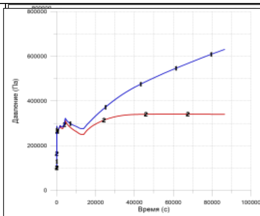
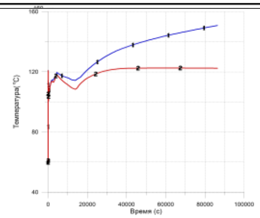
Unique Number	Description	Answers of the departments concerned
G-1	Are periodic safety reviews performed in Belarus, according to national regulations in force? If yes, with which periodicity and what is the detailed scope / content?	Clause 1.2 "National Requirements and Regulations in the Sphere of Nuclear and Radiation Safety" shall be supplemented by the following: Resolution of the Council of Ministers of the Republic of Belarus of December 7, 2010 No. 1781 has approved the regulation on the procedure for the examination of documents substantiating the provision of nuclear and radiation safety in the implementation of activities in the field of the use of atomic energy and sources of ionizing radiation. In compliance with the above mentioned regulation Gosatomnadzor has organized 5 (five) safety examinations at all stages of the NPP licensing. Safety examinations are carried out both at the request of the Operator to modify the valid license and when required by Gosatomnadzor. The scope of a safety examination is determined by Gosatomnadzor when developing and approving the technical assignment for the examination.
G-2	The "List of Abbreviations" is incomplete. Please prepare a list with all abbreviations used in the report except they are used as proper names.	The list of the abbreviations has been extended. See Attachment.
G-3	Translation failed: DBE isn't equal to Russian ПЗ. DBE is equal to Russian МРЗ. Russian ПЗ. DBE should be read as OBE always in the text DBE=OBE!	The comment is accepted. In the European terminology a seismic impact of the DBE level (Design Basis Earthquake) corresponds to the Russian "МРЗ" (safe shutdown earthquake). The Russian "ПЗ" corresponds to the term "OBE" (operating basis earthquake). The National Report shall be modified accordingly.
G-4	TKP 566-2015 "evaluation of the frequency of severe damage to the reactor core (for external source of natural and man-made events)" Q1) Exists an English version of this paper? Q2) Could you hand over the English version to the PRT.	There is no English translation of TKP 566-2015 "Evaluation of the frequency of severe damage to the reactor core (for external source of natural and man-made events)" approved by Decree No. 21 dd. April 28, 2015 of the Ministry of Emergency Situations of the Republic of Belarus.
G-5	What is the content of the document "requirements to stress tests (objective safety reassessment) of a nuclear power plant" and exists an English version of this document? Could your hand over the English version to the PRT?	The norms and regulations for ensuring nuclear and radiation safety "The requirements for carrying out stress tests (targeted reassessment of safety) of the nuclear power plant" lay down requirements for the NPP stress-tests at all stages of the NPP life cycle and define the scope of the information to be provided following the results of the stress-tests. The scope of application of the requirements for the stress tests of the NPP is determined by the external initiating events happened at the Fukushima-1 NPP, including their combinations, taking into account the Declaration of ENSREG, Annex 1 "EU "Stress-test" specifications" and the requirements of the "Post-Fukushima "stress tests" of European nuclear power plants – contents and format of National Reports." There is no English translation of the document in question.
G-6	Q1) How did you take into account the results of the European stress tests in 2011 – 2012? The most important outcomes of these stress tests are listed in the ENSREG document "Compilation of Recommendations and Suggestions" of 26/07/2012 - http://www.ensreg.eu/sites/default/files/Compilation%20of%20RecommendationsI_0.pdf. - Q2) Can you hand over a list which gives information to the PRT, which of these recommendations and suggestions have been implemented or will be implemented in the Belarus NPP? And if certain recommendations or suggestions have not been or will not be implemented please explain shortly why.	The results of the European stress tests of 2011-2012 contained in document "Compilation of Recommendations and Suggestions" of 26/07/2012 prepared by the ENSREG have been taken into account when developing stress tests for the Belarusian NPP. The following topics have been considered: 1. Containment integrity. 2. Prevention of accidents resulting from natural hazards and limiting their consequences; 3. Assessment of natural hazards and margins. Also, detailed information is contained in the final Report of the mission of SEED of IAEA; 4. Loss of safety systems 5. Severe accident management. The relevant conclusions based on the results of the stress tests are made in the National Report in Section 8.
G-7	The recurrence of postulated accidents (class 1 and 2) can be defined by probabilistic methods. The hazards are obviously characterised by deterministic methods. How are the assigned recurrence intervals for postulated accidents due to earthquake defined?	In compliance with the regulatory requirements (NP-031-01) the postulated earthquakes are characterized by the following recurrence intervals: DBE 1 (once) every 10000 years, OBE: 1 (once) every 1000 years. These recurrence intervals are assumed for the accidents caused by an earthquake.
G-8	The spent fuel pool is adjacent directly to the reactor cavity and is connected with it via the canal for FA supplying. (open canal with the same water level in the cavity and the fuel pool) Question: Is the bottom of this canal higher than the top of the fuel racks in the spent fuel pool? If yes, how many meters?	The elevation of the bottom of the canal (transport corridor) is above the elevation of the upper part of the racks of the spent fuel pool. The upper elevation of the racks of the spent fuel pool is: +13.500. The height of the rack is 4.52 m. The height from the bottom of the spent fuel pool up to the transport corridor is 9 m. The height from the rack up to the transport corridor is 4.48 m. The National Report does not contain this information. This information is given in the Report on the stress-tests /31/ in Figure 5.1.2.22. Overall dimensions of the spent fuel pool.
G-9	The fuel is stored in the spent fuel pool under protective water layer with boric acid concentration 16-20 g/dm3 . Question: Are the fuel racks designed to keep the fresh fuel assembly with an enrichment of 5% (max. criticality) subcritical even if the spent fuel pool is filled with pure water?	The fuel storage racks are designed with due consideration of the following requirement of NP-061-05: "subcriticality of at least 0.05 (K_{eff} value <0.95) must be ensured in the racks of the spent fuel pool when there is no boric acid in the coolant", which is confirmed by Report of National Research Center "Kurchatov Institute" 2006.P.131.&.00UJA&.00JKA.022.RE.0001 "Design analysis of criticality during storage and transportation of uranium-based fuel at an NPP."
G-10	Which precaution are designed into the fuel pool systems to keep the stored fuel assemblies covered with water? For example: no penetration of the fuel pool walls with pipes below the water level? Was the siphon effect taken into account for pipe breaks connected with the fuel pool (Vacuum braker, Check valves in the piping)?	The pipelines below the water level are equipped with a passive siphon breaker which excludes emptying of the spent fuel pool. Penetrations of the fuel pool walls with pipes are made in a way which excludes emptying of the spent fuel pool in case of a pipe rupture.
G-11	Page 39 and table 3.1.2.2 page 55: The report states that there is no impact of earthquakes on the mobile emergency diesel generators. Please clarify the storage conditions of the 500 kW mobile emergency diesel generators. In which building are they located? To which seismic level is this building qualified? Please also clarify the storage conditions of the emergency mobile pumps (fire trucks). In which building are they located? To which seismic level is this building qualified?	The mobile diesel generator station of the BDBA management system is located on the Unit site outside the buildings and structures on an open concrete pad in an unobstructed area.

G-12	Following schemes and figures and deteailed descriptions are needed for an effective review of the national Report: 1. Overall layout of NPP demonstrating all main facilities; 2. overall technological scheme of power supply (electrical connections and transmissions within the unit); 3. overall technological scheme of ultimate heat sink systems (with the respective parts PA, PE, PC, GA, GH including detailed descriptions of the heat removal chains from the recator as well as from the spent fuel pool; 4. technological scheme of DGs cooling system. Descriptions of the technical components in the containment (figure 2.3.3.1,; p28) as well as a detailed description of the technical components of the safety systems (figure 2.3.3.4; p.33) are miising and needed. Flow diagrams (P&I diagrams) of the safety systems are needed. A plan with the building positions containing the decribed operational and safety systems is missing.	The requested information is given in the SAR. If necessary, this information can be submitted to PRT experts for review within the period from 12.03.2018 to 16.03.2018. Regarding item 1 "Overall layout of NPP demonstrating all main facilities" can be given to PRT within the period of 12-16.03.2018. Regarding item 2 The main wiring diagram and the diagram of the auxiliary power supply are given in file G-12.pdf. Regarding item 3, the overall technological scheme of ultimate heat sink systems can be submitted additionally (separately). Regarding item 4, the technological scheme of the DGs cooling system is contained in file "РДЭС_БДЭС схемы систем охлаждения ДГУ.pdf" (SDPP_UDPS schemes of the DGs cooling system). 5. Figures 2.3.3.1 and 2.3.3.4 shall be supplemented by a description of the technical components. 6. We do not think it is necessary to include into this Report the flow diagrams (P&I diagrams) of the safety systems. In our opinion this documentation can be submitted as supplementary documentation (separately).
G-13	What is the water volume in the primary circuit?	During the operation at power the volume of water in the primary circuit (pressurizer including) is 350 m3.
G-14	The Design provides for a spacial separation of the safety system channels and channel structural protection thus excluding the possibility of common cause failures (due to fire, flooding). More information about separation principles used in Belarussian NPP design is needed.	The separation principle is one of the fundamental principles in designing of a safety system. Application of the separation principle with respect to the Belarusian NPP is described in detail in the Safety Concept (Chapter 1, SAR).
G-15	What comprises "Biological Protection" in the frame of the "barrier system"?	See chapter 11.3.3 SAR. In line with the definition contained in NP-001-15, the biological protection is a set of barriers, including construction structures, designed for protection against the ionizing irradiation. As part of the system of barriers, the biological protection is construction structures that mitigate impact of the ionizing irradiation on the personnel and population.
G-16	What does the phrase " limiting release of radioactive substances into the environment" mean in the context of level 2 of the DiD concept? Normally, at this stage of the DiD the "barrier system" for retention of radioactive substances is fully functioning.	At the level 2 of DiD, in the modes of deviation from the normal operation the design sets forth additional target criteria to limit the radiation impact on the personnel and population below the upper limits established for the normal operation. In line with i. 1.4.4.2 of the Technical Assignment for the Belarusian NPP, in the modes of deviation from normal operation a target limit is established for the annual exposure of the population from gaseous and aerosol releases; in compliance with requirements for the modern European NPP Projects (NPP Khankhikivi, NPP Paks-2, etc.) and EUR recommendations, it is equal to 0.1 mSv on one occurrence.
G-17	More information is needed about containment separation device mentioned in page 28	In the normal operation mode, the containment separation device prevents from a flow of the "contaminated" air from the steam generator boxes to the reactor central hall thus allowing access of the staff to the central hall when operating at power without using the personal respiratory protective equipment.
G-18	The on-site storage facilities for spent nuclear fuel are not available (page 29). What are the measures if spent fuel pool needs to emptied (leakage, inspection, repair)?	During a repair of the lining of the spent fuel pool as a result a leakage, the design does not provide for emptying of the SFP. In case of leakages from the lining, the design provides for a makeup of SFP from systems FAK, JNA, JNG,JMN. Leakages must be repaired using special devices.
G-19	What do the acronyms LPH, HPH, LRW and SRW mean?	The list of the abbreviations has been extended. See Attachment.
G-20	Last sentences of the "boron injection system" descrition: *"In addition, a part of pipelines and equipment of the system performs the function of a barrier preventing radioactivity emission outside the containment;"* What does it mean?	It includes a group of isolation valves of the said system (JDH), which is located at the point where the pipelines of this system cross the containment. In the event of a loss-of-coolant accident, the valves and the pipelines upstream of the discharge of pump JDH 10(20, 30, 40)AP001 located outside of the containment act as a barrier that prevents release of radioactive substances into the environment; they have classified designation 23L (23Л) as per OPB-88/97. The valves and pipelines from pump JDH10(20, 30, 40)AP001 up to tank JNK10(40)BB002 located outside of the containment have classified designation 23.
G-21	Borated water storage system (JNK) - Fuction of the system as operatioal system or safety system is not quite clear. Where is the borated solution stored for make-up water supply in the normal operational mode?	System JNK stores low-concentration borated water (16 gH3B03/kgH20) in tank JNK10(40)BB001 with a total volume of not less than 2000 m^3 and high-concentration borated water (40 gH3B03/kgH20) in 2 tanks JNK10(40)BB002 with a volume of not less than 150 m^3 each, which is required for operation of the NPP in all operating modes. The system is designed to store low-concentration borated water for the following needs: - core emergency cooling in the event of a loss-of-coolant accident; - injection under the contamination during a loss-of-coolant accident or in the event of the steam line rupture within the containment; - make-up water supply to the coolant system under power operation and reactor cooldown at the power unit shutdown; - borated water supply for initial filling of the core catcher heat exchanger in the event of a DBA; - borated water supply to fill the fuel pool, the reactor cavity, and the internals inspection cavity in the refuelling mode. The system is designed to store high-concentration borated water for the following needs: - boron concentration adjustment in the primary circuit under normal operation and anticipated operational occurrences; - injection of borated water into the pressurizer in case of leakages from the primary circuit to the secondary one; - borated water injection into the reactor in case of anticipated operational occurrences accompanied by the anticipated transients without scram (ATWS). In compliance with OPB-88/97 the borated water storage system is by design a safety system important for the NPP safety; as for its functions it is a protective safety system. The components of the borated water storage system that carry out the safety functions refer to safety class 2 as per OPB-88/97; classified designation: 23; group: B as per PNAE G-7-008-89 (with rev. 1); seismic category I as per NP-031-01 All the rest components of the borated water storage system belong to safety class 3 as per OPB-88/97; classified designation: 3N; group: C as per NPAE G-7-008-89 (with rev. 1); seismic category II as per NP-031-01; designation II in the flo System JNK fulfils the functions both of a safety system and of a normal operation system.

G-22	For a better understanding, a figure presenting how the JNB 90 serves as a make-up for the spent fuel pool is needed.	To prevent damage to the spent fuel assemblies due to a decrease of level in the spent fuel (SF) pool in the event of a blackout, the SF pool is made up via the DN 80 mm pipeline by routinely operated pump JNB50AP001 featuring the following characteristics: - nominal pump capacity: 60 m^3/h; - nominal pump head: 132 m; - maximum pump power: 39 kW. See SAR, Chapter 12, cl.12.1.14.2.1, cl.12.1.14.2.2 and Figure 12.1.4.1; also, see the manufacturer's documentation: BLR1.E.344.1.0UJE00.JNB50.021.ZG.0001 Technical passport for electrical pump 1ЦНСг60-132-1-т-К УХЛ4. To ensure an alternative way to make up the SF pool and the emergency heat removal tanks, the existing scheme provides for two additional nozzles with manually operated valves (two per each nozzle) and union nuts of a special type. One nozzle is tied into the suction line of pump JNB50AP001, upstream of valve JNB50AA102; the other nozzle is tied into the discharge pipeline downstream of valve JNB50AA104. In the event of a blackout, these valves allow restoring and maintaining the water level in the SF pool through system JNB50 using alternative water supply sources (mobile diesel pumping unit, fire engines, fire water header, monoblock pump, etc.). The process flow diagram of an alternative way to m
G-23	More information is needed about the emergency heat removal tank make-up line JNB50 subsystem (only one pump and pipeline?) It seems that there will be two mobile diesels and only one pump JNB50AP001?	Per each Unit there will be one pump JNB50 powered from the BDBA fixed power supply system: section BNS90, from which it is possible to supply power to BNS70 and BNS80. It is also possible to power the abovementioned sections from mobile diesel generator XKA70. The design provides for one mobile diesel generator per two units. At the same time, based on the results of the stress-tests it was recommended to have two mobile diesel generators per two Units of the NPP (one mobile DG per each Unit). This recommendation is accepted for implementation. See also the answer to G-22.
G-24	According to page 39 the mobile diesel generator plant operating performance is provided with ambient temperatures -50 °C to +41 °C. According to the table 5.2.1.1 values used in the Belarussian NPP design are -61 °C and +52 °C. It is said that they are placed openly on the NPP site. What is the justification for used diesel generator plant ambient temperatures?	In compliance with NP-064-05 "Accounting external, natural and man-induced impacts on nuclear facilities" the design of the Belarusian NPP considers all the factors characteristic of the site on which it is located that has the frequency of occurrence of at least 10-4 1/year. Based on the said approach Table 5.2.1.1 (the conditions of the Belarusian site) shows the values of extreme temperatures within the range of -50/+37.4 °C. The operating temperature range of the mobile DG (from -50 °C to +41 °C) is selected to cover the range of the site extreme temperatures. The mobile DG is not the determining system to ensure transition and maintaining of the reactor plant in the safe state. The values of the extreme temperatures adopted for the design of the Belarusian NPP and shown in Table 5.2.1.1 cover the abovementioned range of temperatures.
G-25	Since the terminology used in the report differ from IAEA, it would be useful to explain some of the terms, in particular: beyond design basis accidents and severe accidents, safety systems and systems used for management of design extension conditions, inherent and passive safety features.	Main terms: Accident: It is disturbance of the nuclear power plant operation followed by a release of radioactive substances and (or) ionizing irradiation beyond the boundaries established by the design of NPP for a normal operation in the amounts exceeding the preset safety operation limits. The accident is characterized by an initial event, sequence scenarios, and consequences. Emergency protection: It is a safety function of a rapid transfer of the reactor into a subcritical state, and maintaining it subcritical; it is also a set of safety systems that perform the emergency protection function. Active system (element): It is a system (element), functioning of which depends on the normal operation of another system (element), e.g. safety control system , source of power, etc. Intrinsic safety: It is a property that ensures safety basing on natural negative feedback relations and processes. Beyond Design Basis Accident: It is an accident caused by initial events which are not taken into account for the design-basis accidents or which are accompanied by additional failures of safety systems, in comparison with design-basis accidents, in excess of the individual failure, with realization of erroneous decisions of the personnel. Protective safety systems (elements): These are systems (elements) designed to prevent or limit damage of the nuclear fuel, fuel element claddings, equipment and pipelines containing radioactive substances. Conservative approach: This is a design approach when during analyses of accidents such numerical parameters or characteristics are assumed that admittedly lead to most unfavourable results. Localization (or confining) safety systems (elements): These are systems (elements) designed to prevent or limit a spread of radioactive substances and (or) ionizing irradiation resulting from an accident beyond the boundaries established by Maximum design limit of the fuel elements damage: These are admissible values of parameters and characteristics of fuel elements in design basis accidents, exceeding of which can result in destruction of the fuel elements. Disturbance of normal operation of a nuclear power plant: It is a disturbance in operation of a nuclear power plant resulting in its deviation from the established operating limits and conditions. In this case, other limits and conditions established Supporting safety systems (elements): These are systems (elements) designed to supply power and operating medium to the safety systems, and to provide conditions for functioning of the safety systems. Passive safety system (element): This is a system (element) whose functioning is associated only with the event which brought it into operation and does not depend on operation of another active system, e.g. control system, power source, a Design-basis accident: This is an accident for which the initial events and final states are defined by the design and safety systems are provided which, with allowance for the principle of single failure of the safety system or one personnel err Reactor shutdown system This is a system to bring the reactor into the subcritical state and keep it subcritical by means of reactivity control devices. Reactor control and protection system: This is a complex of hard , software and infware designed to safely control the chain fission reaction. Control and protection system is a system important for safety, it combines functions of the normal operation and safety; the system consists of elements of normal operation control system, protection, control and supporting safety systems. Safety systems (elements): These are systems (elements) designed to perform the safety functions. According to the type of performed functions the safety systems and elements are divided into protective, localization, and supporting ones. Systems (elements) important for safety. : These are safety systems (elements) and normal operation systems (elements) whose failures disturb the normal operation of the nuclear power plant or prevent elimination of deviations from norma Severe beyond design basis accident: It is a beyond design basis accident with the fuel damage above the maximum design limit which can lead to the maximum permissible emergency release of the radioactive substances into the environme Severe damage of the reactor core: It is a beyond design basis accident with the fuel damage above the maximum design limit under which the maximum permissible emergency release of the radioactive substances into the environment can Control safety systems (elements): These are systems (elements) designed to actuate the safety systems, control and monitor them during performance of the preset functions. Nuclear accident: It is an accident associated with a damage of fuel elements above the established limits of the safe operation and/or radiation exposure of the personnel above the permissible limits, caused by: -violation of monitoring and control of the chain fission reaction in the reactor core; -occurrence of criticality during reloading, transportation and storage of fuel elements; -deterioration of heat transfer from the fuel elements; -other causes leading to damage of fuel elements. Nuclear safety: It is the state of protection of the population and environment from a deleterious impact of ionizing irradiation of a nuclear installation and (or) storage facility which is achieved through a proper operation conditions, duly organiz The complete list of terms and definitions is provided in NP-001-97 "General Provisions to Ensure Safety of Nuclear Stations (OPB-88/97)".
G-26	Have been any safety related additional studies developed and taken into account after the Fukushima Accident?	The target reassessment of safety (stress tests) performs as additional research which can identify deficiencies or margins of NPP design safety taking into consideration the Fukushima events. Based on the results of the target reassessment of safety of the Belarusian NPP , the margins of safety regarding each of the considered extreme impacts were identified, thus demonstrating a safe protection of the Belarusian NPP against factors typical for the Fukushima accident. In line with the ENSREG specification when implementing stress-tests for the Belarusian NPP it was proposed to introduce potential safety enhancement measures for the considered impacts, which may be implemented, in case of necessity, by means of the Safety Enhancement Program for the Belarusian NPP. Besides, above implementation of the additional safety research (stress tests), the Belarusian party has arranged in the period of 16-20 January 2017 the SEED IAEA mission. To evaluate the Belarusian NPP safety versus special external impacts. During the SEED mission, the team of inspectors has evaluated the information provided by the Belarusian party. Inspectors, following a comparison of the design parameters and the site characteristics, came to the conclusion, that relevant
G-27	What is the approach in Belarus legislation and regulatory practice regarding use of IAEA Safety Standards? Does legislation specifically requires independent verification of safety assessment by the operating organization? Can you please provide more information regarding the **national** requirements applying to accident management (program) for the Belarusian NPP?	1. The IAEA recommendations are considered when elaborating the normative legal acts, including technical normative legal acts of the Republic of Belarus in the sphere of assurance of the nuclear and radiation safety. In accordance with clause "a" of item 24 of the "Regulations on Elaboration of Normative Legal Acts" № 359, approved by the President of the Republic of Belarus on 11.08.2003 «Measures to Enhance the Law-Making Activities" (hereinafter-Measures), prior to start elaborating of the draft of normative legal act, a relevant legislation of the same sphere is studied . This reference legislation includes international treaties of the Republic of Belarus, international laws of foreign countries, IAEA recommendations, proposals of the state bodies and other organizations, R&D, publications. 2. The procedure on the expert evaluation of documents that substantiate the assurance of nuclear and radiation safety when carrying out activities in the sphere of application of nuclear energy and sources of ionizing irradiation is defined by the Decree № 1781 of the Council of Ministers of the Republic of Belarus of 07.12.2010 " On approval of the Regulations on the procedure for the examination of documents substantiating the provision of nuclear and radiation safety in the implementation of activities in the field of the use of atomic energy and sources of ionizing radiation." The Regulator is the organizin

