less than 10^{3}	Less than 10^{2}	Less than 10
Medium-activity	from 0,3 to 10	from 10^{3} to 10^{7}	from 10^{2} to 10^{6}	from 10 to 10^{5}
High-activity	More than 10	More than 10^{7}	More than 10^{6}	More than 10^{5}

The additional classification of solid radioactive waste recommended [13, 15] and practiced at operation in respect of solid waste is their classification by the levels of capacity of a dose of gamma radiation at a distance of 0, 1 m from a surface:

- low-activity - from 1 µSv/h to 300 µSv/h;
- medium-activity - from 0,3 µSv/h to 10 µSv/h;
- high-activity - more than 10 µSv/h.

21. Are there any plans of construction of intermediate warehouses for the spent fuel?

No. The spent nuclear fuel being unloaded from a reactor is being stored in the cooling pond (storage at least three years for activity and residual heat release decay) located in a reactor building. The capacity of a cooling pond provides for storage of the spent nuclear fuel within ten years, including placing defective fuel assemblies in hermetic containers, as well as the possibility of unloading of the whole active zone of a reactor at any moment of Nuclear Power Plant operation. In the course of unloading of a reactor export of the exposed spent nuclear fuel from the Nuclear Power Plant site to the factory of fuel regeneration of the Russian Federation is being carried out.

22. Is construction of a place of active nuclear waste utilization in the Republic of Belarus being planned?

In the Republic of Belarus construction of the regional centre for storage of the radioactive waste being formed as a result of use of nuclear technologies in various spheres of human vital activity, including in nuclear power engineering, is being planned.

The spent nuclear fuel does not relate to radioactive waste and will be returned to the Russian Federation for reprocessing.

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3 LIST OF ABBREVIATIONS

EIA	-	Environment Impact Assessment
NPP	-	Nuclear Power Plant
WMWCPR	-	Water-moderated Water-cooled Power Reactor
RP	-	Reactor Plant
EF	-	Efficiency Factor
CF	-	Capacity Factor
TCSP	-	Technical Code of Standard Practice
MPD	-	Maximum Permissible Discharge
ND	-	Normative Documents
FA	-	Fuel Assembly
FE	-	Fuel Element
ODBA	-	Out-of-Design-Basis Accident
PD	-	Passage Diameter
FRD-A	-	Fast Reducing Device of Vapour Escape in Atmosphere
FRD-C	-	Fast Reducing Device of Steam Dumping
SF	-	Spent Fuel

4 LIST OF LITERATURE

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I hereby certify the authenticity of the translation. Translator V.P.Komarova