G-28	What is the long term concept of spent fuel storage or fuel reprocessing?	The long-term concept of storage and handling of the spent nuclear fuel is set forth in the Agreement between the Governments of the Republic of Belarus and the Russian Federation, specifically: "The nuclear fuel burned in the reactors of the NPP power units that has been acquired from the Russian Party shall be returned to the Russian Federation for reprocessing on the conditions agreed upon by the Parties in a separate agreement." The Strategy and Plan to manage the spent nuclear fuel, which detail all stages of the spent nuclear fuel life-cycle following its removal from SFPs , is currently under development. In frames of development of the Strategy various variants and approachable options on the spent nuclear fuel managements, including the post-reprocessing products handling, are being analysed on the nation level. Also, a feasibility of construction on the Belarusian NPP site of an intermediate-stage spent nuclear fuel repository is under consideration.
G-29	It is difficult to assess adequacy of mitigation countermeasures without information on time progress of bounding severe accidents. Please provide selected information in an appropriate form (tables, plots) about timing and severity of key phenomena during evolution of severe accidents.	This section recapitulates measures to control accidents. A detailed description of evolution of accidents is provided in sections 6 and 7. At the same time, sections 6 and 7 describe measures that allow to control specific scenarios of accidents.
G-30	What are additional parameters of the containment: containment volume, leak rate, ultimate pressure, secondary containment by-pass?	Volume of the outer containment is 92018m3, the design leakage rate in case of LOCA is no more than 0.2% of the volume per 24 hours. There is no by-pass.
G-31	Defence in depth is described in rather general way and IAEA terminology is not used: Could the categories of the plant systems be described more specifically, in particular systems available for level 3 and level 4 of defence in depth?	1. Technical measures for the barriers protection. 1.1 General provisions To perform safety and protection functions, each level of protection in the VVER-1200 NPP design is equipped with technical devices, application of which is duly substantiated. These technical devices include special devices to shut down the reactor, keep it subcritical, remove the residual heat, and confine radioactive substances releases. Functioning of the Unit in the normal operation mode is ensured by automatic control systems, protection and interlocks actuated by NO I&C. The first level of defense features predominantly an automated control, i.e. control by means of automation equipment with involvement of the personnel. The instrumentation and control devices provide the operations, administrative, and technical personnel with information sufficient for their operations-related activities. The design provides for a system of control, monitoring, and diagnostics of the reactor plant that perform diagnosis in the process of operation of the reactor plant main process equipment. 1.3 protection technical resources of level 2. Level 2 technical resources are represented by the normal operation systems. Table 1.3.1 (Appendix to G-31) provides main protection functions and relevant technical resources of the Level 2 , the Table contains also modes of system contr Table 1.3.1- protection technical resources of Level 2 1.4 protection technical resources of level 3. Table 1.4.1 (Appendix to G-31) provides main protection functions and relevant technical resources of Level 3, the Table contains also modes of safety system control under an accident situation. 1.5 Level 4 technical resources Level 4 technical resources are represented by the normal operation systems, safety systems and additional technical devices for BDBA control. Table 1.5.1 (Appendix to G-31) provides main protection functions of Level 4 and relevant ad To protect these systems from the common cause failures and to enhance the NPP safety indices as a whole, the design provides for functional redundancy of the systems for performing the main safety functions, as shown in Table 1.5.2 (1.6 Assurance of the NPP defence levels for the electrical power supply. In line with the safety concept, the design provides for the following electrical power supply systems, which accordingly refer to defence levels 1,2, and 3. - power supply system of facilities of normal operation; - reliable power supply system of facilities of the normal operation (Unit diesel-generator and batteries); - emergency power supply system of special facilities (emergency diesel generators and batteries). 1.7 Assurance of the NPP defence levels for the instrumentation and control systems (I&C) . From the point of view of the safety assurance during failures, the Unit I&C structure is based on the defence-in-depth principle. The multilevel defence with application of various control systems ensures implementation of each main functio Level 1- prevention of anticipated operational occurrences of the normal operation for ensuring a safe operations of the Power Unit and reducing a possibility of occurrence of initial events of an accident. For this purpose are used the syste - controlling the main technological process; - maintaining and reducing impact on the physical barriers within the operation limits;
G-32	From the description it seems that there is a single FAK system to ensure spent fuel cooling under all design basis and beyond design basis conditions; is there any other fixed system to prevent or mitigate severe accident in the case of FAK system failure?	Channels 20 and 30 of system JNA/JNG-1/JMN are used as the backup system to cool down SFP. .
G-33	How power supply for active containment annulus is ensured under station black out conditions?	System KLC11/21/31/41 consists of four equal independent from each other channels. Each channel is fed from the relevant channel of emergency power supply system. Under the NPP black out conditions the power supply to the active elements is not provided. To cut off the annulus premises from the safety system building premises in the BDBA mode with complete loss of power supply onto the air duct connecting the extraction headers of systems KLG and KLC, the design proves for installation of a manually-driven airtight valve KLC31AA001. The valve shall be closed manually by the personnel during 2 hours after the initial event of the accident. The link between the annulus premises serve to equalize pressure between them in case of an accident.
G-34	From the description it seems that for the depressurization of the reactor during severe accidents pressurizer relief valves and emergency gas removal system KTP are available, it means that there is no relevant dedicated system. Is this observation correct?	The design provides for measures to reduce pressure in the primary circuit (POSV of pressurizer and emergency gas removal system), including also under conditions of severe accidents. However, it should be noted that using of pressurizer POSV and emergency gas removal system are the last resort measures. The design provides for other systems - SG PHRS, BRU-A, and SG POSV -that ensure an effective heat removal from the primary circuit.
G-35	Please described quantitatively functioning of the containment passive heat removal system JNB in the case of large break LOCA combined with station blackout. How depressurization of the containment is assumed in such case?	Operation of the containment PHRS system is based on the passive principles. Valves of the system is always opened except for emergency isolation of the leaking heat exchanger-condenser. The emergency heat removal tanks are filled with cooling water. The containment PHRS system enables to maintain pressure in the containment at a level below the design one without participation of an operator within, at least, 24 hours in the entire range of the beyond design-basis accidents connected with mass and energy yielding under the containment. In 24 hours to provide operation of the system mobile equipment and water reserve (for make-up of the emergency heat removal tanks with cooling water) are used. In SAR, chapter 12 (section 12.1.12), Fig. 1 and 2 demonstrate influence of the JMP system operation on the parameters under the containment under severe BDBA with molten core iodine 30 (large leakage DN346 accompanied by failure of the ECCS active part).

		 Рисунок 1 – Давление под ЗО  Рисунок 2 – Температура под ЗО
		Curves 1 given on Figures 1 and 2 demonstrate change of pressure and temperature under the containment with no regard to the containment PHRS. Curves 2 demonstrate change of pressure and temperature under the containment taking into account operation of three of four PHRS channels. For reference only: The stressed containment is designed taking into account the following impacts in the modes of design-basis accidents: - maximum emergency gage pressure of 0,39 MPa;- maximum emergency temperature inside the containment 150°C;- response from emergency ruptures of the pipelines.
G-36	Please provide the key components contributing to the core damage frequency and fuel damage in the spent fuel pool. Is PSA Level 2 available? Is PSA level 2 required for the development of severe accident management program in Belarusian NPP?	According to the Technical Assignment for the Belarusian NPP, presently the PSAs of the first and second levels are under development addressing internal initial events, internal fires and flooding, external impacts of natural human-induced character, seismic impacts. PSAs of the first and second levels address the fuel in the reactor and SFP covering all operation conditions (operation at the nominal and reduced power levels, shut-down mode, and transportation and handling operations during refuelling. At the time PSA of the first level for the Belarusian NPP addressing internal initial events, internal fires and flooding, external impacts of natural human-induced character is ready. The reporting materials were handed over to the Gosatomnadzor for the expert evaluation. PSA-1 of seismic impacts is presently under development, will be completed in the second quarter of 2018; PSA-2 for internal initial events is presently under development, will be completed in the second quarter of 2018; PSA-1 addressing internal fires, flooding cases, external impacts, and seismic impacts is presently under development, will be completed in the fourth quarter of 2018. 2) Results of PSA-2 activities will be taken into consideration when re-editing the Severe Accident Management Manual. 3) The main components of SAR-1 for internal initiating events are given bellow: In the current revision of SAR-1 for internal initiating events the frequency of fuel damage in the reactor under power operation is 3.76 E-7 1/year (at the time of 3.1) The main components of SAR-1 for internal initiating events having an effect on the core damage frequency under power operation: - Very small (compensated) leakage - 2,39E-07 (1/year); 63,4%- Small leaks with DN 10-20mm - 5,74E-08 (1/year); 15,2%- Administrative shutdown due to failure of three or more channels of the safety systems - 1,24E-08 (1/year); 3,3%- R - Loss of heat removal by systems JNA-JNG - 3,67E-08 (1/year); 32,9%- Loss of external power supply - 2,87E-08 (1/year); 25,7%- Loss of normal operation power supply - 2,23E-08 (1/year); 20,0%- Very small (compensated) leakage - 1,13 - Loss of normal operation power supply 1,05E-10 (1/year); 86,1%- Loss of external power supply 1,70E-11 (1/year); 13,9%3.4) The main components of SAR-1 for internal initiating events having an effect on the frequency of fuel damage in - Loss of external power supply 3,22E-08 (1/year); 75,9%- Damage of nuclear fuel during transport operations 5,98E-09 (1/years); 14,1%- Failure of FAK system 2,47E-09 (1/year); 5,8%- Loss of normal operation power supply 1,80E-09 (1/y
G-37	All in Section 8.3 of Belarusian NPP Stress Tests National Report mentioned safety improvement measures as well as all measures that will be identified during peer review mission shall be implemented before the start of operation of Belarussian NPP.	Results of the targeted safety reassessment of the Belarussian NPP have defined sufficiency of the existing design measures to ensure the NPP safety taking into account the Fukushima accident. Insufficiency of safety was not identified and additional measures to enhance the design safety level are not required. Furthermore, for each of the considered extreme impacts the safety margins were defined, thus demonstrating a safe protection of the Belarusian NPP from factors typical for the Fukushima accident. In line with the ENSREG specification , when implementing stress-tests for the Belarusian NPP it was proposed to introduce potential safety enhancement measures for the considered impacts. As insufficiency of safety of the Belarusian NPP was not identified , it is not necessary to implement the proposed measures (following stress-tests results) before the start of commercial operation. At the same time, as set forth in i 8.2 of the National Report ,adhering to the principle that safety is priority, the proposed measures plus measures following the peer-review will be incorporated into the Safety Enhancement Program of the Belarusian NPP. To include these measures in the Program their influence on the NPP safety will be analysed and, depending on the analysis results, priority of their implementation will be determined. The terms of implementation of the measures will be specified by the Safety Enhanc

G-38	According to ENSREG specification for the "stress tests" the approach used in “stress tests” should be essentially deterministic. Only few results from deterministic analysis are demonstrated in Belarusian NPP Stress Tests National Report (hereinafter – Report). In the most cases the details are referred to the “Report on the conduct of a targeted reassessment of safety (stress tests) of the Belarusian NPP” BL-11752” and “Analysis of seismic resistance of the main equipment of the reactor unit of units 1,2 of Belarusian NPP at 8-points MDBE, 491-Pr-1975”.	The documents can be submitted to PRT experts for review within the period from 12.03.2018 through 16.03.2018.
	More detailed deterministic analyses are required. “Report on the conduct of a targeted reassessment of safety (stress tests) of the Belarusian NPP” BL-11752” and “Analysis of seismic resistance of the main equipment of the reactor unit of units 1,2 of Belarusian NPP at 8-points MDBE, 491-Pr-1975”shall be made available for the international nuclear safety community.	
G-39	The Emergency Operating Procedures (EOP’s), Beyond Design Basis Accidents Management Guidelines (BDBAMG’s), Severe Accident Management Guidelines (SAMG’s), on-site and off-site Emergency Preparedness Plans (EPP’s) shall be developed and validated before the start of operation of Belarussian NPP. In accordance with IAEA Requirements (SSR-2/2 (Rev. 1), Requirement 26), both event based approaches and symptom based approaches shall be used for EOP’s, BDBAMG’s and SAMG’s development.	Development of the emergency response documentation for the Belarusian NPP proceeds in line with the schedule for development of the operations documentation, it is reviewed and approved by the Chief Designer of the Reactor Plant, Project’s Scientific Manager, and NPP General Designer. Presently, in line with requirements of the normative documentation of the Republic of Belarus, the following documents were elaborated for NPP: -Procedure on Elimination of Anticipated Operational Occurrences; - Procedure on Elimination of Design Basis Accidents (event-related provisions); - Beyond Design Basis Accident Management Procedure (event-related provisions); - Action Plan for Personnel Protection (in-house emergency plan); - Action Plan for Population Protection (external emergency plan); The Nuclear Power Legislation of the Republic of Belarus does not require that the Emergency Management Procedures for a Nuclear Plant are developed in the symptom-based format Despite this, presently the staff of the Belarusian NPP is developing a package of Emergency Management Procedures in the symptom-based format. Before start of the commercial operation of the Belarusian NPP the Emergency Management Procedures in the symptom-based format will be elaborated.
G-40	Was probabilistic analysis of accidental crash of commercial aircraft on purpose to screen out such event for Belorussian NPP site performed? The general input data for such analysis (airports in vicinity, traffics, aircraft types and mass etc.) should be provided.	This question is beyond the scope of the stress-tests. Within the analysis of the air traffic impact on safety of the Belarusian NPP several evaluations have been made, including those using probabilistic approach. Within the framework of these works special measures have been determined to ensure that the standard probability of a crash of all types of aircraft on the Belarusian NPP site is not exceeded. Among other things the parameters of the prohibited airspace area above the Belarusian NPP site have been determined based on the probability analysis of the aeronautical situation with due consideration of the flying qualities of the aircraft. As of now, all the measures required to ensure safety of the Belarusian NPP with respect to a crash of all types of aircraft have been implemented. The airspace above the Belarusian NPP site is closed for flights (a prohibited airspace area is established); therefore, a probability of a crash of aircraft of any type on the Belarusian NPP site is below 10-6 1/year. The impact of an aircraft crash (including a big passenger airliner) on safety of the Belarusian NPP has been studied and evaluated within the SEED mission of IAEA to the Belarusian NPP, which is stated in the respective report. The inspection of IAEA has made the following conclusion: “Protection of the Belarusian NPP against an aircraft crash has been ensured using the design and administrative measures to control and restrict the air traffic (within the prohibited airsp
	In some sections of the report there is presented information that appropriate systems and components are protected against impact of aircraft crash. However is not clear what type of aircraft is chosen for evaluation of aircraft crash impact for Belorussian NPP as postulated external event and how the impact is evaluated.	This question is beyond the scope of the stress-tests. The design of the Belarusian NPP provides protection against an impact of a crash of a light aircraft (5.7 t.) with a speed of 100 m/s; this is in compliance with the regulatory requirements of the Republic of Belarus (TKP 263-2010 (02300). Moreover, the design of the Belarusian NPP with respect to consideration of the impact of an aircraft crash on the NPP site complies with the requirements and recommendations of the IAEA documents valid at the time of the NPP design: IAEA Safety Guide NS-G-3.1 “External Human Induced Events in Site Evaluation for Nuclear Power Plants,” 2004; IAEA Safety Guide NS-G-1.5 “External Events Excluding Earthquakes in the Design of Nuclear Power Plants” (Section 4 “Aircraft Crash”), 2003 г. The cited international documents do not set specific requirements for the aircraft characteristics. The IAEA recommendations do not presuppose consideration within the design basis of an intentional aircraft crash and stipulate protection against such an event as a task of the physical protection.

G-41	It is necessary to specify what type (weight) of aircrafts are evaluated in the Belarusian NPP design and provide information on results on evaluation. The Belarusian NPP is being constructed in the close vicinity of EU border. Taking into account this fact the evaluation of aircraft crash shall be performed in compliance with the position on safety objectives for new power reactors of European regulatory bodies. This position is set in documents "WENRA Reactor Harmonization Working Group study "Safety of new NPP designs", March 2013, "WENRA Reactor Harmonization Working Group study "Safety Objectives for New Power Reactors", October 2009 and "WENRA statement on safety objectives for new nuclear power plants", November 2010. The evaluation of intentional crash of a commercial airplane (much larger than small or military airplanes) for Belarusian NPP shall be performed. The evaluation shall include the effects of direct and secondary impacts on mechanical resistance of safety structures and systems required to bring and maintain the plant in a safe state after airplane crash; effects of vibrations on safety structures and systems required to bring and maintain the plant in a safe state after airplane crash; effects of combustion and/or explosion c	The abovementioned documents that regulate how an aircraft impact should be taken into account in the NPP design recommend to use a probabilistic approach when initiating events with a probability of less than 10-6 a year may be disregar The proposed measures include redistribution of the air traffic in the area of the Belarusian NPP and in accordance with Decrees of the Belarus Ministry of Defence No. 19 dd. 27.09.2017, and No. 21 dd. 13.12.2017 a prohibited airspace area As for the WENRA recommendations quoted in question G-41, we would like to point out that when these documents were being developed the Belarusian NPP was already under construction. Meanwhile, the WENRA documents in question The impact of an aircraft crash (including a big passenger airliner) on safety of the Belarusian NPP has been studied and evaluated within the SEED mission of IAEA to the Belarusian NPP, which is stated in the respective report. The inspecti
G-42	The presented information concerning licence holder is very limited – just fact about establishment of operating organisation is provided.	1. The organizational structure adopted in March of 2017 is attached. 2. The design number of the personnel of the Belarusian NPP is 2321 persons. As of 01.01.2018, the total number of the personnel of the Belarusian NPP is 1140 (49% of the design number). It is planned that in 2018 the total number of the personnel will be increased up to 1680 persons (72% of the design number) and in 2019 the NPP will be staffed by 100%. 3. It is planned that by the time of commissioning of the first Unit of the Belarusian NPP the number of the operating personnel who have completed their training and are permitted to carry out their duties without supervision will be 1160 persons: - foreign managers and specialists having higher education in the respective sphere and experience in working at NPP - 69 persons (6% of the total number); - managers and specialists of the Republic of Belarus having higher education in the respective sphere and experience in working at thermal power plants and other enterprises of the power industry – 472 persons (41% of the total number); - young specialists of the Republic of Belarus (graduates of higher education institutions) having education in the respective sphere – 255 persons (23% of the total number); - young specialists of the Republic of Belarus (graduates of specialized secondary schools) having education in the respective sphere – 68 persons (6% of the total number); - other specialists and workers of the Republic of Belarus employed as independent contractors – 296 persons (26% of the total number). 4. The number of the personnel required for commissioning of the first Unit of the Belarusian NPP: - operating personnel: 465 persons (40% of the NPP operating and maintenance personnel). As of 01.01.2018, 267 persons have already been employed (57% of the number required for the first Unit). The personnel is being trained under th - repair personnel: 318 persons (33% of the NPP operating and maintenance personnel). As of 01.01.2018, 186 persons have already been employed (49% of the number required for the first Unit). The repair personnel are among the last to 5. The personnel of the Belarusian NPP that must have the permit issued by the Department of Nuclear and Radiation Safety of the Ministry of Emergency Situations of the Republic of Belarus (Gosatomnadzor) for work in the nuclear power f Obtaining permits for performing technological process by the personnel is planned in Quarter 4 of 2018. Licensing of the Belarusian NPP personnel performing other types of activity is planned to finish in Quarter 3 of 2018
	The information on structure, personnel competences, staff number, staffing plans should be provided on purpose to ascertain if the human recourses needed for safe operation, accident management and emergency preparedness is or will be in place before commissioning of the 1st unit of Belorussian NPP.	
G-43	In the table 2.2.1 solid radioactive waste storage facilities with areas 777.5 m2 and 673.5 m2 are mentioned).	The Belarusian NPP project provides SRW storage facilities: one SRWSF per each power unit. The SRW storage facilities are located in the reactor island in buildings 10UKT and 20UKT. The storage facility is a reinforced concrete structure providing biological protection of the personnel and environment. The storage facility is designed for interim storage of the conditioned SRW and solidified LRW. The storage consists of the following main rooms: room for storage of very low- and low-level SRW; room for storage of intermediate-level SRW; room for SLRW ; reinforced concrete compartment for storage of high-level SRW; room of SRW processing plant (only in building 10UKT of power unit No. 1). The conditioned very low-, low- and intermediate-level SRW is stored in steel drums (0,2 m3) arranged in 6 rows in height close to each other. The solidified LRW is stored in square reinforced concrete non-returnable containers (1,5 m3) arranged in 8 rows in height close to each other. The high-level SRW is stored in steel capsules arranged in tubes in 9 rows in height. Capacity of the rooms for storage of very low-, low- and intermediate-level SRW and SLRW is designed for 10 years of NPP operation. Capacity of the high-level SRW storage compartment is designed for the entire NPP service life. The SRW processing plant is designed for SRW sorting, shredding and compaction with the subsequent loading in steel drums and for SRW drums data-sheet production. It is planned to commission the storage facility prior to the power unit start-up. Layout of the buildings and structures is given the Belarusian NPP design documentation in section 2 "Area layout scheme", volume 2 "Drawings", book 1 "Drawings".
	Please provide more detailed information about these facilities: where they are located on the NPP site, when they will be put into operation, in which type packages will be stored solid radioactive waste ant etc.). The layout of the NPP site demonstrating all main facilities should be provided.	
G-44	In the **table 2.2.1** at position 21"Number of main feed water pumps, and type of drive" it is written "Provisionally: 5 FEP. (electric drive)".	It must be written as follows: «~~Provisionally:~~ 5 FEP. ~~(Electric drive)~~»
	Why at this moment the exact amount of equipment is not clear?	The comment is accepted. The amount of equipment is known. It will be modified.
	What number of equipment is confirmed in Safety analysis report approved by Belarusian regulator?	In SAR 5 FEPs are indicated in accordance with the design.

	Please provide the requested information.	
G-45	In the **table 2.2.1 Basic characteristic of the NPP unit with VVER-1200** (Page 18) it is stated: "Reactor spent fuel pool (storage pool), spent FA storage system description located in the reactor compartment, as well as systems that provide fuel transportation and installation are given in [31]". Please provide the document "Report on the conduct of a targeted reassessment of safety (stress tests) of the Belarusian NPP" BL-11752" or provide more information on the issue (layouts, capacities etc.)	The spent fuel (SF) pool is located in the sealed area of the reactor compartment within the SG box between the main circulating loops close to the reactor shaft; the SF pool is connected with the reactor shaft via a transport corridor designed for transporting one fuel assembly at a time. Between the corridor and the SF pool there is a sluice gate. The upper elevation of the reactor shaft and the SF pool (el.+26.300) is preconditioned by the reactor design and the height of the protective water level above the core of the spent fuel assembly during its transportation. The transport corridor also connects the SF pool with the refuelling cavity. There is also a sluice gate between the transport corridor and the refuelling cavity. The refuelling cavity is used during transportation of nuclear fuel from (to) SF pool. The refuelling cavity has a multipurpose seat at its bottom and an intermediate stop. A transportation casing with new nuclear fuel or a transportation packaging set for spent fuel assemblies is placed in the seat. The SF pool is designed to store spent fuel in the reactor building during 10 years (the scheduled refuelling and unloading of the whole reactor core at any moment of the NPP operation are considered). The SF pool has 12 compactly sp
G-46	In the table **2.2.1 - Basic characteristic of the NPP unit with VVER-1200 Basic Characteristics of the Units** (Page 19) it is stated: „-design overpressure – 0.4 MPa". In the section **3.2.2. Earthquake Intensity Leading to Loss of Containment Integrity** (Page 64) it is stated: "...overpressure 0.39 MPa is accepted with the safety factor of 1.5". Please explain, why different design overpressures of the internal containment are given in different places of the Report. What design overpressure value is accepted for inner containment?	The design overpressure (the pressure in case of a design basis accident: LOCA) is adopted as equal to 0.39 MPa. For the purpose of the strength analysis of the internal containment the overpressure value is taken with a safety factor of 1.5. The value "0.4" is a typing error (the pressure recalculated per 4 kg/cm2 and back).
G-47	The design basis overpressure and design basis temperature are presented only for internal containment on the table 2.2.1. The parameters should be provided for outer containment as well. The design requirements for tightness of the containments (authorised leak rate etc.) shall be provided too.	The internal containment's function is to localize internal impacts (temperature, pressure). The outer containment serves to protect against external impacts. There are no design requirements for tightness of the outer containment. For the inner containment the design authorised leak rate must not exceed 0.2% of the total volume per day.
G-48	It is stated: "…in case of BDBA, the radiation exposure is limited to acceptable values". What are radiation exposure limited acceptable values in case of BDBA? Please provide Belarus legal norms in which provided radiation exposure limited acceptable values in case of BDBA.	In compliance with the Technical Assignment for the Belarusian NPP the following target criteria in the event of a BDBA are set (including severe accidents with a probability of emergency release exceeding 1E-7 1/year-reactor) with due consideration of the Russian regulatory requirements, European and international recommendations EUR, rev.C/D: - the design radius of the compulsory population evacuation zone when level B (Б) of the predicted dose of radiation exposure during the first 10 days has been reached (NRB-99/2009) must not exceed 800 m from the reactor compartment; - the compulsory population protection measures zone when level B (Б) of the predicted dose of radiation exposure during the first 10 days has been reached (NRB-99/2009) must not exceed 3 km from the Unit. - the target limit of the Cs-137 release into the environment in the event of a severe accident with core melt must be less than 100 Tbq.

G-49	In section 2.3.1.3 the Design Limits are presented. Regarding the severe beyond design-basis accidents with the core melting as a limit "concentration of gas mixture generated in the reactor and in space under the reactor after the drop-over of corium shall *not reach a hazardous explosive value"* is specified. But it is not clear what is the numerical value of such *hazardous explosive value* .	NP-040-02 stipulates that the design must prevent detonation processes. In terms of the Shapiro-Moffette diagram the transition from burning to detonation is represented as a plane curve drawn with reference to three coordinates (concentration of hydrogen, oxygen and water steam). Position of this boundary is determined by a combination of pressure and temperature. Consequently, there is an unlimited number of set of values (concentration of hydrogen, oxygen, steam, pressure and temperature) that can be called the numerical value of gas mixture design limit. That is the reason why this value cannot be given.
	The numerical value of gas mixture design limit should be specified.	
G-50	In the section 2.3.2.1 the Defense- in-Depth Principles are presented. The objectives and functions of defense in-depth are provided for the all 1-5 levels. But the radiological consequences are not mentioned.	In compliance with the Technical Assignment for the Belarusian NPP, the requirements of the Russian and Belarusian regulatory documents, as well as international recommendations EUR, rev C/D, the design establishes the following target criteria: - during normal operation, the doses to which the population is exposed as per a separate radiation factor (releases/emissions) must not exceed 10 μSv per year; - in the event of deviations from normal operation, the doses to which the population is exposed must not exceed 10 μSv per year per an event; - in the event of a design-basis accident with a probability above 1E-4 1/year, the effective dose per year must be below 1 mSv per an event; - in the event of a design basis accident with a probability below 1E-4 1/year, the effective dose per year must be below 5 mSv per an event. For the beyond design basis accidents, including severe accidents, the design establishes the area of the compulsory population protection measures zone with due consideration of the criteria for implementation of the protective measures set forth both by the Russian and Belarusian regulatory documents.
	The radiological consequences in each level of defence in depth shall be discussed.	
G-51	In the section 2.3.2.2. it is stated: "…Reactors of this generation within the nuclear safety framework feature the following advantages compared to the PWR reactors: - retention of safe operation conditions for a loner time; - Longer time of the operator non-interference; - Reduced probability of the core melting accidents; ".	The requested information is contained in Section 1.12, Chapter 1 of SAR.
	Please provide a more detailed explanation of the statement.	
G-52	In the section 2.3.3. it is stated: "…Nuclear and radiation safety during SNF reloading is ensured by the organizational and technical measures specified in the project, namely: ... equipment is designed to withstand natural phenomena (earthquakes) and other impacts during the NPP operation;".	The equipment is designed both for natural phenomena and human-induced , impacts (air shock wave with blast pressure up to 30 kPa, crash of an aircraft with weight 5,7 t and speed 100 m/s).
	Please specify other impacts during the NPP operation.	
G-53	In the section 2.3.3 it is stated that "The on-site storage facilities for spent nuclear fuel (wet or dry) are not available". This means, that the spent nuclear fuel assemblies will be transported in containers using railway to the fuel recycling plant. The accidents could occur during spent nuclear fuel transportation also.	The norms and regulations for ensuring nuclear and radiation safety "The requirements for carrying out stress tests (targeted reassessment of safety) of the nuclear power plant" define as a subject for consideration an analysis of all operating states of the NPP, as well as of simultaneous impact on all the reactors and spent fuel pools at the NPP site. The cited above requirements of the Norms and Regulations correspond to the requirements of the EU "stress-test" specifications: "...The reassessment means assessment of the impact on a nuclear power plant when subjected to extreme external events.." Analysis of accidents that can occur outside of the Belarusian NPP site during transportation of spent fuel to a spent nuclear fuel reprocessing plant is beyond the scope of the stress tests. As provided by the design of the Belarusian NPP, once unloaded from the reactor, the spent fuel assemblies are temporarily stored in the on-site storage of spent fuel (in the SF pool) until the radiation characteristics and decay heat decrease, which will allow safe transportation of the spent fuel assemblies from the reactor compartment. All the technical solutions provided by the design of the Belarusian NPP (including those related to spent fuel handling) have been consi How to manage the spent fuel after its storage in the SF pool shall be decided by the Strategy on spent fuel management, which at present is under development. The abovementioned Strategy on spent fuel management considers two variant – transporting the spent fuel assemblies from the reactor compartment to the spent nuclear fuel reprocessing plant in the Russian Federation; – organizing on the territory of the Republic of Belarus of a dry storage facility for spent fuel with the aim to store spent fuel until it is transported to the Russian Federation. At the same time the basic principle of the National policy on spent fuel management is set forth in the Agreement between the governments of the Republic of Belarus and the Russian Federation: "The nuclear fuel burned in the reactors of th Implementation of any of the abovementioned variants can imply transportation of spent nuclear fuel in containers using railway. For transportation of the spent fuel assemblies to the spent nuclear fuel reprocessing plant in the Russian Federation, in compliance with the Russian legislation, the authorized organization develops "The General Plan for Foreign Trade Opera – a safety analysis report; – the materials substantiating reduction of the risk of radiation exposure and enhancement of environmental safety as a result of implementation of the General Plan; – the materials for the environmental impact assessment; – the required environmental permits and licenses; – transportation flow charts, etc. Thus, for transportation of spent fuel to the spent nuclear fuel reprocessing plant in the Russian Federation, an analysis of possible accidents (including accidents during transportation of spent fuel in containers using railway) is made during c If a dry storage facility for spent fuel is organized on the territory of the Republic of Belarus, then the required design and survey works will be done, as well as the safety analysis reports and environmental impact assessment will be made in f – the national legislation requires to implement all the procedures stipulated by the Convention. The Republic of Belarus is a party to The Convention on Environmental Impact Assessment in a Transboundary Context and The Convention on A – The Law of the Republic of Belarus on the State Environmental Expert Review and Environmental Impact Assessment requires that the stationary facilities and/or structures designed for storage of nuclear materials, spent nuclear materials – public hearings on the environmental impact assessment reports are held in accordance with the Procedure for organizing and conducting public hearings devoted to the projects significant from the point of view of the environmental impact, – environmental impact assessment (including assessment of possible transboundary impact) is organized and financed by the Customer and is conducted in accordance with the Procedure for carrying out environmental impact assessment (Also, it is worth mentioning that regardless of which variant of the spent nuclear fuel management will be selected, transportation of spent nuclear fuel within the territory of the Republic of Belarus is regulated by the national legislation: – Law N 32-3 dd. 06.06.2001 of the Republic of Belarus "On Transportation of Dangerous Goods"; – Decree of the Ministry of Emergency Situations of Belarus N 73 dd. 28.12.2012 "On the rules for ensuring safety of transportation of dangerous goods by railway within the territory of the Republic of Belarus"; – Decree of the Ministry of Emergency Situations of Belarus N 61 dd. 08.12.2010 "On the rules for ensuring safety of transportation of dangerous goods by road within the territory of the Republic of Belarus." To summarize the above we can conclude that an analysis of possible accidents and other safety analysis reports related to transportation of spent nuclear fuel outside of the NPP site (or a temporary storage facility for spent fuel) shall be mad
	Accidents during the transportation of spent fuel assemblies to the fuel recycling plant shall be discussed.	
G-54	In the section **2.3.3.** it is stated: "When the period of interim storage is over very low-level, low-level and intermediate-level radwaste is transported to the disposal site for long-term storage and/or disposal."	In compliance with the Strategy for managing radioactive waste from the Belarusian nuclear power plant (enacted by Decree of the Council of Ministers of the Republic of Belarus No. 460 dd. 02.06.2015) preparatory works will be carried out and by 2028 there will be constructed the first stage of the shallow ground burial structure for disposal of the very low-active, low-active and medium-active radioactive waste generated at the NPP during ten years of its operation. At present the conceptual design of the burial structure for disposal of radioactive waste is being developed.
	Design characteristics and place of the disposal site for long-term storage and/or disposal shall be described.	

G-55	No explanations of abbreviations are provided in Figure 2.3.3.1 "Reactor building with elevations of the NPP unit equipment installation." Please provide explanations of abbreviations.	The KKS codes of the systems shown in Figure 2.3.3.1 are detailed in Table 3.1.2.1.
G-56	In the table 2.3.3.1 System of hydrogen removal from the containment (1st subsystem) has 1 channel with 100 % efficiency. Please provide an explanation why this safety system does not have a redundancy. Please provide more detailed information on 1st subsystem and 2st subsystem of System of hydrogen removal from the containment.	There is a misprint in the National Report. Instead of "the system of hydrogen removal from the containment" it should be written "the system for monitoring of hydrogen concentration in the containment." Item 25 of Table 2.3.3.1 shall be read as follows: "The system for monitoring of hydrogen concentration in the containment has 2 channels (2x100%)." Information on the system for ensuring hydrogen explosion safety is given in cl. 7.3.7 of the National Report.
G-56	No explanations on pointed by numbers equipment are provided in Figure 2.3.3.4 „Principle diagram of safety systems, equipment and facilities for BDBA control." Please provide list of equipment drawn on the figure 2.3.3.4.	The requested information is contained in Report BL-11752 /31/. The document can be submitted to PRT for review within the period from 12.03.2018 to 16.03.2018.
G-57	In the section 2.3.3 the special-purpose equipment and facilities of Belarusian NPP are described. Regarding the containment it is written: "Outer containment is made of reinforced concrete and is designed to protect the reactor building from *external effects"*. But these external effects are not specified. The external effects, which are taken into account in the design of containment should be specified.	The loads adopted in the design are described in Section 3.10.1, Chapter 3 of SAR. The outer containment of the reactor building is designed to withstand the following impacts: extreme natural hazards (snow, wind, temperature, tornado) and anthropogenic hazards (explosion, aircraft crash, vehicular impact).
G-58	In the section 2.3.3, the system of passive residual heat removal from the reactor via steam generators (JNB) and system of passive residual heat removal from containment (JMP) are described. It is mentioned, that "system design ensures its fully off-line operation without the operator intervention for at least 24 hours in accidents resulting in complete blackout". 24 hours is to short time for the cooldown of reactor core and depressurization of cooling circuit. The specialized guidelines for the operator actions after these 24 hours should be developed. The documents, describing operators' actions (accident management) and obtaining of necessary equipment following requirement of the "stress test" specification, mentioned in the footnote 3 at page 11, should be specified	The time of operation is conditioned by the volume of water stored in the emergency heat removal tanks (EHRT); operation with 3 SG PHRS 4 EHRTs: 72 hours (Assessment after Fukushima). For 3 EHRT 0-24 hours (routine case). If within 24 hours the emergency power supply system or the normal operation power supply system have not been restored, then it will be necessary to implement measures to make up the EHRTs. The possible sources for making up of the EHRTs are described in the National Report.
G-59	In the section 2.3.3. it is stated: „Inner containment is made of prestressed concrete with a steel sealing cladding, the containment is designed for the design basis accidents (DBA) parameters in combination with safe shutdown earthquake (SSE) and is able to limit the release of radioactive substances generated at the same time". Will the containment perform the safety functions at BDBA parameters? The containment shall fulfil Requirement 54 of IAEA Safety Standards Series No. SSR-2/1 (Rev. 1) "Safety of Nuclear Power Plants: Design. Specific Safety Requirements"	The containment completely satisfies requirement 54 SRR-2/1. The combination of DBA+DBE is shown as being the most dangerous. In the event of a BDBA pressure in the containment does not exceed the design value.
	No information on "leakage localization system of the containment KLC11/21/31/41 and safety building ventilation system valves and air ducts KLG01AA101, KLG01AA102 ,KLG02AA101 , KLG02AA102" is given, just markings.	KLC11/21/31/41 Leakage localization system of the containment KLC11/21/31/41 is designed to create and maintain negative pressure in the annulus of the reactor building and safety system building in the event of accidents considered in the design and to purify the exhaust air before discharging it into the atmosphere through a ventilation stack. The system operates in emergency conditions related to an increase of pressure in the containment. System KLC11/21/31/41 has four identical independent channels with a common ventilation network. The capacity of each channel is 3600 m^3/h. The unit of each channel is powered from the corresponding channel of the emergency power supply system and includes the following components: - check valves; - combined filtration plant for purification from radioactive aerosols and iodine; - motorized sealed shut-off valves; - motorized fire dampers; - a fan. If pressure in the containment rises above 0.129 MPa, then sealed shut-off valves KLG01AA101, KLG01AA102, KLG02AA101, KLG02AA102 installed on the boundary between the auxiliary building and the safety system building are closed, the control valves on the air supply ducts of the annulus are closed, and two trains of filters come into operation automatically. Air is removed from the annulus and the safety system building through the trains of filters and via the exha In normal operation conditions and in the event of the anticipated operational occurrences the leakage localization system of the containment is not in operation. In the event of a BDBA and when carrying out accident response activities, operation of the system is the same as during a DBA, provided that it remains operational and it remains possible to connect it to external systems. BDBA with a blackout Sealed manually-operated valves (KLC11AA001, KLC11AA002, KLC21AA001, KLC31AA001, KLC31AA002, KLC41AA001) are installed in the safety system building on the boundary of the safety channels and the ann KLG01AA101, KLG01AA102, KLG02AA101, KLG02AA102 General supply and exhaust ventilation KLG01, KLG02, KLG13, KLG23, KLG33, KLG43 is designed to supply and remove the required amount of air to/from the rooms of the safety system building and to maintain negative pressure in them. Ventilation systems KLG01, KLG02, KLG13, KLG23, KLG33, KLG43 are a network of supply and exhaust air ducts with the installed valves connected on the boundary with the auxiliary building to the air ducts of systems KLE10 и KLE20 (KL In the event of a DBA with an increase of pressure in the containment above 0.129 MPa shut-off valves on the supply and exhaust air ducts KLG01AA101, KLG01AA102, KLG02AA101, KLG02AA102 on the boundary with the auxiliary buildin The pressure relief valves and the control valves on the supply air ducts are used to maintain negative pressure of at least 50 Pa in the rooms of safety system building and to direct air flows from least contaminated areas to the most contamin Fire dampers are installed at the intersections with fire barriers in order to prevent the spread of fire. At the air duct inlet into the channel from the common channel corridor two fire dampers are installed in series; these dampers are powered f

G-60	The information on these systems (purpose, principles of operation etc.) should be provided.	
G-61	In the section 2.3.3 it is stated that during SNF reloading continuous monitoring of water level and temperature in the spent fuel pool is performed. What are operational limits of equipment for water level and temperature monitoring in the spent fuel pool? May monitoring of water level and temperature in the spent fuel pool be performed in conditions of beyond design basis accident (is it of sufficient capacity, appropriate qualification etc.) ? The safety reference level F4.15 of "WENRA Safety Reference Levels for Existing Reactors" should be fulfilled.	Operational limits: For water temperature in the spent fuel pool: 60°C (the upper operational limit is preconditioned by the need to ensure operability of the underwater closed circuit television system of the refuelling machine). Water level in the SF pool, m: - for fuel storage (17.200-17.400); - for refuelling (24.800-25.000). The lower operational limit is preconditioned by the need to ensure during refuelling biological protection of the personnel working at the maintenance level of the reactor building. The upper operational limit is selected to prevent flooding of the connectors of the electrical wiring unit. The monitoring means that preserve their operability in the event of a BDBA and used to monitor the temperature and level in the SF pool are described in Section 6.3.9 /31/
G-62	„Units No. 1 and No. 2 are constructed in accordance with the Belarusian NPP project documentation establishing the same basic technical requirements to all the systems and equipment of both units. All the differences of units No. 1 and No. 2, their systems and equipment, implemented based on the above design requirements, will be defined on further stages of the Belarusian NPP project." Please elaborate what particular differences of units No. 1 and No. 2, their systems and equipment, implemented based on the above design requirements, exist and on what stages of the Belarusian NPP project will they be defined? In case of any, the stress tests results shall be reassessed taking into account the differences.	There are no differences between the power units in what concerns the safety systems; therefore, reassessment of stress-test results is not required.
T1-1	The report does not consider seismic resistance of the outer containment and effect of its possible destruction during an earthquake on the inner containment.	Outer shell of the containment is designed according to the 1-st seismic resistance category. Limiting value of PGA for RC structures including outer shell of the containment is 0.62g. Limiting value of PGA for inner shell of the containment is 0.51g. Thus inner shell of the containment will fail first then comes structural failure of outer shell of the containment. There is no impact of outer shell of the containment on inner shell of the containment under seismic loads.
T1-2	The report does not present information on seismic resistance margin of equipment of power supply support systems, systems for monitoring and control of additional technical means, whose operation is needed in case of beyond design-basis and severe accidents.	Electrical equipment is designed for peak horizontal acceleration 0.12g (DBE level adopted in the design basis). DBE level is set equal to 0.1g for the site. Values of peak horizontal accelerations (PGA) obtained as a result of field research during seismic microzoning were less than 0.1g (0.069g). Consequently, electrical equipment margin (in terms of seismic resistance) relative to the site DBE is 20%, relative to the site seismic conditions - over 70 %.
T1-3	According to the text, the design basis standard map (TKP 45--3.02-108-2008 at the scale of 1 : 10.000.000) was used to determine the ground motion of the design basis earthquake. Is it correct to understand that no site-specific seismic hazard assessment in accordance with IAEA SSG-9 and/or WENRA 2014 and WENRA 2016 has been performed, e.g., using a PSHA methodology? TKP 45-3.02-108-2008 a general requirement for all types of civil engineering structures or is it a special rule for the design of nuclear power plants?	TKP 45-3.02-108-2008 "High-rise buildings. design standards" are intended for seismic assessment of the regions where nuclear power plants, high-rise buildings, hydroelectric power stations and other critical facilities are located - they are part of Set of Maps of General Seismic Zoning developed in Russian Federation in 1997 with participation of Belarusian specialists. DSZ related (detailed seismic zoning) works and seismic risk zoning using a set of methods were used for seismic assessment to confirm GSZ (general seismic zoning). Types, scope, techniques and results are provided in section 2.4 SAR

T1-4	The average value of frequency of nuclear fuel damage in the reactor obtained from PSA-1 for internal initiating events is at power operation: $7.7x10^{-7}$ per year. Q 1) What are the main contributions in respect to the calculated internal initiating events? Q2) What are the additional contributions of external events as earthquake, flooding and extreme weather conditions in respect to power operation (Please report the specific contributions).	Information on the main components having an effect on the frequency of nuclear fuel damage in the reactor and considered for SAR-1 for internal initiating events under power operation is given in response to G-36. As per requirements of NP-064-05 external impact parameters of natural character are determined and specified in Annex 1 of this document. Design parameters are specified in section 2.8 SAR
T1-5	The average value of total frequency of nuclear fuel damage in the spent fuel pool is at power operation for internal initiating events very low. Q) How is the situation in respect to the external events as earthquake, flooding and extreme weather. (Please report that specific contributions.)	All basic initial data for the estimated interior initiating events are specified in PSA-1, values of nuclear fuel damage frequencies (both in the reactor core and in spent fuel cooling pool) are specified for all NPP operating states in chapter 11, PSA-1, substantiation and selection of initiating events are specified in chapter 6, PSA-1, all numerical values of initiating events (frequencies of initiating events, numerical parameters of basic events, personnel mistakes, general cause failures) are specified in chapter 9. Weather extreme conditions are analysed in chapter 15. We have just received seismic activity curve, it is being analysed, due consideration will be done for potential of an earthquake in chapter 15. Flooding is not considered because ground elevation of NPP is by 70 meters higher than the river level. All external factors are considered in a unified integral model of PSA-1.
T1-6	In Seismic Instruments, 2014, Vol. 50 , No. 4 "General seismic zoning of the territory of Russian Federation: GSZ-2012" new data in respect to the general seismic hazard have been published. Table 2 shows major differences between the used GSZ-1997 and the present GSZ-2012 especially in the areas with intensity 7. Q) Have these new findings been taken into account to define the SSE and DBE levels.?	Maps GSZ-2012 were compiled for the territory of Russian Federation. These changes are not related to RF territory. Maps of GSZ - 97 are valid on the territory of Belarus.
T1-7	See Ch. 3.1.3 p.58 & Ch. 3.1 p.41: Is there a contradiction between the statement that the design level has a minimum margin of 10% and the provided value of 0.069g of the seismic risk zoning?	In cl.3.1.3 - misprint, design seismic levels (0.12g) have margin with respect to PES (possible earthquake source) zones (0.069) is not less than 73%. Cl. 3.1 says about seismic margin as an outcome of assessed stress-tests (0.13g) relative to SSE level for the site (0.1g).
T1-8	Which macroseismic intensity scale is used in the map TKP 45--3.02-108-2008 and the map OCP-97-D (Fig. 3.1.1)?	Maps TKP 45--3.02-108-2008 and GSZ-97D use scale MSK-64
T1-9	SSE and DBE levels - 0.12 g and 0.06: do these numbers refer to maximum ground acceleration (PGA) or maximum horizontal ground acceleration (PGAh)?	These figures refer to peak ground acceleration (PGA). At the same time it is equal to horizontal peak acceleration, vertical acceleration is considered equal to 2/3 of horizontal peak acceleration as per NP-031-01 requirements
T1-10	For the SSE level (exceedance probability 10-4 per year) intensity 7 has been chosen, for the DBE (exceedance probability 10-3 per year): how are the intensity value converted to ground motion? What is the uncertainty related to the conversion from macroseismic intensity to ground motion values?	Intensity values specified in MSK-64 scale points correspond to acceleration values: 8 points - 0.2g, 7 points - 0.1g, 6 points - 0.05g, etc. Intensity value is used for general assessment of seismicity level. Acceleration values within one intensity level may vary in the range +/- 40-50% of the basic value. Seismic level specified in PGA acceleration value is used for calculations and as design input data.
T1-11	The margin ("reserve") for the ground motion value PGA=0.12g of 0.01g is regarded to be extremely small. Which uncertainty is related to the PGA value 0.12g? Is it the mean, median, 84% percentile of the hazard curve? How is the extremely small margin justified?	PGA value = 0.12g is accepted as initial data for development of Basic Design. Margin 0.01g is determined relative to the set value in Basic design. Relative to design value PGA applicable for Belarussian NPP (0.1g) the margin is 0.03g, relative to PGA level for the site (0.069g) the margin is 0.061g
T1-12	Please clarify whether there is a seismic monitoring system inside the plant. Does SL-1 trigger an automatic plant shutdown?	In the neighbouring area of the Belarussian NPP there is a local network of seismic stations to provide monitoring of earthquakes (National Report cl.3.1.1 p.47) The Belarussian NPP Project provides for a system instrumentation aseismic protection as part of automated process control system. The system of industrial aseismic protection is information-management system to generate signals of exceeded admissible levels of seismic impact on civil structures of the power unit for timely reactor scram. Industrial aseismic protection system has a four-channel structure and consists of two sets. Each set ensures seismic control of its proper monitoring points. Each protection channel contains equipment of two sets for industrial aseismic protection system including: - two three-piece seismic sensors of the first and second set; - two switching units of the first and second set for multiplication the signals received from seismic sensors to Reactor emergency protection system part for signal initiating and fuel handling equipment. To ensure implementation of information functions and also to provide analysis of seismic activity at any moment, each set has three additional monitoring seismic sensors installed which signals are registered by in-built tools. Once emergency level of seismic impact threshold (6 points specified in MSK-64 scale) is exceeded, industrial aseismic protection system will generate discrete signal of emergency protection and transfer it to initiating part of emergency prote Level SL-1 corresponds to OBE level in the Russian and Belarusian regulatory documents which corresponds to 6 points in MSK 64 scale. According to requirements of General provisions to ensure safety of nuclear power plants and NP-031 - 01, seismic resistant NPP must ensure yield (generation) of electrical and heat energy up to OBE level included, however NPP Custome Due to this fact in design of the Belarussian NPP it was decided to initiate reactor plant transfer by protection to subcritical state to increase safety level in case of 6 point earthquake resulting in tripped main equipment of the power units, termi

T1-13	There seems to be a confusion with the definition of the SSE, DBE, SL1 & SL-2 levels. SL-1 should correspond to the OBE, not the DBE. SL-2 correctly corresponds to the SSE but should be considered as the DBE as a result (probability of occurrence of 1E-4/yr). SL-2 = DBE = SSE Comment to DBE and SSE - Based on the IAEA safety guide NS-G-1.6 "*Seismic design and qualification for nuclear power plants*" (2.3) it is good practice to evaluate two levels of ground motion hazards. Both hazard levels should generate a number of design basis earthquakes grouped into two series, seismic level 1 (SL-1) and seismic level 2 (SL-2). - SL-1 corresponds to a level with a probability of being exceeded of 1 x 10-2 (mean value) per reactor and year and SL-2 corresponds to a level with a probability of being exceeded in the range 1 x 10-3 to 1 x 10-4 (mean values) or 1 x 10-4 to 1 x 10-5 (median value) per reactor per year. - (2.4) SL-1 or "operating base earthquake" is usually not associated with safety requirements but is related to operational requirements only. - SL-2 is often denoted as safe shutdown earthquake (SSE). - (2.7) A SL-2 design basis earthquake should be adopted for the design of safety classified items. The minimum level should correspond to a peak ground	This is a case of translation inaccuracy. Meaning of Design earthquake (ПЗ) corresponds to SL-1 and OBE, meaning of Maximum design earthquake (МРЗ) corresponds to SL-2, DBE and SSE - is used subject to accepted terminology. Regarding second part of the question expert's clarifications are required.
T1-14	What method has been used in 1998 for the definition of the 10000 years return period earthquake intensity?	Map GSZ-97D is normative. It is part of SNiP II-7-81*. Questions related to applied methods during its build-up must be made to the technique developers.
T1-15	Was it an intensity based probabilistic seismic hazard assessment?	Time for development of seismic PSA- 2 quarter of 2018.
T1-16	Real margin can be assessed taking into account the hazard and the fragility as well, and modelling the plant response (as minimum the success path). The acceptability of seismic design basis from the point of view of "margin" can be assessed, for example, applying Regulatory Guide 1.208 (or ASCE/SEI 43-05). In this context it is not evident whether the 0,3 is sufficient or not.	This clause (3.1) describes design basis related to seismic hazard, seismic levels as approved in the project and specified for the NPP site. The Project analysis from the point of seismic margins evaluation for SSC and generally for the power plant is provided further.
T1-17	This is some margin regarding Design Base Earthquake PGA, only.	This clause (3.1) describes design basis related to seismic hazard, seismic levels as approved in the project and specified for the NPP site. The Project analysis from the point of seismic margins evaluation for SSC and generally for the power plant is provided further.
T1-18	Map developed in 1997 and published in 1999. Is it not obsolete?	Map GSZ-98 is actual for the territory of Belarus.
T1-19	Please explain the methods and approaches which were used to derive the seismic impacts from the distant Vrancea zone (including attenuation functions or ground motion prediction equations).	PSAR, Unit 2, Chapter 2, Book 3, item 2.4.2.3 "OBE and DBE(SL-2) from the Vrancea area were evaluated by calculation according to the seismic activity values of the described area, and the patterns of seismic shocks propagation." ; "Evaluation of maximum magnitude for the earthquakes of the Vrancea area, according to various sources, varies within Mmax = 7.4–7.8 or for Mmax = 8.0. These evaluations are obtained both by analysing the seismic and geological data, and by using a formal approach: correlation of seismic activity and thickness of a seismogenic layer, or using statistical appraisal of maximum likelihood of the method of matching moments under the limited exponential distribution of magnitudes." 1) page 2.4.2-53,54 Strong crustal earthquakes of the Vrancea area with M at or above 6.5 were retrieved; a recurrence curve is plotted. Extrapolation of the recurrence curve into the region of low probabilities P shows that as P = 0.0001, Mmax reaches the value of 7.8. 2) page 2.4.2-54 Energy approach. It is assumed that the position of the margin lines on the graph of the time dependence of seismic energy emission in the area of crustal sources specifies the long-term pattern of the supposed time-independent process, and the ordinate difference between them corresponds to the energy value of the ductility-level earthquake (maximum possible earthquake). The authors of the paper evaluated the
T1-20	Impact of Vrancea earthquakes: the Vrancea 1977 M=7.2 earthquake was felt with intensity IV in Minsk and the region close to the site (macroseismic intensity map by Radu, 1979). This is about the same value as the intensity value determined for the design basis earthquake, which should have a occurrence probability of 10-3 per year. How likely is it that the 10-3 earthquake occurred just 40 years ago?	According to the results of deterministic and probability calculations, a strong earthquake in the Vrancea area will not exceed the magnitude of 6 points at the Belarusian NPP site, adopted for OBE description in the project.
T1-21	Please provide a digital datafile with the Earthquake Catalogue of the East European Platform to the reviewers.	The data on the Eastern European plateau used when evaluating the seismic hazard are given in SAR, Section 2.4, and are adopted by the manual "Earthquakes and microseismicity in the problems of modern geodynamics of the East European Platform" edited by N.V.Sharov, Karelian Research Center of the Russian Academy of Sciences, 2007.
T1-22	Please explain the meaning of XV-level geodynamic zones , XIV-level geodynamic zones and XIII-level zones as well as the N-Q-period (latest tectonic movement). Which data constrain the latest tectonic movement?	The hierarchical block territorial division of the region for Belarusian NPP site location is based on the empirical Piotrovsky-Kaye series. The levels (XV, XIV and XIII) and rank (global, regional, territorial, local) of the structures, including geodynamical zones, are defined by the extent (size). 2. The modern relief is largely originated by neotectonic movements within the Neogene-Quaternary period. PR, Pz and Mz folded mountains were quickly eroded, and by the end of Mz became almost plains. The Altai, Sayan, Tien Shan, Transbaikalia became flattened low-hill terrains. During the Neogene period (23.3 million years ago) and the Quaternary period (2.588 million years ago), "rejuvenation" of the described mountains began, i.e. folded-block mountains appeared with a large amplitude of block differentiation per the faults. On the platforms, the neotectonic movements were predominantly inherited (continuing earlier movements) with small amplitude. The results are specified in section 2.4 SAR. The types, methods, approaches of work are given in the report of the Institute of environmental geology of the Russian Academy of Sciences BL-01370 c/o, listed in the references for SAR developme
T1-23	Please provide a map showing the location of potential XIV-level PES zones, other geodynamic zones, and the location of the site of the Belarusian NPP.	The required maps are given in the SAR, Chapter 2, and Appendices thereto.

T1-24	Velocity gradient of quaternary-Neogene movements: how is the deformation velocity defined, and how is it measured? What is the meaning of the number 4.45*10-9 per year? Is it a strain rate?	Velocity gradient of the Neogene-Quaternary (latest) tectonic movements is calculated based on parameters of geodynamical active zones defined as a result of remote sensing and morphostructural analysis of the area. The results are itemized in Section 2.4 of SAR. The value of 4.45*10-9 per year is a maximum velocity gradient or deformation velocity of the Neogene-Quaternary (latest) tectonic movements in the geodynamical zones of the region for Belarusian NPP site location, that is typical for low-active areas.
T1-25	Location of the site at 4 km distance from the block border between XIII and XIV-level geodynamic zone: how are the geodynamic zones defined?	The geodynamical conditions for the NPP location (defining the geodynamical zones, their orders and parameters) are described based on the results of remote sensing and morphostructural analysis of the area (on a scale of 1:500 000) within a radius of 300 km and the neighbouring area (on a scale of 1:50 000) within a radius of 30 km from the Belarusian NPP site. The following data have been considered to define the geodynamical active zones (GDAZ): 1) the lineament structure of the area; 2) development and propagation of exogenous geological processes; 3) structural features of the relief; 4) the regular distribution and properties of the base and peak surfaces; 5) any faults found by the geological-geomorphological and geological-geophysical methods. For the GDAZ order, please refer to comment T1-22. The results are specified in section 2.4 SAR. (to be considered together with the response to comment T1-22) For the defined geodynamical active zones of the 14th order the following parameters have been considered: their coordinates, length, width, displacement trends for the neotectonic stage and the values of relative displacement. The length of a geodynamical active zone was defined as the length of the fault section located between the intersecting zones of the same or higher order. The width of a geodynamical active zone is defined based on the data processing as the width be
T1-26	Pages 43-44: The Oshmayaniy seismogenic zone - Vilnius zone is described as an "active fault intersection of the first level". Please explain the meaning of this statement, and provide information on the strike-slip fault mentioned in the text (fault length, orientation). How is Mmax=4.5 determined? What is the unertainty is associated with Mmax? How is the assumed depth of only 5 km constrained?	Misinterpretation of the Oshmyany zone definition specified in the National Report. Quote (page 46): "The Oshmyany seismic zone is the continuation of the Vilnius zone. This zone is in vicinity of the active fault intersection of the first level. Given the kinematics, the fault zone is defined as a strike-slip or slip type". 1) Inaccurate definition of the fault levels: the faults were not previously graded based on the empirical Piotrovsky-Kaye series. 2) Definition of the fault zone as "strike-slip or slip type" refers to the Oshmyany zone of faults and it is considered as assumptive. Clarifications: Among the faults penetrating the sedimentary cover, the largest fault is the Oshmyany regional fault passing near Ostrovets, the Oshmyany railway station, and south of Smorgon. It extends in the north-west direction, limits the Volozhin graben from the north-east, and has an amplitude of first several dozens of meters. Most likely, it has a "slip type" component. Only the south-eastern part of the Vilnius regional fault that is echelon-like located against the Oshmyany regional fault, extends into the tectonic zoning. The Oshmyany seismogenic zone is distinguished by geological-geophysical data, remote survey data as a lineament zone comprising several faults, and, given the kinematics, is distinguished as strike-slip provided that linear morpho-anomalies correspond to the faults. A fragment of the active north-western fault was defined as a 3) Based on design methods with Mmax monitoring approved for this domain as per zoning map PES GEZ-97D. When defining Mмax of the PES zones, the margin was not taken into account. A specific approach was implemented for theore
T1-27	Page 43-44: 2 seismic zones of interest are mentioned in the direct vicinity of the plant: Oshmyany zone (Mmax 4.5 at 19km), Daugavpils zone (Mmax 4.5 at 67.5 km). What margin has been considered when determining Mmax?	The magnitude of the PES (possible earthquake source) zones is determined: 1) In terms of magnitude of the strongest earthquake for this structure (with available seismic activity); 2) by analogy with similar structures of other ancient platforms or with geostructures of this region (provided that recorded earthquakes are missing); 2) Based on design methods with Mmax monitoring approved for this domain as per zoning map PES GEZ-97D. When defining Mмax of the PES zones, the margin is not taken into account.
T1-28	Real margin can be assessed taking into account the hazard and the fragility as well. The acceptability of seismic design basis from the point of view of "margin" can be assessed, for example, applying Regulatory Guide 1.208 (or ASCE/SEI 43-05). In this context it is not evident whether the 0,3 is sufficient or not.	see Response to comment T1-16
T1-29	This is some margin regarding Design Base Earthquake PGA, only.	see Response to comment T1-17
T1-30	What method has been used in 1998 for the definition of the 10000 years return period earthquake intensity? Was it an intensity based probabilistic seismic hazard assessment? What basis/standard has been applied to link/correlate the intensity with PGA?	In 1998 earthquakes with a frequency period of 1 time per 10 000 years were determined based on Map GSZ-97D. Earthquake intensity dependence on peak ground acceleration is defined based on the curves as provided in MSK-64 scale. Assessment results of earthquake intensity dependence on peak ground acceleration are provided in section 2.4 of SAR
T1-31	Maps of local sources are missing.	If local earthquake sources are assumed, then related data are provided in SAR, section 2.4
T1-32	How the gradient has been measured? How it has been classified? РБ-019-01,	Gradient is a design value, it is not measured. Velocity gradient of quaternary-neogene (latest) tectonic movements is calculated based on parameters of geodynamical active zones as defined as a result of remote sensing and morphostructural analysis of the territory. Classification of gradients is not considered in the normative documentation. Certain gradients for geodynamic zones of the region (max 4,45*10-9 per year) are specific for subactive platform territories. Are specified in section 2.4 SAR
T1-33	Are Mmax values stated in magnitude or intensity? If numbers refer to magnitudes: which type?	Mмax are the magnitude values. Magnitude is a logarithm of the maximum calculated record amplitude (in microns) which the standard short-period torsional seismograph (T_0 = 0.8 with, V = 2800, h = 0.8) would have registered at a distance of 100 km from an earthquake focus (RB-019-01). Magnitude type is determined for registered earthquake magnitudes. No type is determined for design magnitudes.
T1-34	Daugavpils seismic zone: how is Mmax=4.5 determined?	Mmax for each zone is determined with account for at least three factors: - In terms of magnitude of the strongest earthquake for this structure (with available seismicity); - by analogy with similar structures of other ancient platforms or with geostructures of this region (provided that recorded earthquakes are missing); - Based on design methods with Mmax monitoring approved for this domain as per zoning map PES GEZ-97D.
T1-35	Kaliningrad-Lithuanian seismogenic zone: how is the Mmax=4 and H=8km determined?	See T1-34

T1-36	The process of probabilistic evaluation (par. 9) is not fully understood. It is particularly unclear which attenuation functions (ground motion prediction equations) were used, and how the assumed hypocenter depth are justified. It is further not understood how the "DBE- and SSE-induced shocks for average soil conditions of 4.6 and 7.2 points [I MSK64] should be understood at the background that macroseismic intensity is only defined for integer numbers. How are "intensities" of 4.6 and 7.2 converted into numbers relevant for engineering design (ground acceleration)?	Using probabilistic approach considering data of the Gryuntal catalogue. The Gryuntal catalogue contains 21 events with a magnitude ranged from 2.5 to 5.4. Accuracy of magnitude assessment is +/-0.5 of magnitude unit. Source depths are varied in the range from 3 to 21km. Schedule of frequency of occurrence for the NPP location area with account for monitoring period of 230 years is: lgN = -0,36M-2,36, where N – number of events with M>Mi per year, modified to a unit square 1000 km^2. Considering hypotheses of scattered seismicity according to provided ratio directly under the NPP site one may expect OBE and DBE(SL-2) level magnitude of earthquake equal to 1.8 and 4.6 accordingly. Alternative estimates of shock intensity are received with account for N. V. Shebalin ratio with the relevant average world coefficients of intensity attenuation ranged from an earthquake source: α = 1.5; v = 3.5; c = 3.0. Design depth of earthquake source is accepted for shocks intensity calculations: lgh = 0,1M+0,6. Taking into account M_OBE and M_DBE and the most probable depth of the seismic origin, the intensity values of OBE- and DBE-induced shocks for average soils of the site can reach 4 and 6 points of MSK-64 scale respectively. Monitoring of representativity period for different magnitude ranges may modify slope of frequency schedule and impact on the above mentioned assessments of OBE and DBE(SL-2 According to hypotheses of scattered seismicity there is a potential for earthquakes of M_OBE and M_DBE level at any point of the territory under review including the area directly under NPP site. With account for minimum remoteness from the site of potential PES zones capable to generate similar seismic events, the most probabilistic shock intensity assessments in case of OBE and DBE(SL-2) as applicable for average ground cond
T1-37	It is stated that the integrated seismological and geodynamic research for the NPP was compiled at scales of 1:500.000 for the "site location area" and 1:50 000 for the "neighbouring area". Do these areas correspond to the "region" and "near-region" as defined by IAEA SSG-9 (chapter 3)? Are data and maps available for the "site vicinity", for which IAEA requests maps at scales of 1:5.000?	The Report says about integrated seismological and geodynamic studies of the location area (scale 1:500000) and the neighbouring area of the NPP site as per NP-031-01 requirements. Studies in scale 1:5000 were done as per requirements of NP-031-01 in the course of construction and erection works of the site and the neighbouring area.
T1-38	The report states that […] the most probabilistic intensity value of SSE-induced shocks for average soil conditions are 7.2pts. Therefore why is the SSE maintained at 7pts on the MSK scale? 7 is not equal to 7.2. It should be upgraded to 7.2 pts and the corresponding SSE pga value should be provided. It is probable that with a 7.2pts level SSE, the very limited existing margin of the design (0.01g PGA) will be exceeded.	According to hypotheses of scattered seismicity there is a potential for earthquakes of M_OBE and M_DBE level at any point of the territory under review including the area directly under NPP site. Taking into account M_OBE and M_DBE and the most probable depth of the seismic origin, the intensity values of OBE- and DBE-induced shocks for average soils of the site can reach 4 and 6 points of MSK-64 scale respectively. With account for minimum remoteness from the site of potential PES zones capable to generate similar seismic events, shock intensity rates in case of DBE(SL-2) as applicable for average ground conditions may reach 7.2 points. This assessment is conservative enough and as per NP-031-01 non-integral values are subject to rounding off. Acceleration values within one intensity level may vary in the range +/- 40-50% of the basic value. Seismic level specified in PGA acceleration value is used for calculations and as design input data. In the design bases the PGA value is 0.12g and margin is 0.01g, i.e PGA = 0.13g. For local conditions of the NPP site the values of peak horizontal accelerations is 0.069g.
T1-39	Missing reference to "9) Probabilistic evaluations according to the available lists of earthquakes (probabilistic values of seismic hazard were obtained based on the list of historical earthquakes of the region of NPP site location within 1602-2012 prepared by the Center of Geophysical Monitoring of National Academy of Sciences of Belarus taking into account the list made by Gryuntal).	The question is not quite clear. Where and for what the reference is missing? On p.46 9) of the Report there is a reference to catalogue of historical earthquakes of the NPP location area within the period from 1602 to 2012 prepared by the Center of Geophysical Monitoring of National Academy of Sciences of Belarus with account for the catalogue compiled by Gryuntal). The catalogue itself is not provided to avoid overloading of the Report (catalogue is provided in section 2.4 of SAR)
T1-40	"The man-induced changes of conditions, i.e. rising of groundwater level, excavating a pit, and soil bedding, etc." What is the reason for increasing ground water level? Dewatering the excavation pit is usually result in decreasing of the groundwater-level.	During economic development of any territory natural conditions will be subjected to man-induced change and natural-man-induced systems "object-natural medium" will be formed. Groundwater. It is first to response to man-induced interventions as it is the most dynamic part of the system. This is a well known fact. During 2008-2013 natural hydrodynamic and hydrochemical groundwater mode of the site and the neighbouring territory, filtration characteristics of hydrophilic and waterproof strata were studied, change forecasting in the NPP construction and operation were developed. The main sources of the changed mode are: - changed conditions of feeding underground water with underground precipitations during the territory planning (changed module of runoff); - changed conditions of evaporation on hard-surfaced areas; - leakages from the water bearing utility lines; - additional drop irrigation due to wind carry-over of water from steam-drop exhaust of cooling towers; - operation of stratum drainages under the main buildings. As both natural and predicted level of ground water is much below than foundation bases and foundations of all buildings and structures, there is no need in civil dewatering and its impact on ground water is not analysed.
T1-41	Please provide the reviewers with local geological map and a soil profile of the site to be able to assess the studies on soil liquefaction.	Required materials are presented in section of 2.4 of SAR, and also BL-01377 pm, BL-01380pm and BL-01626 s/o, indicated in basic materials for SAR development.
T1-42	Text in chapter 4.1 indicates that the site is located on top of a terminal moraine, i.e., unconsolidated soft sediment. Have shear wave velocity profiles been obtained from the site to account for site conditions in the seismic hazard model? What is the thickness of the moraine? What is the potential of the soft-sediment in terms of ground motion amplification?	The soils are not unconsolidated and loose, but they are disperse type grounds according to GOST 25100-2012 "Soils. Classification" All soil column mass of the compressed zone was investigated and split into engineering-geological elements in terms of composition, state and physical and mechanical properties. All unsuitable soils are deleted from the building and structures subsoils. A set of seismic surveys was done on the site using methods of vertical seismic profile shooting in wells and surface seismic profile shooting on refracted waves with determined speed of longitudinal and transversal seismic waves. Results of these works were used during seismic zoning and are provided in section 2.4 of SAR. They are provided in full scope, i.e. with described types, volumes, methods and techniques, actual material, processing results and interpretation in the reports: 45833 c/o, 45837 c/o, БЛ-00521 c/o, indicated in basic documents of SAR
T1-43	Did the seismic hazard assessment include detailed site and near-regional investigations as described in IAEA SSG-9 and/or WENRA 2016 (i.e., detailed geological, geomorphological, geophysical and paleoseismological investigations)?	The analysis of seismic hazard is made in accordance with the requirements of NP-031-01 which overlap the requirements of IAEA Guide of SSG-9 and/or 2016 WENRA
T1-44	How sensitive is the seismic observation network, i.e., what is the smallest magnitude of a local earthquake which an be recorded and localised by the network?	Mmin от -0,5 Oshmyany PES zone earthquakes to 1.0 from PES zones of the nearest area.
T1-45	Pages 45ff: Please provide explanation on how "seismic category I", "seismic category II" and "seismic category III" should be understood.	Seismic resistance categories are assigned according to provisions of NP-031-01
T1-46	Is there a map of the seismic monitoring network round the NPP available?	Yes, we have. It is specified in section 2.4 of SAR
T1-47	Which parameters are automatically recorded by the seismic monitoring network?	Date, time, amplitude and shift period of seismic event
T1-48	3.1.2 NPP Protection under OBE and SSE Q1) How many OBE is assumed during the operational lifetime? Service levels for OBE? "The systems and components required for the RP safe shutdown and their functions (depending on operation conditions - NO, AOO, DBA and BDBA) are given in Table 3.1.2.1." It is not indicated in the table.	**1)** Earthquake of OBE level is defined as an earthquake of maximum frequency of occurrence 1 time in 1000 years. With nuclear power plant service life of 60 years no OBE occurrence is predicted. The question is not quite correct. **2)** question about "Q1) How many OBE is assumed during the operational lifetime? Service levels for OBE? " is not clear. **3)** see reply to comment T1-50 (note is missing in the table).

T1-49	Pages 46-51: Table 3.1.2.1 lists SSCs required for safe shutdown without providing information on the seismic capacity of the individual SSCs. Please provide such information (seismic capacity of individual SSCs). How large are the safety margins of the SSCs to withstand loads above those of the SSE earthquake? To what seismic level is the Ultimate Heat Sink (UHS) qualified (both in normal mode and emergency mode)? What is the seismic margin of the UHS (in normal and emergency mode) above the SSE level? What is the seismic resistance of the 7-10 km long pipes from the Neris river basin to the plant and of the associated pumps? What is the seismic resistance of the spray pools system, and of the emergency heat removal tanks?	All SSC used for safe shutdown refer to the I seismic resistance category and designed to suit seismic level DBE (SL-2). SSC margin is defined as limit value PGA = 0.13g. The ultimate heat sink (UHS) when in operating mode refers to the II seismic resistance category and is designed to suit level PGA=0.06g. These SSC systems include water supply systems from Neris river. These systems are not reviewed in stress-test conditions. Spray pools are used as UHS for emergency mode. The spray pools are designed assigned to the I seismic resistance category to suit DBE (SL-2) level. PHRT systems and spray system are assigned in design to I seismic resistance category, accordingly design seismicity level is DBE (SL-2)
T1-50	Pages 46 - 51 - Meaning of the asterisks?	Note to the table is missing. Value of applied designations is as follows: *) – necessity and sufficiency at NO and AOO; **) – necessity and sufficiency at DBA; ***) – necessity and sufficiency at BDBA.
T1-51	How to compare with Single Failure Proof Cranes Compliant with ASME NOG-1, NUREG 0554 & NUREG 0612?	The project was developed as per TA for NPP with account for Russian regulations
T1-52	Table 3.1.2.2 The fire extinguishing system is a critical system in case of earthquake, as demonstrated for instance during the Kashiwazaki Kariwa earthquake. Is it correctly understood that fire-fighting systems referring to seismic category II and III are not designed to withstand an SSE level earthquake? What is the protection concept for internal fire subsequent to an SSE earthquake? Does the protection concept credit the availability of a fire-fighting system?	Fire-fighting water supply system (indoor fire-fighting pipeline inside the buildings) is assigned to seismic resistance category III as per NP-031-01. In case of an earthquake with intensity of up to DBE (SL-2) inclusive and in accident modes water will be taken from fire reservoir within the area of FERU-2 of the Belarussian NPP by fire engines during fire to ensure outdoor and indoor firefighting (when necessary) by fire teams. Water fire-fighting systems are not designed for the DBE (SL-2) level earthquake. Equipment and elements of automatic gas fire-fighting systems in the main buildings to protect system elements assigned to I seismic resistance category as per NP-031-01 are compliant with this category. Fire confinement principle is used along with using active fire suppression systems to handle the rooms of top fire hazard rating and using active fire protection systems as a rule only to reduce material damage and ensure personnel safety in case of fire. When fire confinement principle is used stationary fire-fighting systems are not considered for safety substantiation because safety must be ensured by reached level of passive fire protection. Fire protection as provided for by design is based on combined principle of fire confinement (passive protection) and principle of impact on fire (active protection). Fire confinement principle assumes that during fire all combustibles in the fire z
T1-53	In case of an earthquake above the design basis, fire-fighting tanks are available but not the associated piping system. How will the function of the system be ensured?	According to RF normative documents fire water pipeline refers to seismic resistance category III as per NP-031-01. Fire confinement principle is used along with using active fire suppression systems to handle the rooms of top fire hazard rating and using active fire protection systems as a rule only to reduce material damage and ensure personnel safety in case of fire. Fire protection as provided for by design is based on combined principle of fire confinement (passive protection) and principle of impact on fire (active protection). Fire confinement principle assumes that during fire all combustibles in the fire zone may get burnt, at the same time fire resistant enclosing structures of fire zone (or fire safe distances) will ensure fire confinement until fire is finally suppressed. Fire confinement principle excludes a possibility of fire spreading to other fire zones through ventilation systems, common drainage systems, interrelated electrical network and other common utilities. At this design stage minimum degree of fire resistance of fire zone boundaries is accepted in a similar way to calculations and substantiations conducted for Leningrad NPP-2, namely - REI90.
T1-54	Information required in accordance with ENSREG ST Specifications is absent in Ch.3.1.3 of NSTR “Compliance of the plants with licensing requirements”. The Section should be revised.	Section is called “License compliance of NPP” and section name corresponds to its content.
T1-55	This is part of the justification that the design has been made for Design Basis Earthquake design input.	It is exactly so, license requirements establish necessity for design considering DBE (SL-2) and design meets these requirements. Cl.3.1.2 of National report: “The accepted design solutions provide the relevant seismic inventory of power unit buildings and structures according to the accepted DBE (SL-2) level”.
T1-56	"The regular actions of the operating organization personnel..." How the procedures comply with: Safety Reports Series No 66, Earthquake Preparedness and Response for Nuclear Power Plants, IAEA, Vienna, 2011, ISBN 978–92–0–108810–9 IAEA, 2012, see also Pre-earthquake Planning and Immediate Nuclear Power Plant Operator Post-earthquake Actions, Regulatory Guide 1.166, U.S. NRC, (1997) Guidelines for Nuclear Plant Response to an Earthquake, Rep. EPRI-NP-6695, EPRI, Palo Alto, CA (1989) Guidelines for Nuclear Plant Response to an Earthquake, EPRI Technical Report 3002000720, October 2013	OKB Gidropress uses only limited measures as defined in project to achieve safe and controlled state in scope of TA for RP/NPP. In addition, see T-3-59.
T1-57	"Additional actions of the operating organization personnel to provide..." Does the procedure define the rules for seismic housekeeping like in “Benchmarking for Seismic Housekeeping at Nuclear Power Plants: Compilation of Industry Practices.”, EPRI, Palo Alto, CA, and Seismic Qualification Utility Group (SQUG): 2008. 1018352.?	No additional actions by the personnel to bring the RP to a safe state are required at seismic impacts. When a seismic impact reaches the set point of the emergency protection actuation (6 points), the RP is automatically shut down. Actions of the personnel are specified in the instruction for emergency response at initiating event “emergency protection actuation”. Within the scope of the operational documentation, “Instruction on operation of the automatic monitoring signal subsystem of the automated stationary system for monitoring of technical condition of NPP civil and process structures for the Belarusian NPP facility. Power units 1 and 2” and “Instruction on operation of the automated periodical (unscheduled) monitoring signal subsystem of the automated stationary system for monitoring of technical condition of NPP civil and process structures for the Belarusian NPP facility. Power units 1 and 2” will be developed. These instructions will consider in particular the issues of maintaining seismic resistance of the NPP.

T1-58	Pages 60-61: - To what PGA does MSK 8tpts correspond? - What max PGA can the ECCS resist? (0.162g?) - What max PGA can the SFP racks resist? (0.144g?) - What max PGA can the MCR resist? - What max PGA can the ECR resist? - What max PGA can the EDGs resist?	An earthquake with intensity of 8 points in MSK-64 scale corresponds to a level of accelerations 0.2g. ECCS system meets strength criteria as per PNAE-G-002-86 and NP-031-01 applicable for an earthquake with horizontal peak ground acceleration up to 0.162g inclusive, fuel pool racks - with an earthquake of up to 0.144g inclusive. DG, EPSS, MCR, ECR are designed for horizontal peak acceleration 0.12g (DBE (SL-2) level).
T1-59	How is the result of the seismic margin assessment (macroseismic intensity 8 points) converted into ground motion values (PGA) that can be used for seismic design (design spectra)? What is the basis of the conversation (empirical correlations, expert judgement, or else)?	Seismic impact with intensity of 8 points corresponds to acceleration level PGA=0.2g. Seismic resistance margins for RP equipment and pipelines for the Belarussian NPP were assessed in topical report 491-Pr-1975. Assessment of margins was done using seismic resistance substantiation results of referent project RP V-491 for LNPP-2 which proved to be to equal to DBE (SL-2) 7 points in MSK-64 scale (PGA=0,12 g). In the course of these assessments input of seismic impacts into total design values of provided stress values was analysed. When seismic impact is increased as compared to design one, input of seismic impacts will increase in proportion to impact level. In this way seismic resistance of RP equipment and pipelines was analysed with DBE (SL-2) 8 points (PGA=0,24 g) and elements were found which do not have required margins to accommodate such loads.
T1-60	The reactor upper unit is provided with a 10% seismic margin regarding the SSE level of 7 points: is it correct to understand that the seismic margin is PGAH=0.12g + 10% = 0.132g?	Yes, for upper reactor unit maximum value of PGA is 0.132g. In cl. 3.2.1.1 it is noted that metalwork of reactor upper unit has the least margin (equal to 10%) relative to DBE (SL-2) design level of 7 points out of RP equipment and pipelines. Pipes between cross arm and top plate
T1-61	Is the reference [32] dated back 1975? Is it an evergreen study? MSK-64 intensity grade (ball) 8 means 2 m/s2 (~0.2g) PGA according to 03-NP-031-01. - Q) Is the margin per design approximately 100%. - Plant level margin should be 40% (or 67%). The justification of sufficient margin is scarce.	1975 - it is number of the report. The report was released in 2016. Assessment of 8 points in MSK-64 scale really corresponds to PGA=0.2g. Reactor plant is designed with seismic resistance margin also when additional fixation rods are used. Main RP equipment - reactor (except for SFP metalwork), SG, RCPU, RCP, Pressurizer, electrical connection block, connecting pipeline really have a two times seismic resistance margins. General seismic margin level for the NPP in [%] relative to seismicity of the site is (0.13-0.069)/0.069=88%
T1-62	The report states that "The safe-related electrical equipment refers to seismic category I as per NP-031-01 and maintains operation ability under an earthquake of the 7-points level as per the MSK-64 scale." Does that mean that safety related electrical equipment has no margin above the SSE level? Please confirm the max PGA that safety related electrical equipment can resist.	Electrical equipment is designed for peak horizontal acceleration 0.12g (DBE level adopted in the design basis). DBE level is set equal to 0.1g for the site. Values of peak horizontal accelerations (PGA) obtained as a result of field research during seismic microzoning were less than 0.1g (0.069g). Consequently, electrical equipment margin (in terms of seismic resistance) relative to the site DBE is 20%, relative to the site seismic conditions - over 70 %.
T1-63	Page 62: The report states that "Taking into account the accepted resistance margin for the equipment and pipelines the maximum admissible acceleration is 0.12 x 1.07=0.13g." (See also page 64 above.) Please list precisely which safety-related Systems Structures and Components that have this max PGA of 0.13g.	This is a common conclusion for all SSC designed as per PNAE-G-002-86 norms. It is important to keep in mind that in case of failed supply/discharge pipelines operability of the equipment item does not make sense. Conservatively, the conclusion covers all SSC of the safety systems. In cl. 3.2.1.1 it is noted that metalwork of reactor upper unit has the least margin (equal to 10%) relative to DBE (SL-2) design level of 7 points out of RP equipment and pipelines. Pipes between cross arm and top plate of reactor upper unit meet strength criteria as per PNAE-G-002-86 and NP-031-01 applicable for an earthquake with horizontal peak ground acceleration up to 0.132g inclusive. Other RP components design seismic resistance margins are higher.
T1-64	"Under seismic impacts the major power factors..." This isn't an absolutely correct argumentation. "The safe-related electrical equipment..." The NP-031-01 does not refer to standards that prescribe how to perform the seismic qualification of active components. See Regulatory Guide 1.100 and other international or national regulations.	Electrical equipment is certified in terms of seismic resistance as per NP-031-01 using by experimental and (or) calculation methods. To confirm seismic resistance by experimental way equipment items must undergo vibration resistance and vibration strength tests. Equipment items of I seismic resistance category are tested while subjected to real or harmonic loads equal to seismic impact at DBE (SL-2). Equipment items are tested in assembled, mounted, adjusted and operable state in the mode which imitates operating state. Parameters of load modes when tested are monitored in the base of equipment item fastening. Equipment item must be fastened to the test bench plate similar to fastening way when operated. Governing standards for techniques are: GOST 17516.1, GOST 16962.2, GOST 30546.1, GOST 30546.2, GOST 30546.3, GOST 30630.1.8 (IEC 60068-2-57:1989), GOST P 53166 (IEC 60721-2-6:1990).
T1-65	"During Seismic Margin Assessment (SMA) for buildings and structures ..." It is correct. However the Belarus NPP site-specific response spectra is not comparable with PSHA median or with the NUREG/CR-0098 spectra. Therefore is not clear whether the 84% site-specific and the selected design response spectra would have margin compared to PSHA median one.	This is a correct comment but this margin (reduced median values of spectrum with 50% probability) is provided only for CR-0098 as an example and is not used during final assessment of the NPP margin.

T1-66	The reactor developer recommends to improve the seismic resistance for several systems e.g. ECCS, pressuriser injection and discharge pipelines, etc.. Are those recommendations followed up by the regulator?	Designer of reactor plant showed that seismic resistance of reactor plant can be increased with the use of additional measures. Need for increase of seismic resistance must be justified. With reference to the Belarussian NPP site seismicity level does not reach the values which require increased seismic resistance. According to the project all RP equipment including spent fuel pool is designed to suit design basis earthquake (DBE (SL-2)) level of 7 points in MSK-64 scale, operating earthquake (OBE) - 6 points what corresponds to Technical assignment for NPP. At the same time based on test results done as part of stress-tests it is determined that most of reactor plant systems satisfy strength criteria with level of 8 points. According to cl.3.2.1.1 of National report: “main equipment of the reactor plant...have the necessary margins for load accommodation at 8-point earthquake”. Resistance conditions under an 8-point earthquake are not provided for the emergency core cooling system (ECCS), injection and discharge pipelines and pressurizer system, metalwork of the reactor upper unit, spent fuel pool, RCPU anti-seismic fixation rod”. At the same time for ECCS, pipelines for injection and discharge of pressurizer system, metalworks of the reactor upper unit, spent fuel cooling pool, RCPU aseismic fixation rod the relevant seismic resistance margins relative to DBE (SL-2) 7 points are determined. Results of the targeted safety reassessment of the Belarussian NPP have defined This way, as insufficiency of safety of the Belarusian NPP was not identified , it is not necessary to implement the proposed measures resulting from the stress-tests. At the same time, as set forth in i 8.2 of the National Report ,adhering to th
T1-67	The report states that the max PGA that the "main structures" can resist is 0.62g. Which buildings are concerned here (in particular, the containment building has a lower resistance as stated on page 64)? Please list precisely the max PGA of each safety-related building.	The indicated level is determined for all buildings of the nuclear island; that is for all the buildings accommodating safety systems. The only exception is the inner containment. Its ultimate seismic resistance capacity is determined as 0.51g considering the requirement for leak-tightness.
T1-68	"E.g. for the reinforced concrete structures relative attenuation value is 0.10, not 0.07 as for the SSE analysis." It is true, if the stress level is higher than the yield, see IAEA SRS 28. Is it?" - When K1=0.625 the seismic load reduction factor is..." See the acceptable values in SRS 28 TABLE III.2. These are differing from those given in the text (and in the referenced standards). "At the same time, the buildings bars are fractured under 70 (elongation at fracture..." It is not correct. "The mentioned-above margins prevent the possibility of cliff-edge effect..." There is no objections: The design is conservative, but the margin is unknown. In the simplest case the success path for ensuring the basic safety function have to be modelled. That is consisting active and passive components, and the weakest element in the Min-Max model will define the margin. " In this case the maximum acceleration value must not exceed 0.62g." Might be, but it is not a justification. It is a qualitative judgement valid mainl	**1)** An increased attenuation value in a structure is indeed realized if the yield point of steel reinforcement is higher. This is implied in the sentence: "large inelastic deformations in structures are allowed". **2)** The comment is not accepted. For reinforcement steel A400 (A-III) as per GOST 5781 (reinforcement of civil reinforced concrete structures) the elongation at fracture is taken to be not less than 0.14; the elastic elongation is adopted as per SP 63.13330: 0.002. **3)** In this case, when evaluating the limit value not all of the described safety margins are used; thus the determined value of the seismic safety margin is not the ultimate limit value. Besides, from the point of view of structural mechanics, the building structures constitute a statically indeterminate system, and when one of the elements fails the internal forces are redistributed. Thus, the cliff-edge effect for the whole building does not occur. **4)** In this case that is exactly the passive SCs, building civil structures, that are considered. The 0.62g value is not determining for seismic resistance of the NPP as a whole; thus the shown safety margin is sufficient for the stress-test assessment.
T1-69	It is stated that the seismic robustness of piping and pipelines important to safety limits the overall seismic margin of the NPP to 0.13g. Please describe the accident scenario that potentially results from an earthquake load exceeding this value of 0.13g. Which SSCs important to safety will be lost? Does the 0.13g value correspond to a cliff-edge?	In this case the seismic margin is considered based on the regulatory requirements imposed for all the SSCs designed under PNAE-G-002-86. This value can be specified following the results of a seismic walkdown inspection after the NPP commissioning.
T1-70	"Thus, the determining factor in assessment..." This isn't the margin of the plant. This is again margin of some passive SCs.	In other parts of the Report it is shown that the seismic safety margins for other components (civil structures, reactor plant) are higher. Thus, the 0.13g level is the minimum permissible, and that is why it is considered as the safety margin of the NPP as a whole.
T1-71	"By calculations the containment withstands the load 0.324g (2.7 times higher than the SSE load) under design strength criteria" Is this valid for SSE+LOCA or for SSE only?	The considered design combinations are in compliance with ASME BPVC; they included SSE+LOCA.
T1-72	"Thus, threshold seismic acceleration Amax is 0.12g * 1.1 * 4.54 = 0,6 g" There is no objections to the conservative character of the whole design. Nevertheless, this are some arguments for justification of the qualitative statement regarding conservative design but it isn't the evaluation of the margin.	By design the conservative approach is applied. When assessing the ultimate bearing capacity (stress-tests) excessive conservatism is avoided; a possibility for inelastic operation of a structure is assumed. This is a basic approach for carrying out the stress-tests.
T1-73	"Based on the calculations made in 1972 by the Central Research Institute..." Is the reference correct? (29. The Code of Administrative Offenses of the Republic of Belarus of April 21, 2003.)	This is a misprint; the document which is meant is entitled: “Report. NPP in the Republic of Belarus. Hydraulic and mathematical simulation of the water intake structures of the NPP service water supply system”. Central Research Institute for Complex Use of Water Resources. Minsk, 2013. Arch. №БЛ-01423с/о (reference number in JSC NIAEP).
T1-74	Is the proposed reassessment of seismic margins using the SMA method ongoing or planned?	In compliance with the IAEA recommendations periodic (once every 10 years) reassessment of safety of the NPP is planned.

T1-75	The most important issues in case of an earthquake are the safe shutdown of the plant and the long-term removal of the decay heat. Corresponding to the first sentence of chapter 3.1.2 (page 45) "All equipment of the NPP required for the RP safe shutdown refers to seismic category I (designed for SSE)" and in table 3.1.2.1 the "Systems and the elements components required for the RP safe shutdown" are listed. In chapter, 3.2.1.4 is stated (page 63): "For the buildings and structures referred to seismic I the ground seismic acceleration which exceeding may result in immediate damage is 0.62g." However, the following sentences make limitations. E.g., the first sentence on page 64 says: "For the equipment and piping the maximum admissible acceleration is 0.13 g considering the accepted safety margin." - Q1) What is meant with accepted safety margin? Q2) What are the SSCs on the safe shutdown path with the lowest seismic resistance and Q3) which seismic impact (PGA) do they cope with to keep their integrity and as far as necessary also their function? Q4) Is it in the minimum 0.62g and 0.51g for the inner containment integrity? - Important is the minimum seismic resilience of the whole shutdown path (including all insofar necessary SSCs). An	**1)** The accepted safety margin means the design margin as per PNAE-G-002-86; **2)** For safe shutdown, the SSC of the first seismic category have been singled out; **3)** The design seismic impact level PGA=0.12g, at PGA=0.13g operability is maintained; 4) Leak-tightness of the containment is maintained at PGA up to 0.51g 5) To remove heat from the reactor plant seismic resistance of the SSCs of seismic category I is required. 6) Seismic resistance of the spent fuel pool is required to prevent accidents in the SF pool.
T1-76	"It is proposed to reassess seismic margins..." The SMA is focusing on the assurance of the basic safety functions (success path + reserve) and not necessarily on the Seismic Category I, only.	In this case it is said about the critical value (0.13 g) determining the lower limit of the NPP seismic stability. The assessment shall be performed for the systems ensuring safety during an earthquake. These are systems of seismic category I. When reassessing seismic margins, following SMA method, a detailed list of SSC to be reassessed will be specified. Equipment and pipelines of the reactor plant involved in the transfer to the sub-critical state of the reactor and heat removal from the core after a seismic impact exceeding OBE level refer to seismic category I.
T1-77	Highest and lowest fluctuations of Viliya water levels on Figure 4.1.1.2 (p70) seem to not correspond with estimated probability of exceedance in Table 4.1.1 (p67). Please clarify this or correct if necessary.	Table 4.1.1 (page 76) shows average annual water levels with various probability, i.e. the design values, and Fig.4.1.1.2 (page 79) shows the observed water level behaviour in the specific year of 2015. These are different things. Fig. 4.1.1.2 is given to illustrate the conclusions that: 1. There is no correlation between the ground water level at the NPP site and the level in the Viliya river; 2. There is no correlation between the ground water level at the NPP site and precipitation, i.e. the site run-off factor is close to $\omega \approx 1.0$; 3. There is no correlation between the precipitation in the NPP area and the water level in the Viliya river at the water abstraction point of the unit pump station.
T1-78	Tables 4.1.1 and 4.1.2 show probabilities of exceedance up to 10-2 per year. WENRA 2014, however, requires to consider floods with exceedance probabilities not higher than 10-4 per year. Have these flood levels been determined? Are they relevant for the water intake structure? Can high water flow in the Viliya river cause clogging of the cooling water inlet in the river? If yes, which measures have been taken to avoid clogging by debris, wood, leaves etc. during high water situations?	Water loss in the source for making up the circulating cooling water system does not lead to violation of the NPP safe operation limits. As the circulating water system (cooling towers) is used, loss of the make-up will not result in instantaneous loss of the ultimate heat sink. The NPP personnel will have time to take the required measures for shutdown of the unit. The ultimate heat sink for the systems of cooldown at emergency shutdown are spray pools. The make-up water system (GA) is a non-safety related normal operation system which refers to safety class 4 as per NP-001-97 (OPB 88/7), seismic category II as per NP-031-01. In accordance with i.2.1.7 of RD 210.006-90, the highest water level in the river with probability of exceedance 0.01% corresponding to 127.790 m BES elevation is assumed as the design water line. In accordance with i.14.2 of SNiP 2.04.02-84*, the grading elevation of the site is assumed as equal to 130.150 m, which is 0.50 m higher than the design maximum water level in the river with account for wind surges and height of wind wave swash against the bank side. At the design documentation development stage, hydraulic and mathematical simulation of the scoop water intake structure operation and velocity structure of the flow in the water intake structure for hazardous hydrological phenomena was performed (at spring flood levels and frozen frazil slush phenom
T1-79	The design elevation of the NPP site is 179.3 m, the ground water level is said to be between about 157.8 and 162.67 m. What ate the elevation of the basements of safety-relevant buildings of the NPP?	The foundation depth elevations for safety-related buildings are within -13.700…-5.000 range (absolute elevations 165.6/174.3
T1-80	There is not a formalized reference water level applied for the design. This should be provided. The methodologies used for the characterization of the hazards of flooding depending on their origin are not presented in the report.	For civil engineering part of the design, the ground water level at site was conservatively assumed up to foundation elevation 0.00.
T1-81	Regarding the dam failure, the conclusions are made on the basis of studies conducted in 1972 by "CNIIKIVR". The corresponding studies have to be reviewed taking into account the last data and knowledge about changes in the region able to modify spreading of the released water.	Known changes in the region able to modify spreading of the released water since construction of Vileisk reservoir dam have no significant impact on the design level in the river of Viliya at the water abstraction point of the NPP unit pump station. A break wave from the dike at a distance of 30 km will mainly flatten out. Therefore, in the transboundary area of the river of Viliya the impact of this wave will insignificant - without considerable changes in the natural hydrological conditions of the river of Viliya. Besides, the volume of water in Vileisk water basin has reduced due to sludge setting. The calculations are implemented using the most efficient design non-stationary mathematical model based on numerical solution of hydrodynamical equation for continuous and rip flow (undertow) streams. These models are presently used for such calculations for example using a special software MIKE FLOOD 1D/2D River with additional modules for calculation of dike breaking (developed by Danish hydrological institute DHI). Morphometric characteristics of the Viliya river from dam of the Vileysky water reservoir to frontier with Lithuania also were not subjected to significant changes within the years elapsed after construction of the Vileysky water reservoir in 1976
T1-82	"...calculations made in 1972 ..." No objection regarding conclusion, but the study could be updated.	see T1-81
T1-83		
T1-84	How long is the time period for which measurements of the ground water level below the site are available?	Observations of the ground water dynamics were started at the site selection stage in 2008. Further observations were performed at the design documentation development stage and have been continued up to now under the integrated environmental monitoring program (Book 1 Section 3).
T1-85	Q1) It is correct, that the data shown in figure 4.1.1.2 represent randomly the year 2015. Q2) Is it further correct that data of one year can not really proof whether there is a correlation between the fluctuations of the level of the Viliya River and the fluctuations of the groundwater level on the NPP site.	It is correct, that the data shown in Figure 4.1.1.2 represent randomly the year 2015. Q2 - it is incorrect, as water is the most dynamic fluid in the nature. If a correlation between level dynamics of different water bodies exists, it is revealed within one hydrological year.
T1-86	In the figure there is a jump in the ground water level (December-January).	Due to climatic conditions in 2015, intensive snow melting occurred in December. However, a temporary rise in the ground water level did not exceed the long-time average annual values, let alone the forecast values.

T1-87	The report presents the provisions of protection against flooding taken in the design: drainage, gutters, storm water drainage system. The conclusion not having floods in the compartments of the site in case of unavailability of some of these devices (power loss) is not justified. No data is provided on the considered scenario, the intensity of the hazard, the duration of the phenomenon etc.	In this case, for floods in the compartments of the site located below elevation 0.000, a conservative scenario of complete flooding was considered. It was shown that NPP safety is ensured in this case as well.
T1-88	With regard to the requirement of the ENSREG ST Specification to report on Provisions to protect the plant against the DBF, i) Main operating provisions ..., and ii) Situation outside the plant, including preventing or delaying access of personnel and equipment to the site, no information is provided. The report should present conclusive information on these.	There is no possibility of flooding for the site.
T1-89	WENRA 2014 and WENRA 2016b require to identify all possible sources of water for flooding analysis including precipitation, flash flood, snow melt, runoff directed to the site, large volumes of water stored in on-site tanks etc. Have these water sources systematically been identified and analysed? What are the design basis values (runoff, standing water height etc.) for the different design events with exceedance probabilities of 10-4 per year?	Precipitation, flash flood, snow melt, run-off directed to the site were considered within the scope of integrated environmental monitoring, their parameters are monitored (measured as part of monitoring observations).
T1-90	Please provide a micro-topography map of the site to be able to judge the protection against flooding by precipitation, snow melt etc.	Topographic maps of 1:10 000 - 1:25 000 scale are additionally provided. The required maps are presented in Appendices A, B, C of SAR Chapter 2.
T1-91	Pages 71-72: What is the capacity of the storm water drain system, and is the capacity adequate to protect against the design basis floods (exceedance probabilities 10-4 per year) obtained for precipitation, flash flood, snow melt etc.? Are the capacities of the drainage pumps adequate for such events?	The capacity of the storm water drain system is 700 m3/h. It is adequate to protect against the design basis floods (exceedance probabilities 10-4 per year). The capacities of the drainage pumps are adequate for precipitation (exceedance probabilities 10-4 per year).
T1-92	1. What is the Design Basis precipitation? 2. What is the maximum precipitation corresponding to T=10000 years (this value is missing in table 5.2.1.1)? What would be the resulting water level on site? 3. Are the storm water treatment system and the drainage systems needed to cope with the 10000 years max precipitation, or can the plant survive without them?	1. The design basis maximum daily precipitation is 160 mm. 2. The maximum precipitation corresponding to T=10000 years is 160 mm/day. In case of normal operation of the treatment facilities, precipitation will not hold at site. 3. The storm water treatment system is required in accordance with regulatory documents of the Russian Federation. In case of the system failure during period of precipitation corresponding to T=10000 years, the depth of precipitation will be 5.3 mm. Taking into consideration the relief on the NPP site, this storm water will partially soak into the soil and partially accumulate around gullies on roadways. Also, as the pavement around the buildings is 150 mm, underflooding of the NPP site will not occur.
T1-93	The report states "In case of electric power failure, the storm water treatment system and drainage systems will not operate." Storm is generally associated with an increased risk of Loss Of Offsite Power (LOOP). How does the plant avoid flooding of safety-related parts in case of LOOP caused by storm with heavy precipitations (in particular those systems located below 0.0 m level indicated on p74)?	The maximum volume of storm water calculated as per i.7.2.2 of SP 32.13.330.2012 Wdaily max=10×Hdaily max×(∑Fi×Yдi), m3/day where: Hdaily max – maximum daily precipitation equal to 160 mm with exceedance probability 0.01 according to report BLR1.P.130.&.&&&&&&.&&&&&.002.HF.0002B. Fi - water catchment area with various surface types, ha, Yдi - annual average run-off factor for liquid and combined precipitation for various surface types. - after commissioning of units 1 and 2: Wdaily max=10x160x104.4x0.37=61804.8 m3/day. In case of electric power failure and if the drainage pump station and treatment facilities become inoperable, part of this volume – 6908.16 m3 will stay in pumps and wells of the drainage systems. The remaining volume 54896.64 m3 will be distributed over the entire NPP area, and the depth of precipitation will be 5.3 mm. Taking into consideration the relief on the NPP site, this storm water will partially soak into the soil and partially accumulate around gullies on roadways. Also, as the pavement around the buildings is 150 mm, their underflooding will not occur.
T1-94	On page 15 of the NR is reported that the site is graded and that the absolute elevation is 174.5 to 182.7 m BES. In page 74 first sentence is reported that the "absolute elevation is 179.3 m". Q1) Is this elevation of 179.3 m BES the 0.00 m level of both NPPs and of all of their safety relevant buildings? If not, please specify in a plan the 0.00 m level of all safety relevant buildings of both NPPs in m BES. - The last sentence on page 72 of the NR says that the "perinatal pavement around the building is 150 mm high" (protection barrier against ingress of outside water). Q2) Are there any doors or other openings to outside areas in safety relevant buildings lower than these 150 mm above the outside elevations? And if yes, please give a list with all concerned safety relevant buildings and give information to the relevant openings or pipes are and how much they are below this + 150 mm level (protection barrier).	All underground utility lines located below the ground level are laid in tunnels. Tunnels approaching safety-related buildings also refer to seismic category I and are designed for the respective impacts. Elevation 179.3 is level 0.0 for both units. There are no openings below the pavement.
T1-95	The report states the compliance without further analysis. The report has to be completed to address the requirements of the ENSREG ST Specification, requiring moreover, that analysis has to cover all the plant states (reactor, pool) and the induced possible consequences (for example possible link of flood with fire events occurring due to short circuits created by water spreading).	The stress tests (list of initiating events) were developed in accordance with TKP 566-2015 (33130).

T1-96	With regard to the requirement of the ENSREG ST Specification to report on Plant compliance with its current licensing basis; i) *Licensee's process to ensure that off-site mobile equipment/supplies ...are available and remain fit* ; no information is provided, in particular related to the location of the mobile devices and of their support systems (fuel, oil, cooling) and their respective protection against flooding risk; ii) *Any known deviation and consequences in terms of safety; planning of remedial actions* ; no information is provided. The report should provide the relevant information on these aspects.	The mobile diesel generator station of the BDBA management system is located on the Unit site outside the buildings and structures on an open concrete pad in an unobstructed area.
T1-97	The report states "-in terms of afterheat transfer from the cooling pool: in 41 hours (following the results of calculations in section 5.1.2) arrange feed of SFP. This can be made by connecting in an unconventional means to two process connectors of JNB50 system located at the external side of the UJE building (at +0.690 and 0.730, with water pumped from LCU tanks via pump of the fire engine to JNB50 system piping and further to the coolant pool) with flanges and plugs installed; " Question: Are any operator action necessary on the fuel pool floor to fill up the fuel pool?	No personnel actions are required on the pool floor. In this mode the spent fuel pool is filled with water. Only make-up of the spent fuel pool is required for heat removal due to water boil-off.
T1-98	The report states in chapter 4.2.1 "In case of extreme precipitation, even if considering failure of the UGU pump stations, the level of water on site can only rise 5.3 mm, which due to 150 mm perimeter pavement around the buildings which eliminates the possibility of a design basis flood". As long as the buildings are sealed and the area is relative even around the plant side and has a continues slope to the river, the mentioned water level is plausible. Question: Due to heavy rainfall (precipitation> 50 mm within 12 hours or less) was roof ponding also evaluated, if the drainage of the roofs are clogged?	Roofs of the safety-related buildings are designed for crash of an aircraft at an angle of 45. In this case the (vertical component) load is 11 mN. The roof is provided with a parapet of 600 mm high. Water level cannot rise above 600 mm that corresponds to the load of 6 kPa. Thus, structural reliability of the roof is provided with a 1800 times margin.
T1-99	The list of analysed dangerous meteorological phenomena on page 76 does not mention low temperature, even if low temperatures are considered later in the chapter. Only high temperature is mentioned in this list.	The minimum observed temperature is 39,8C. The minimum design temperature is (E10-4) - 41,5C. The complete data are given in SAR, sections 2.3 and 2.8.
T1-100	Dangerous meteorological phenomena is providing details about frequencies for not so rare nor dangerous values. Perhaps this is to illustrate expected weather conditions. However, more attention should be focused on the extreme conditions. This is partially covered by next chapter 5.1.2 in the Table 5.1.2.1 and later on in the section about threshold analysis (Table 5.2.1.1). It would be much better to first introduce all extreme meteorological values with needed explanations.	The list of the considered dangerous and extremely dangerous atmospheric phenomena is accepted according to NP-064-01 "Accounting external, natural and man-induced impacts". Design values of different frequency are given in SAR, sections 2.3 and 2.8.
T1-101	In the report the value given for heavy snowfalls is precipitation> 20 mm. Is the dimension a printing mistake?	According to Belgidromet classification, a snowfall where precipitation is > 20 mm for 12 hours and less is a hazardous meteorological phenomenon. In the report everything is correct.
T1-102	It is not apparent from the descriptions how the effects of the different weather phenomena have been taken into account in regards of different consequences at the plant (However, the impact of some combinations of events are described in Table 5.1.2.1).	The question is not clear.
T1-103	In table 5.1.2.1 is shown, that the SG PHRS and the containment PHRS is essential to keep the plant in a safe state under extreme weather conditions (low temperature and wind). Under BDBA conditions the PHRS-tank is filled up via the JNB50 system pumping water from EHRT after 24h. Question: Is a water well available for long term operation of the PHRS , probably protected in a building seismic class 1, which could be used to refill the PHRS-tanks (EHRT) or LCU-tank with well water without using pipes outside of buildings?	The design does not provide for a well to fill the EHRT and LCU tanks.
T1-104	In the report it is argued that the combinations of rare (exceedance frequencies of approx. 10-4/y) weather events lead to very low exceedance frequencies below the typical screening criteria. As combinations of not-too-severe weather events (e.g. with exceedance frequencies of approx. 10-2/y) may have effects beyond the sum of the individual effects, also such combinations need to be assessed. In Table 5.1.2.1. of the report only a qualitative assessment of combinations of extreme weather events is provided. The listed consequences are typically loss of off-site power and loss of ultimate heat sink. At least loss of ultimate heat sink is a beyond design basis event. Thus this bounding qualitative assessment does not fulfil the ENSREG requirement to describe combinations of extreme weather events included in the design basis. The combinations of extreme weather conditions included in the design basis should be identified. This holds in particular for causally linked weather conditions (e.g. strong winds, heavy precipitation and lightning as a result of a storm passing over the site).	In the stress tests a case of total loss of external power supplies and simultaneous loss of ultimate heat sink is considered. It is shown that NPP safety is ensured in this case.
T1-105	In the first sentence on page 79 is mentioned: "Information on the development of a full-scale PSA-1 (…) are given in section 2.4 of the national report." Q) On which page can we find section 2.4 in the national report?	Misprint - in section 2.3.4 of the National Report There must be "in section 2.3.4" instead of "in section 2.4".

T1-106	In table 5.1.2.1 is reported, that in case of very high wind and extreme precipitation "failure of drainage systems due to water ingress into buildings of nuclear power plants or ventilation ducts" will happen. Furthermore loss of the high-voltage power lines and additional "a loss of power supply for own needs" will occur. The NPPs have to be shut down and the residual heat removal has to be done by the" SG PHRS and containment PHRS operation." Q1) are the mentioned conditions as wind stronger than 54 m/s and rain more than 101 mm/24 hours cliff edges? Q2) By which openings, pipes etc. the water ingress will happen in which safety relevant buildings?	In this case buildings of storm water drain systems GU (buildings and structures UGV, UGU, drain pump systems and treatment systems) are meant. Water ingress into safety related buildings does not occur.
T1-107	regarding PSA, the reference should be made to the Section 2.3.4.	Accepted. Reference shall be made to section 2.3.4.
T1-108	The assessment of safety margins is limited to a direct comparison between design values and loads resulting from events with exceedance frequencies of 10-4/y. Potential cliff-edge effects and the corresponding margins are not identified. Besides this, the argument that extreme weather conditions are covered by the design against blast wave and aircraft crash is only true for certain failure modes of building structures. With regard to the safety systems only qualitative information is given for the ventilation systems. Potential cliff-edge effects should be identified. The available margins between the design basis and the identified cliff-edge effects should be quantified. In particular for building structures, it should be verified (e.g. in the framework of a national action plan) that human-induced events cover all aspects of extreme weather conditions. In addition, safety margins for systems should be evaluated and quantitative information should be provided.	Safety systems located in the buildings are not exposed to direct impact of weather conditions. Extreme weather conditions can have impact on ventilation systems only, for which analysis was performed. From the point of protection function fulfilment by the building structures, the safety margin assumed with account for man-caused impacts allows to state that the threshold values are unreachable for natural weather conditions.
T1-109	Content of the Section does not comply with the title.	The comment is not clear.
T1-110	"For the Belarusian NPP project, the seismic impact and that of the aircraft, ..." The statement is correct regarding those structures that are designed for aircraft crash (e.g. main building). Several structures important for safety are exposed to severe whether conditions but not designed for aircraft crash. Is the diesel building designed for aircraft crash?	The standby diesel power station refers to seismic category I as per NP-031-01 and is designed for crash of an aircraft with weight 5,7 t and speed 100 m/s (calculation BL-21626s/o).
T1-111	It states on p 83 that "Supply and discharge pipes of the cooling water system of the PE essential consumers are placed in underground passageway tunnels of the UQZ and URZ safety systems, which excludes their freezing." How is this determined and for which freezing conditions? Please clarify in the document what are the UQZ and URZ safety systems and how they exclude freezing of the cooling water.	The tunnel floor slab is arranged at a depth from 2,5 to 3.7 m from the ground surface while the depth of freezing at the Belarusian NPP site is from 0,9 m to 1,3 m. The document contains description of the cooling water system for essential loads (RE). Tunnels of the UQZ and URZ safety systems are the PE system constructions designed for pipelines routing.
T1-112	What are the references or methods used for deriving values of extreme natural impacts in the Table 5.2.1.1 (p 84)? Why return values for 10000 years (or other appropriate values) are not provided for other dangerous meteorological phenomena like heavy rain and heavy snowfalls? Table 5.1.2.1 is presenting some more extreme values but without reference to frequency nor reasons for selection.	In the Russian Federation all methods are standardized. There is a relevan standard for each phenomenon. Parameters of the extreme natural phenomena are given in SAR, section 2.8. The maximum one-day rainfall with a probability of E10-4 is 160 mm. The snow load with the same probability is 3 kPa. The design water equivalent of snow cover is 270 mm.
T1-113	Table 5.2.1.1 gives design values and values for events with exceedance frequencies of 10-4/y, but no information is provided on how the values have been derived (e.g. statistical method used to extrapolate from the limited meteorological observations to rare events). This information is necessary to verify that the loads given for events with exceedance frequencies of 10-4/y are reasonable and reliable. Moreover, for wind loads it is not clear from the report whether the given values refer to mean wind velocities or to gusts.	In the Russian Federation all methods are standardized. There is a relevan standard for each phenomenon. Parameters of the extreme natural phenomena are given in SAR, section 2.8. The extreme wind speed (wind blast) is 54 m/s (SAR, section 2.8)
T1-114	Table 5.2.1.1 states that design minimum temperature is -61˚C while extreme temperature with frequency 1E-4 1/year is -50◦C. Clear extreme temperature limits are provided, for example, for mobile generator plant operation with minimum -50◦C (p 39). Please elaborate that in more details including relevance to other temperature sensitive safety systems (e.g. diesel supply for other DGs, cooling systems).	Since the the standby diesel power station/unit diesel power station are the heated buildings the fuel and cooling systems are protected from influence of low temperatures. As far as the intermediate diesel fuel storage is concerned, the equipment and pipelines are located underground, therefore low temperatures have no impact. "Analysis of cooling capacity of the spray pool under extreme temperatures" BL-12183 has been performed.
T1-115	It is stated on page 85 how impossible is to predict reliable supply of diesel fuel at a late stage of an accident. Could you please clarify that extreme low temperature is not affecting diesel fuel supply in the early stage with specific low temperature?	The design minimum temperature is -61 ◦C. As far as standby diesel power station/unit diesel power station are concerned, the fuel system is located in the heated building and protected from the impact of low temperatures. As far as the intermediate diesel fuel storage is concerned, the equipment and pipelines are located underground, therefore low temperatures have no impact.
T1-116	The considerations regarding margins with respect to meteorological extremes are rather qualitative. No objections: Design is conservative, but the argumentation is scarce.	The quantitative assessment of the margins is presented in table 5.2.1.1.
	In section 2.3.4 the application of Probabilistic Safety Analysis as a constituent part of the Safety Assessment is discussed. It is mentioned, that "For the Belarusian NPP, comprehensive PSA-1 (for internal initiating events, internal fires and flooding, seismic PSA and PSA for external impacts) and comprehensive PSA-2 based on PSA-1 are developed".	In section 2.3.4 of the National Report it is not stated that PSA-1 and PSA-2 are completed. Only completed PSA-1 for internal initiating events is mentioned. References to the PSA-1 completed works for the Belarusian NPP are given bellow. Results of PSA-1 for internal initiating events are given in document: - "Probabilistic Safety Analysis of level 1. Power unit 1. Belarusian NPP. ED -10819 пм Chapter 11 Quantitative calculations of nuclear fuel damage probability taking into account internal initiating events. BLR1.B.130.1.&&&&&&.1101&.022.HH.0001". Results of PSA-1 for internal fire are given in document: - "Probabilistic Safety Analysis of level 1. Power unit 1. Belarusian NPP. ED -10819 пм Chapter 12. Book 2. Internal fire analysis. BLR1.B.130.1.&&&&&&.1202&.022.HH.0001".
		Results of PSA-1 for internal flooding are given in document - "Probabilistic Safety Analysis of level 1. Power unit 1. Belarusian NPP. ED -10819 пм Chapter 13. Internal flooding analysis. BLR1.B.130.1.&&&&&&.1301&.022.HH.0001".

T1-117	The references to the completed PSA-1 and PSA-2 for the Belarusian NPP if it is available should be presented to the international nuclear safety community, or it should be clearly stated that these documents are still under development.	Results of PSA-1 for external hazards are given in document - "Probabilistic Safety Analysis of level 1. Power unit 1. Belarusian NPP. ED -10819 пм Chapter 15 Analysis of other external hazards BLR1.B.130.1.&&&&&&.1501&.022.HH.0001.»
T1-118	According the ENSREG specification of "Stress tests", regarding the earthquake it is very important to describe the Methodology for Design-Basis Earthquake evaluation, where the return period, past events considered and reason for choice should be presented. However, in the Report it is just mentioned, that "Historical and measured felt earthquakes of the western part of the East European Platform during 1602-2007 were listed based on literary and archival data". The past events were not specified. The information on earthquakes close to NPP at 1908 and 1987 (approximately 20 km from the center of the nuclear power plant site) is not presented. The earthquake at 30.12.1908 took place in Gudogai (at a distance of ~ 20 km from the Belarus NPP), the hypo central depths range from 9 km to 10 km, ML = 4.5, Io = 7. The earthquakes in the past should be mentioned and taken into consideration in the analysis.	The mentioned list includes both historical and instrumentally recorded earthquakes required and sufficient for probabilistic seismic analysis of the site (see comments T1-21 и T1-31). List of seismic events considered in the analysis is presented in section 2.4 of SAR
T1-119	Please, explain the applied methodology to determine the intensity values of DBE and SSE for scattered seismic activity and structured seismic activity?	See comments T1-36 и T1-38.
T1-120	It is stated (page 41) that assessment of „seismic level" for the Ostravets site is based on the "Fragment of the temporary map of seismic risk zoning of the Russian Federation OCP-97-D (1:10000000) with inclusion of the territory of Belarus)" and„- Maximum horizontal acceleration of the SSE level – 0.12 g (7 points as per the MSK-64 scale) ". Further on the page 44 it is stated that: "using a probabilistic approach, the evaluations of intensity of SSE reaching 6 points of MSK=64 scale for average soils were obtained." **Questions:** 1. The temporary map of seismic risk zoning of the Russian Federation OCP-97-D (1:10000000), was compiled in year 1997, i.e. before the Kaliningrad earthquakes of 2004 occurred and do not evaluate influence from the Kaliningrad earthquakes of MW=5.2 (Gregersen, 2007); Russian Academy of Sciences - magnitude of main shock - Mb=5.4. Please, provide more detail information on how the DBE and SSE values for Belarussian NPP have been determined on the basis temporary seismic risk zoning map of the Russian Federation OCP-97-D on a scale 1:10000000 of year 1997? Please, explain if the values of the strongest earthquake in the region, as high as MW=5.2 and Mb=5.4 have been considered in the assessment of DBE and SSE values? 2. From the provided information is not clear if the direct probabilistic seismic hazard assessment following the recommendations of NS-R-3, NS-G-1.6 and SSG-9 items 1.2 and 6.4. indicating „need for seismic hazard curves and ground motion spectra for the probabilistic safety assessment of external events for new and existing nuclear installations" has been carried out for the Ostravets site? 3. It is stated in the Report that, using a probabilistic approach, the evaluations of intensity of SSE reaching 6 points of MSK-64 scale were obtained, but, according to the GSZ-97-D map the Belarusian NPP site refers to the 7-points zone that corresponds to the level of SSE. Please, explain what is the exact value SSE in intensity points and explain reason for the choice? 4. Please, explain how probabilistic seismic hazard calculations for the Ostravets site following the recommendations of the IAEA documents and providing seismic hazard levels in terms of obtaining ground motion values (Peak Ground Acceleration) for the NPP design basis earthquake (DBE) and safe shutdown earthquake (SSE) values in the Ostrovets site have been carried out? 5. Please, explain how macroseismic intensity points of MSK-64 scale were converted to commonly accepted peak acceleration of soil particles (Peak Ground Acceleration, PGA) values?	1. The Kaliningrad earthquake was considered in DSZ (detailed seismic zoning). DBE was determined on the basis of map OCP-97D. In accordance with the established practice, OBE value is assessed lower than DBE by one point. All significant strong earthquakes of the region were considered in DSZ 2. Seismic hazard curves and ground motion spectra for probabilistic safety analysis of external events are under development 3. Seismic assessment of the site was performed by various methods, the final DBE assessment was based on a set of methods. The completed set of field, topical and calculation works on specification of geodynamic and seismotectonic conditions (specification of initial seismic activity) for an area with a radius of 300 km from the site (scale 1:500 000) and the nearest area with a radius of 30 km (scale 1:50 000) as a whole confirmed representativity of the design seismic impacts in GSZ (general seismic zoning). Based on the specified data on initial seismic activity, DBE (SL-2) intensity is equal to 7 points, OBE intensity is equal to 6 points with account for rounding of the obtained values of shaking to whole-number points as per NP-031-01. In particular, based on the deterministic assessment, DBE intensity is equal to 7.0 points, OBE intensity is equal to 6.0 points; based on the probabilistic assessment, DBE intensity is equal to 6.0 points, OBE
	It is stated that: „-Maximum horizontal acceleration of the SSE level – 0.12 g (7 points as per the MSK-64 scale) - Maximum horizontal acceleration of the DBE level – 0.06 g (6 points as per the MSK-64 scale)."	These figures refer to peak ground acceleration (PGA). At the same time it is equal to horizontal peak acceleration, vertical acceleration is considered equal to 2/3 of horizontal peak acceleration as per NP-031-01 requirements. PGA - horizontal peak ground acceleration

T1-121	**Questions:**	
	1. Please explain, if the term "Maximum horizontal acceleration" describes the same parameter as „the maximum peak (horizontal) accelerations (PHA)" further used in the same Report?	
	2. Please explain, does the term „PGA" describes the same parameter as „PHA"? Does the term "PGA" refer to horizontal peak ground acceleration, or peak ground acceleration?	
T1-122	It is stated that "the maximum peak (horizontal) accelerations (PHA) received by the results of field research during seismic risk zoning is < 0.1g (0.069g).<...> In the design bases the PHA value is 0.12g".	1) PGA=0.12g value was adopted for the Basic Design. All systems of seismic category I are designed for DBE with intensity 0.12g. 2) In this case the translation is inaccurate - DBE is literally translated as "design basis earthquake" but has a different meaning in the Russian terminology - "safe shutdown earthquake". DBE (design basis earthquake) corresponds to OBE when translated. PGA values for DBE is 0.12g, for OBE is 0.06g
	Questions:	
	1. Please explain, how PHA value of 0.12g for DBE was estimated?	
	2. Please explain, why different information on PGA values for DBE are provided in the Report (page 41), namely the statement "DBE level – 0,06g (6 points as per MSK-64 scale" is inconsistent with statement that "in the design bases the value PGA=0.12g <....> is accepted"?	
	3. Please explain, what are the exact value of DBE in terms of PGA?	
T1-123	It is stated that „In the design bases the value PGA=0.12g (The project VVER-1200, 2006) with a reserve 0.01g, i.e. 0.13g is accepted. Thus, for an extreme earthquake which exceeds the maximum values provided by the project of the Belarusian NPP actually the reserve of exceeding of seismic influences makes 0.03g or 30% in relation to the corresponding MDBE value."	
	Questions:	
	1. Please explain what parameter describes the term „MDBE" ? How the term „MDBE" is related to „SSE" parameter?	1) In this case the translation is inaccurate, there must be DBE (see the original text)
	2. Please, explain how the reserve of exceeding of seismic influences of 0.03g or 30% in relation to the corresponding MDBE value has been estimated?	2) The margin (reserve of exceeding of seismic influences) has been estimated in relation to design value 0.1g for the Belarusian NPP site. The peak acceleration at which NPP safety is ensured was determined during the stress tests and is equal to 0.13g
	3. Please explain how SSE, DBE and MDBE parameters are related to the seismic hazard levels SL-1 and SL-2 in terms of obtaining ground motion values (Peak Ground Acceleration)?	3) DBE parameters correspond to SL-2. OBE parameters correspond to SL-1. The term "MDBE" is a translation mistake, there must be DBE.
	It has to be mentioned that the information in the National Report does not fully comply with the information provided by Belarus in the previous documents, namely:	
	In the Environmental Impact Assessment Report of 2010-07-06, table 14, describing seismic resistance characteristics of two-unit NPP with power of 2340 MW, it was stated that: **"Maximal calculated earthquake (MCE) is - 0,25g** and **Project value (PV) 0,12g"**	0.25g value is incorrect.
	In the Report of the Bilateral Belarussian-Lithuanian experts' meeting 21-22, June, in Vilnius and 13-14 September 2016 in Minsk it was stated that:	
	"Belarus made the probability calculations of seismic hazard for the Ostrovets site. Probabilistic assessments of peak accelerations were made in accordance with the IAEA standards SSG-9 «Seismic Hazards in Site Evaluation for Nuclear Installations» and comprised: **0,05g of maximum design earthquake (MDE) level (SL-2)** and **0,035g of design basis earthquake (DE) level (SL-1).**"	These are correct values. 0.069g value is indicated in the National Report.
	1.3. In the answers to the questions posted by Lithuania to the National Report of Republic of Belarus for 7-the NSC Meeting it was stated that (abbreviation – Answers), Answer to question 98 it was stated that:	
	„The peak accelerations at 50% probability estimated by explosion method are as follows:	There is a mistake. There must be as follows:
	The SSE: horizontal component of 67.22 cm / s2 (0.069 g)	**DBE: horizontal component of 67.22 cm / s2 (0.069 g)**
	The SSE: vertical component of 44.8 cm / s2 (0.046 g)	DBE: vertical component of 44.8 cm / s2 (0.046 g)
	The OBE: horizontal component of 54.29 cm / s2 (0.055 g)	**OBE: horizontal component of 54.29 cm / s2 (0.055 g)**
	The OBE: vertical component of 36.2 cm / s2 (0.037 g).	OBE: vertical component of 36.2 cm / s2 (0.037 g)
	In accordance with NP-031-01 requirements for the newly constructed nuclear power plants, the peak acceleration values at the SSE level should be no less than 0.1 g, and at the OBE level – no less than 0.05 g. Therefore, the final scores are as follows:	
	The SSE: horizontal component of 0.1 g	
	The SSE: vertical component of 0.067 g	
	The OBE: horizontal component of 0.055 g	
	The OBE: vertical component of 0.037 g."	
	Please, provide the final information on the consistency of the evaluation of DBE, MDBE and SSE parameters and more detail information on the methodology to evaluate DBE, including return period, margins, validity of data in time etc.?	These are final values of peak accelerations for local site conditions of OBE and DBE levels.
	According to the Figure 3.1.1, the site for Belarusian NPP was selected in the most intensive seismic zone in the Belarus (with the intensity of 7 according MSK-64 scale and near two active seismic faults).	1) OBE value in the design is 6 points. DBE value is 7 points. Individual engineering survey assessments are given in pages 42-43. The final assessment is given at the end of the section. 2) The NPP is designed for maximum earthquake of over 7 points.

T1-124	1. Please, explain why Design basis earthquake value is 4 point while figure 3.1.1 shows the value 7 for this site?	
	2. Are there adequate measures taken in NPP design to withstand 7 points earthquake or only 4 points?	
T1-125	The information provided in the points 5-6 in the Sub-Chapter 3.1.1. of the Report on the determination and activity of the geodynamic zones and possible earthquake source zones (Pes) at an adjacent to NPP site could not be assessed on the basis of information provide in the Report, as no sufficient data on the distribution of these zones, methods and criteria of evaluation of their capability is provided in the Report.	1. Parameters of the XV and XIV level geodynamic zones exclude probable fast faults, thrusts, upcast faults, strike-slip faults and other crustal faults accompanied with strong oscillations and seismotectonic troubles 2. see T1-23, attachment T1-23. Maps of GDAZ and PES zones 3. The approved complete results are specified in section 2.4 SAR.
	The information about distribution of fault system in the Ostrovets site and its structural relationship with Oshmyany potential seismogenic zone and with capable Oshmyany fault is not clearly explained in the Report.	
	1. Please explain how the information (page 43) that "for the NPP site, 23 XV-level geodynamic zones and 185 XIV–level geodynamic zones have been <….> activated within the N-Q period (the latest tectonic movement)" comply with the information in the Report (page 45) that NPP site "exclude probable normal faults, thrusts, upcast faults, strike-slip faults and other crustal faults accompanied with strong oscillations and seismotectonic troubles"?	
	2. Please, provide consistent information about the fault system in the Ostrovets site and its structural relationship with capable Oshmyany fault and the other data proving the absence of potential for surface faulting and capability of faults at the Ostrovets site?	
T1-126	It is stated that „the PES zones were determined according to the seismic and geological data and evaluation (value) of their seismotectonic potential (Mmax). The value of Mmax for each zone has been defined as follows: by the magnitude of the strongest earthquake for this structure <...>." Following item 4.12 of the IAEA document SSG-9, the parameter M_{max} for intracratonic areas of low seismicity has to be assessed using commonly accepted safety margin of 0.5, e.g. $M_{max}=M_{max\_observed}$ + 0,5.	1. Mmax for each zone is determined with account for at least three factors: - In terms of magnitude of the strongest earthquake for this structure (with available seismicity); - by analogy with similar structures of other ancient platforms or with geostructures of this region (provided that recorded earthquakes are missing); - Based on design methods with Mmax monitoring approved for this domain as per zoning map PES GEZ-97D. 2. For Oshmyany PES zone Mmax = 4.5, which correlates with earthquake in Gudogai with an intensity of 7 points and a magnitude of 4.5. 3. The Kaliningrad earthquake in 2004 has MS = 4.3 and 4.5 based on different data. The Kaliningrad-Lithuanian seismogenic zone is located on the western continuation of a large Kurzeme-Polotsk fault zone, and comprises three subzones with Mmax from 4.0 to 4.3:
	1. Please explain, why in the deterministic seismic hazard evaluation for the two closest seismogenic zones (Daugavpils and Oshmyany) to the Ostrovets site the recommendations of IAEA document SSG-9 have not been followed?	
	In the EIA Report is provided information about two instrumentally recorded earthquakes in the in Oshmyany seismogenic source zone („on 17th October 1987 with epicenter located 10 km to the east from Ostrovets", and „on 27th February 1987 at 23:37:22 UTC time (magnitude 2.5, epicenter located 10 km to the east from Ostrovets) recorded by three seismic stations)" and historical Gudogai earthquake with intensity 6 to 7 (MSK-64 scale), that contradict the explanations of the methodologies adopted for deterministic seismic hazard assessment.	
	2. Please explain the methodology how the value Mmax =4.5 was obtained for Oshmyany PES zone and how it considers the data on the historical Gudogai earthquake of intensity ~7.0 that occured in this zone in year 1908?	
	3. Please explain the methodology how the Mmax = 4 value the Kaliningrad seismogenic zone has been determined, considering the Kaliningrad earthquake of as MW=5.2 and Mb=5.4 occuring in this zone in year 2004?	
T1-127	The estimated probabilistic intensity value of SSE for average soil conditions of 7.2 points exceeds the deterministic value of the maximum intensity of 7.	6. The completed set of field, topical and calculation works on specification of geodynamic and seismotectonic conditions (specification of initial seismic activity) for an area with a radius of 300 km from the site (scale 1:500 000) and the nearest area with a radius of 30 km (scale 1:50 000) as a whole confirmed representativity of the design seismic impacts in GSZ (general seismic zoning). Based on the specified data on initial seismic activity, DBE (SL-2) intensity is equal to 7 points, OBE intensity is equal to 6 points with account for rounding of the obtained values of shaking to whole-number points as per NP-031-01. In particular, based on the deterministic assessment, DBE intensity is equal to 7.0 points, OBE intensity is equal to 6.0 points; based on the probabilistic assessment, DBE intensity is equal to 6.0 points, OBE intensity is equal to 4.0 points. With the nearest PES zone 4 km away from the site, based on the specified data on initial seismic activity, DBE intensity is 7.2 points, OBE intensity - 4.6 points, which does not exceed the deterministic assessments. As shown, for OBE level the assessment of specified initial seismic activity of 6 points is conservative enough, and for DBE the assessment of 7 points is stable enough. PGA is assumed as 0.12g, which corresponds to both 7 and 7.2 points. Besides, the design provides for a margin of 0.01g and PGA = 0.13g.
	Please, explain why this difference has not been considered in final determination of SSE value?	
T1-128	Please, explain the applied methodology to conclude that „the man-induced changes of conditions, i.e. rising of groundwater level, <…>" will not cause significant changes in seismic activity values of the Belarusian NPP site determined for natural soil conditions."?	According to the seismic microzoning procedure by the method of seismic stiffness analysis, ∆J intensity increment in points due to the ground water rise is considerable if the ground water level is set within 3-10 m range from the ground surface. The higher the ground water level, the more the increment, and at a ground water level of 3 m it reaches ∆J+1 point. For the NPP site the ground water level is set at a depth of 17-22 m and its rise due to man-caused changes in feeding and draining is forecast to a depth of 14-18 m. Even in case of emergency leakages from the water bearing utility lines and a long delay in their elimination, the ground water level cannot rise higher than 7 m, as at this depth the stratum drainages are constructed. Under these hypothetical conditions ∆J is less than 0.5 points and rounded to whole numbers, remaining ∆J=7 points DBE.
T-129	Please, explain the applied methodology and data to conclude that „soil liquefaction at the site under DBE- and SSE-induced seismic loads does not occur"?	The method is described in report BL-01778 c/o, named in the basic materials to section 2.4 SAR. Information on the ground profile is given in reports BL-01377, BL--1626 c/o, BL-00368 c/o, BL-45836 c/o, named in the basic materials to section 2.4 SAR, and in section 2.4 SAR.
	Please provide more detailed information on the soil profile: e.g. on the type of soils in the site; soil settlement expected during an earthquake etc.?	
T1-130	It is stated that „<…> no local earthquakes were recorded."	No local earthquakes were recorded by local seismic stations in the nearest area (R = 30 km) of the Belarusian NPP during the period from 2012 to 2017.
	Please provide more detailed information on the time period since when no local earthquakes were recorded?	

	Please provide more exact information on the territory where no local earthquakes were recorded?	
T1-131	Please, explain what exactly SSE values in MSK-64 scale are adopted for the reactor unit including the cooling pool equipment? Please clarify if the possibility to adapt the V-491 RP Project for the Belarusian NPP site with 8-point earthquake intensity by the MSK-64 scale is or would be implemented in the construction of NPP?	A value of 0.12g is used in the design to determine DBE level. Possible measures to improve seismic resistance of the reactor plant are described in the report. At present the NPP has sufficient safety in relation to the site conditions. Designer of reactor plant showed that seismic resistance of reactor plant can be increased with the use of additional measures. With reference to the Belarussian NPP site seismicity level does not reach the values which require increased seismic resistance. In addition, see the answer to T1-66.
T1-132	Most of systems and elements listed in the Table 3.1.2.1 have one, or two, or three signs *), **), ***). Please provide information that is subject of these bookmarks.	Note to the table is missing. Value of applied designations is as follows: *) – necessity and sufficiency at NO and AOO; **) – necessity and sufficiency at DBA; ***) – necessity and sufficiency at BDBA.
T1-133	Table 3.1.2.1, *"PEA/ Spray cooling pools…"*	Table 3.1.2.1 will be revised in regard to system PEA, "trash screen, water purification machine" will be added instead of "spray cooling pools", "heat removal from the primary circuit" will be changed to "cooling water purification from trash".
	It is necessary to describe the volume and seismic category of Spray cooling pools.	Description of spray cooling pool structures and its classification will be provided. Each spray cooling pool is designed to cool two channels of system PE. It is an underground structure divided into two sections (based on the number of channels) with an open part and a closed part, which ensures preservation of water volume in the pool in case of tornado impact. Layout of a spray cooling pool is based on the following main principles: - provide necessary conditions for normal behaviour of the designed processes regardless of external impacts; - allow for maintenance and repair. The overall dimensions of each pool in the plan are 70.0x 120.0 m, depth 5.9 m. The design characteristics of the spray cooling pool (capacity, overall dimensions, type and nozzle arrangement) are determined by the thermal hydraulic analysis of all operation modes of the system, based on the necessity to ensure the cooldown of the reactor plant in the hot period of a year 10% at a wind with frequency of occurrence once in 10 years (regardless of the direction) and to ensure a temperature of cooling water supplied to the reactor compartment not higher than + 31°C. A covering is provided around the pool with a slope towards the pool. To maintain the water volume in the pool in case of tornado and prevent water carry-over during operation of the nozzles, a closed part 12 m wide is provided a
	Is it allowed to take out of service one of the sections of the spray cooling pool for maintenance (cleaning and revision) when the unit is under power operation?	Each spray cooling pool is divided into two sections to allow for maintenance and repair.
	It is necessary to indicate the time of filling of the emptied spray cooling pool.	The time of the spray cooling pool filling will be additionally indicated. The time of the spray cooling pool filling with demineralized water is 8 days (at pumping equipment capacity of 100 m3/h).
T1-134	The main RP equipment – reactor, steam generator, reactor coolant pump, reactor coolant pipeline, pressurizer and connecting piping – is provided with the required margins to withstand the 8-point SSE loads. For ECCS, injection and discharge pipelines of the pressurizes system, metalwork of the reactor upper unit and spent fuel storage pool the strength conditions under the 8-point SSE are not provided. For these elements the reactor developer gives recommendations to improve seismic resistance...	Both the reactor plant and surrounding equipment are designed for design loads. The safety margin of the reactor plant allows to withstand loads higher than the design ones. The emergency core cooling system shall withstand design loads, the safety margin in the design is limited by minimum regulatory requirements. The equipment designer (OKB Gidropress, also the RP designer) suggests possible measures to increase seismic resistance. Application of the measures to increase seismic resistance shall be substantiated. At present the NPP has sufficient safety in relation to the site conditions. The main RP equipment - reactor (except for the spent fuel pool structures), steam generator, reactor coolant pump unit, reactor coolant pipeline, pressurizer - have significant seismic resistance margins as they were initially designed for higher seismic impacts as compared with those adopted for the Belarusian NPP to allow for their adaptation to other potential sites with high seismic impact level without significant modification of their design. Seismic resistance of the other RP components can be increased by installing additional antiseismic supports on the pipelines, reactor upper unit, strengthening of the ECCS tank support structure.
	Please, explain how it could be that reactor and surrounding equipment can withstand 8-point SSE loads, but emergency cooling system equipment cannot withstand the same? Should emergency cooling system withstand higher loads or the same at least? Even reactor developer recommends to improve seismic resistance.	
T1-135	Seismic stability of the inner shell of the containment is discussed in the paragraph 3.2.2.	PGA limit value of the outer shell of the containment corresponds to other reinforced concrete structures and is equal to 0.61g.
	What is the seismic stability of outer shell?	
T1-136	In the section 3.2.2. it is stated: "...overpressure 0.39 MPa is accepted with the safety factor of 1.5".	For analysis of the inner shell of the containment, the factors were assigned in accordance with ACI Standard 359-13 (ASME BPVC III Rules for Construction of Nuclear Facility Components Div. 2 Code for Concrete Containments), CC-3230.
	Please explain in detail, why according to the requirements of ACI Standard 359-13 the overpressure 0.39 MPa is accepted with the safety factor of 1.5.	
	Please explain meaning of safety factor 1,5 and please provide justification of it.	
T1-137	In the section 3.2.2. it is stated: "The inner containment is designed in the form of a pre-stressed reinforced concrete structure. It is designed according to the requirements of the American regulations ACI Standard 359-13 "ASME BPVC – ASME Boiler and Pressure Vessel Code, Part III "Rules for Construction of Nuclear Facility Components", Division 2 "Code for Concrete Containments". This standard is the most comprehensive and well-developed international document in the field of containment design. The inner containment is designed also according to the Russian regulations PNAE G-10-007-89, Regulations for design of the reinforced concrete structures of localizing safety systems of nuclear plants (NP-010-98) and Rules to design and operate localizing safety systems of nuclear plants (NP-031-01). The inner containment is designed and installed according to the ASME regulations considered to be the tightest ones by combinations of loads and acceptance requirements".	The shell of the containment was designed with simultaneous application of ACI and Russian standards (PNAEG, NP). It fully complies with both ACI and Russian standards. Strength analysis was performed twice following different standards, and the worst option was adopted as the result. Full information is contained in report LN2P.D.110.1.0UJA&&.&&&&&.012.RF.0005.
	Is the inner containment fully compliant with the ACI Standard 359-13? Has an assessment of the compatibility of the standards (ACI Standard 359-13 and Russian regulations (PNAE G-10-007-89, NP-010-98 NP-031-01)) been performed? Please provide that information and the conclusions of such evaluation.	

T1-138	It is stated that the assessment of the subsequent Flooding of the NPP Site due to Earthquake Exceeding the DBA Level for the NPP is based on the calculations made in year 1972 by the Central Research Institute for Complex Use of Water Resources and the Institute of Hydrodynamics (Siberian department of the USSR Academy of Science, Novosibirsk). Please explain if this assessment has been updated considering the climate, surface and ground water level and soil regime changes in last 36 years? Please provide the full list of all potential flood scenarios caused by earthquakes, with the associated consequences.	The reassessment of the NPP site flooding possibility was not performed because it was not necessary: Absolute elevation of the NPP site is 179.3 m, maximum absolute water level in the Vileisk water reservoir with 0.01% probability (10-4) is 159.8 м. Flooding of the site with a break wave is impossible under any conditions because the site is located 19.5 m higher than the wave crest (at the initial moment of the break). Later on the wave only flattens out and the difference in height increases even more. This information is given in section 2.2 SAR In addition see the answer to T1-81.
T1-139	Temporary loss of water make-up source for the turbine equipment cooling system <...> does not affect safety of the NPP and is compensated by organizational and technical measures. 1. Please, explain how the loss of water in main source for cooling water (river Viliya) could not affect cooling system? Are there other effective water supply sources than from river Viliya? 2. What are these „organizational and technical measures"? It is not specified.	1) Water loss in the source for making up the circulating cooling water system does not lead to violation of the NPP safe operation limits. As the circulating water system (cooling towers) is used, loss of the make-up will not result in instantaneous loss of the ultimate heat sink. The NPP personnel will have time to take the required measures for shutdown of the unit. The ultimate heat sink for the systems of cooldown at emergency shutdown are spray pools. 2) organizational and technical measures are personnel actions for shutdown of the Unit in accordance with the instructions.
T1-140	In the section 3.2.4 "Possible Measures to Improve NPP Seismic Resistance" the organizational and technical measures are proposed to moderate the consequences of earthquakes exceeding the design values. The guaranteed confirmation, that these measures will be implemented before the start of operation of NPP shall be presented.	Measures to increase safety level shall be substantiated. At present the project has sufficient safety level in relation to the NPP site. This section has been developed on the basis of TKP 566-2015 "Assessment of the frequency of severe damage to the reactor core (for external source events of natural and man-made nature)." In addition, see the answer to T1-66, T1-131.
T1-141	Tables 4.1.1 and 4.1.2 show the probabilities of maximum and minimum water levels. Similarly, also the minimum water discharge shall be evaluated, whether it will be sufficient for cooling. The Lithuanian Hydrometeorological Service (LHMS) carries out measurements at Neris (Viliya) by Buivydžiai, which are about 36 km downstream the NPP. According to these data, 95% water discharge is 27.0 m^3/s, 97% - 26.5 m^3/s, 99% - 23.8 m^3/s, 99.9% - 20.2 m^3/s. The lowest recorded water discharge was 15.3 m^3/s, resulting from river ice jam upstream the measurement point. The minimum water discharge shall be evaluated.	Table 4.1.1 shows the estimated water levels corresponding to average annual water flows, table 4.1.2 - the estimated water levels corresponding to maximum water flows. Parameters of maximum and minimum design water discharge in the river of Viliya at the water abstraction point of the NPP unit pump station are specified in section 2.8 of SAR (pages 321-325).
T1-142	It is stated: "Maximum water levels are conditioned by the wave after the break of the Vileyka reservoir which is located higher, based on the calculations made in 1972 by the Central Research Institute for Complex Use of Water Resources and the Institute of Hydrodynamics (Siberian department of the USSR Academy of Science, Novosibirsk) [29] will not exceed the level elevation with 1% confidence as the break wave from the dam location to the supposed water intake point will mostly calm." 1. The justification is grounded on very old investigation performed in 1972. In accordance with international practices the safety issues concerning external events should be revised at least every 10 years. The updated investigation or confirmation, that input data, analysis methods and assumptions what were used are still valid are needed; 2. the reference [29] refer to "29. The Code of Administrative Offenses of the Republic of Belarus of April 21, 2003." Please clarify the reference and (or) provide the copy of document.	In regard to these calculations, Stress Test Report (target reassessment of safety) for Belarusian NPP contains a reference to the report dated 2013 ("Report. NPP in the Republic of Belarus. Hydraulic and mathematical simulation of the water intake structures of the NPP service water supply system". Central Research Institute for Complex Use of Water Resources. Minsk, 2013. Arch. No.BL-01423c/o). Reference to /31/ needs to be indicated in the National Report.
T1-143	In section 4.1.1 the Design-Basis Flooding of NPP is discussed. However the hypothetical event – failure all three water reservoirs of Viliya (Vileyka, Olkhovska and Snihiany reservoirs) at the same time (as a consequences of earthquake, for example) is not analyzed. The area flooded in case of all water reservoirs failure at the same time shall be analyzed and possible consequences to NPP shall be presented.	Possible water level rise in the river of Viliya in case of failure of the Vileisk water reservoir was analyzed. At a distance of 150 km from the dam to the water abstraction point of the unit pump station, the break wave flattens out and does not exceed the maximum design level considered. The results are specified in section 2.3 of SAR (page 319). The water volume of the Vileisk water reservoir is 260 mln.m3. The water volume in the Olkhovskoe water reservoir is 2.1 mln.m3, the water volume in the Snihiany water reservoir is 2.29 mln.m3, i.e. The total volume of both reservoirs is less than 2% of the water volume in the Vileisk water reservoir. Therefore, simultaneous failure of the Olkhovskoe and Snihiany reservoirs will have no impact on the water level in the Viliya river at the water abstraction point.
T1-144	No design basis flood threat foreseen in the design but conservatively applied consideration that the flood could locate to all NPP building below „0" level. This flood affects safety systems critical for heat transfer from RU and spent nuclear fuel. This means, that some safety systems would be flooded. Please, explain what the effect could be and how it affects NPP safety?	As shown in the stress test analysis, the conservative flooding scenario assumes failure of all the systems located below elevation 0.00. In this case NPP safety is ensured.

T1-145	In the section 4.2.2 "Potential Measures to Improve NPP Resistance to Flooding" the organizational and technical measures are proposed to improve tolerance of the NPP to floods. The guaranteed confirmation, that these measures will be implemented before the start of operation of NPP shall be presented.	It is said in the National Report that the inter-agency committee for coordinating and monitoring implementation of the plan of key organizational actions for the nuclear power plant construction in the Republic of Belarus, approved by Resolution of the Council of Ministers of the Republic of Belarus dated November 5, 2012, No. 1010, agreed to the Action Plan (road map) for establishing and implementing activities developed based on the results of the objective safety reassessment of the Belarusian NPP (Protocol No.03/37пр-дсп dated 27.09.2017) which are intended to improve the safety level of the Belarusian NPP. The above-mentioned Action Plan includes development and approval of the list of measures intended to improve the safety level of the Belarusian NPP and developed based on the results of the objective safety reassessment of the Belarusian NPP (in particular based on the results of the peer-review). See the answer to T1-66, T-131, T1-140.
T1-146	It is stated that, for example, strong winds have been analyzed on the basis of the 1961-2000 data. The question arises why no strong winds recorded until 2017 or 2015 were analyzed? The statistics of squalls are presented on the basis of the "Climate of Belarus" 1996 edition, which contains already obsolete information. Since disastrous meteorological phenomena are often local-scale events, the question arises whether the investigation of the recurrence of dangerous meteorological phenomena in the region of NPP did not allow to include information about the recorded hazardous meteorological phenomena in Eastern Lithuania? Overall, the Belorussian NPP is being constructed on the border with Lithuania.	"Recalculating of strong wind data on the basis of data dated up to 2017 is unreasonable due to ongoing tendencies on repeatability of strong winds, as well as within the period of 1961 - 2000". This also relates to a repeatability tendency on destructive squalls specified based on data of Reference book 1996. Not a single case of exceeded maximum wind gusts was recorded within the period of 2001 - 2017 in Lyntupy settlement and Oshmyany city as specified in the National Report. Spreading extent of hazardous weather events such as strong winds and squalls often occur locally. In this case hazardous weather events monitored on the territory of the Republic of Lithuania are not recorded by hydrometeorological service of Republic of Belarus, therefore they cannot affect safe operation of the Belarusian NPP. In view of the above we think that the data obtained from the Belgidromet network of weather stations for calculation of strong winds and squalls are sufficient for calculation.
T1-147	Drought is not analyzed as a dangerous meteorological phenomenon that can affect the operation of the NPP. But drought, for example, can affect the forest fires, which can be attributed to hazards. There is also no mention about hydrological drought and low water level in the river. The list of dangerous meteorological phenomena should be justified and the impact of the selected phenomena on NPP shall be evaluated.	List of dangerous meteorological phenomena is adopted in accordance with NP-064-05. The Viliya river does not dry up. Minimum water levels in the Viliya river are indicated in page 320 of PSAR, arch. No.BL-01065 пм, minimum water flows are indicated in pages 323-325.
T1-148	The section 5.1.1 does not mention among dangerous meteorological phenomena such dangerous meteorological phenomenon as lightning (thunderstorm); it remains unclear whether their recurrence has been assessed. It is also indicated that hazardous rainfall is when falls >50 mm in <12 hours. Why 12 hours period has been selected, since the amount of precipitation is later mentioned for 24 hours?	Meteorological phenomenon "lightning" is considered in i.2.3 of SAR, number of lightning strikes is specified as 3 per 1 km^2 per year. (page 220 of SAR) In section 2.3 of SAR, rain as a dangerous atmospheric phenomenon is considered. It is specified that the observed maximum is 101 mm/day, the design maximum is 10E-4 per year - 160 mm/day. (pages 194-195 of SAR). There are no contradictions. For thunderstorms, frequency of occurrence was assessed (see SAR). For lightning protection, the lightning protection system is provided at the NPP. Assessment of precipitation for 12 hours gives higher intensity, which allows to assess the consequences more conservatively. According to the List of hazardous natural hydro-meteorological phenomena by Belgidromet, precipitation of min.50 mm for max.12 hours is considered to be hazardous in the Republic of Belarus.
T1-149	Section 5.1.2 states that quote: "There are no possible sources of external fire and smoke in the two-kilometer zone of the NPP industrial site" possibly due to this statement consideration of external fire and smoke impact on NPP is completely omitted in the report. Nevertheless, it must be noted that the area around the NPP site is heavily forested. Although 2 km. distance of possible fire from the site might be considered as sufficient to omit a heat impact on NPP, smoke produced by significant forest fire with combination of unfavorable wind direction might have a significant impact on habitability of NPP site and it's compartments. It also must be noted that the document TCCP 566-2015 "Assessment of the frequency of severe damage to the reactor core (for external source events of natural and man-made nature)" referenced in the Report and presumably used in development of PSA also omits possible smoke impact in case of external fire. 1. The possible impact due to external fire and smoke from the woodlands shall be assessed. 2. How the smoke impact from external fire was considered in PSA and evaluation of combination of events? 3. What are the organisational measures and technical measures foreseen in the NPP design ensuring necessary habitability of NPP site and compartments in case of heavy smoke from an external fire?	According to i.8.3 NS-G-3.1 of IAEA Safety Guide and NP-064-05, the radius sufficient for considering the impact of forest fire hazardous factors (including those from smoking) on the NPP safety is 1-2 km. According to i.1, 3. For protection of MCR and ECR personnel from impacts of hazardous factors, including forest fire in close vicinity to the NPP, the life support system (LSS) is provided. In accordance with i.1.3.3.1 of LSS.0000.000TA, the life support system includes a filtration unit to remove flue gas (air with solid combustion products, particle size 0.3 μm) and toxic substances - min.90% from the supply air. If required, by the operator's decision, the MCR and ECR air conditioning/ventilation system can be switched over to full recirculation (self-contained operation mode). Thus, external forest fires have no impact on operators' activity in the MCR and ECR and, consequently, on the Unit safety as a whole. The other safety-related ventilation systems are not related to providing microclimate in the rooms constantly attended by the personnel and continue functioning in the normal operation mode. Necessity for continued operation of the normal operation ventilation systems of the other buildings (non-safety related) is determined in accordance with the internal regulations of the Belarusian NPP; the ventilation systems may be switched In this zone there are several insignificant forested areas. Distance from each of them to the NPP structures exceeds the required fire clearance, i.e. thermal impact may be neglected.
T1-150	In section 5.1.2 the selective analysis of possible combinations of initial external effects is presented in Table 5.1.2.1. It is not explained how these combinations are selected. It is not clear what methodology was used for selection of such combinations. It is not clear from the text if PSA-1 is already performed fully. 1. The methodology of selection of combinations of external effects shall be clearly presented. The justification, that selected combination will lead to the most dangerous consequences shall be provided. 2. Please clarify that is a current state of PSA-1? Please provide the main input data ant the results.	Combinations of external impacts were selected in accordance with NP-064-05. PSA-1 for external impacts has been completed. For the analysis of combinations of external impacts, the methods specified in documents "PiN AE-5.6 Norms and regulations in nuclear power industry", "SP 20.13330.2011 Loads and impacts" were applied. The main principles of the analysis of external impacts are based on IAEA SSG-3. Information on the methods applied for analysis of possible combinations of external impacts is given in he National Report in page 78, section 5.1.1: «Combinations of loads and impacts for buildings and structures are adopted in accordance with PiN AE-5.6, SP 20.13330.2011 "Loads and Impacts". A detailed analysis is carried out as described in PSA-1». Reference to SSG-3 is given in the Report in page 78, section 5.1.2. By the time of the Report development, development of PSA-1 for internal initiating events was completed, PSA-1 for external initiating events is under development as mentioned in section 2.3.4 of the Report. The detailed information on the completed works in regard to PSA-1 and the works in progress in regard to PSA-1 and PSA-2 is given in the answers to G-36 and T1-117.

T1-151	The Table 5.1.2.1 presents the impact on safety of different combinations of external events. The most events leads to the total loss of power supply and in this case the NPP safe mode is maintained due to Passive Heat Removal System through Steam Generators (SG PHRS) and Passive Heat Removal System for heat removal from containment (containment PHRS) operation. This demonstrated, that these very innovative Passive Heat Removal Systems, which removes heat from reactor through steam generator and from the containment, are extremely important for Belarusian NPP. However, the justification of reliability of these systems is not presented. It is just mentioned that some calculation (see page 83) showed the functioning of containment and SG passive heat removal systems at extremely low temperatures (up to – 61°C for more than 20 days). From the other hand it is mentioned about the possibility of condensation hydraulic shocks inside the SG PHRS tanks (see page 92). The detail description and justification of reliability and functionality of SG PHRS and containment PHRS systems at different initial loads, different (extreme) atmospheric conditions and level of the Safe Shutdown Earthquake (SSE) shall be presented.	PHRS reliability analysis is presented in Chapter 12 of SAR. Analysis of PHRS serviceability at extremely low temperatures is presented in Report "Assessment of SG PHRS and Containment PHRS Serviceability at Extremely Low Outdoor Air Temperatures". Report No.0-0-22-OT-002. Page 92 of the National Report says about impossibility of condensation hydraulic shocks inside the SG PHRS tanks due to installation of steam dump devices. 1. The performed calculation has shown functioning of the containment PHRS and SG PHRS at extremely low temperatures (down to – 60°C for more than 20 days). The design air temperature in the rooms where EHRT tanks and pipelines are located remained positive during the entire base period. Water temperature in EHRT decreased to a minimum of 0.9 °C [report VVER.0-0-22-OT-002]. One of the main mechanisms for maintaining positive temperature in EHRT at extremely low ambient temperatures is coolant circulation through the containment PHRS circuit. Water is heated in the heat-exchangers installed outside the NPP containment. 2. Functioning of the containment PHRS was substantiated by calculations (CFD) and experimentally at two large-scale stands: Containment PHRS in JSC "Afrikantov Experimental Design Bureau for Mechanical Engineering" (JSC "Afrikantov OKBM"), Nizhny Novgorod; • large-scale stand in FSUE "Alexandrov Research Institute of Technology" (NITI), Sosnoviy Bor, Leningrad region. A full-sized heat-exchanger with full-scale cooling loop was tested at the containment PHRS stand in OKBM. 36 various experimental modes were implemented. The test results showed that the adopted engineering solutions are substantiated
T1-152	Response spectra of two buildings are given under seismic, air-blast wave and aircraft crash impact. What air-blast wave parameters and aircraft characteristic (mass, speed) were used in calculations? The previously presented information demonstrated that the containment of Belarussian NPP can withstand a crash of light low-speed airplane only (less than 6 tons).	The Belarusian NPP design provides for protection from loads caused by 5.7t light aircraft crash at a speed of 100 m/s. The Belarusian NPP design considers impact of air blast ΔP=30kPa, air blast propagation is horizontal (as per NP-064-05).
T1-153	In section 5.2.1 it is stated: "The performed calculation showed the functioning of containment and SG passive heat removal systems at extremely low temperatures (up to – 61°C for more than 20 days)". In table Table 5.2.1.1, page 84 it is written that value of extreme effects for the baseline design and for the Belarusian NPP site concerning low temperature is the same -61°C. Does it mean that this is design basis temperature for all safety related equipment? If no, what is design basis low temperature applied for safety related equipment?	Table 5.2.1.1 shows that the design air temperature value is -61°C, the extreme value for the site conditions is -50°C. Safety-related equipment is designed for a design temperature of -61°C. No. ITR for designing the pumping equipment of PE system in pump station UQC contains requirements for ambient air temperature: from plus 15°C to plus 35°C. Table 5.2.1.1 shows that the design air temperature value is -61°C, the extreme value for the site conditions is -50°C. Safety-related equipment is designed for a design outdoor air temperature of -61°C.
T1-154	The potential impact of meteorological phenomena should be assessed not only by analyzing the values of meteorological elements recorded so far. In the face of global climate change, not only is air temperature rising, but also dangerous meteorological phenomena are becoming increasingly extreme and severe. Therefore, it is important to estimate the values of predicted meteorological elements by selecting climate change scenarios up to year 2100. In this case, the Representative Concentration Pathways (RCP) 8.5 climate change scenario should be selected. The 10,000 year probabilities (Table 5.2.1.1) should have been prepared in the light of the forecasts by the RCP 8.5 climate change scenario.	The design considers maximum values of atmospheric phenomena parameters for the period of once in 10000 years, and certainly they cover possible climate changes during the NPP service life.
T1-155	It is impossible to predict the reliable supply of diesel fuel through the pipelines (diesel fuel freezing). This supposes, that due to cold weather conditions diesel fuel for emergency diesel generators could freeze also. Please, explain what measures would be taken to ensure reliable diesel supply for emergency diesel generators.	The design minimum temperature is -61 ◦C. As far as standby diesel power station/unit diesel power station are concerned, the fuel system is located in the heated building and protected from the impact of low temperatures. As far as the intermediate diesel fuel storage is concerned, the equipment and pipelines are located underground, therefore low temperatures have no impact.
T2-1	Where is the substation Vilia located? The substations are rather vulnerable, much more than the Category I transformer.	110 kV Viliya substation is located in the area adjacent to the NPP, the cable line length is about 2.5 km. The emergency standby transformer is used in case of emergency blackout of 330 kV switchgear and loss of the main and standby transformers, i.e. loss of all internal standby power supplies.
T2-2	Pages 146-149 - Although there are no objections regarding conservative design, the plant level margin is not evaluated, the avoidance of cliff-edge effect is not demonstrated. Considerations on the electrical and I&C, and on the margin of active components lacking. It is questionable whether a seismic PSA (or any other external hazard PSA) can be performed since the hazard curves are missing.	For I&C equipment designed as per GOST 29075-91 "NUCLEAR INSTRUMENTATION SYSTEMS FOR NUCLEAR POWER STATIONS. GENERAL REQUIREMENTS) and GOST 30631-99 " General requirements for machines, instruments and other industrial products as to environment mechanical stability" the minimum vibration acceleration is 0.5g, which is much higher than the required seismic margin for the NPP. Electrical equipment is designed for peak horizontal acceleration 0.12g (DBE level adopted in the design basis). DBE level is set equal to 0.1g for the site. Values of peak horizontal accelerations (PGA) obtained as a result of field research during seismic microzoning were less than 0.1g (0.069g). Consequently, electrical equipment margin (in terms of seismic resistance) relative to the site DBE is 20%, relative to the site seismic conditions - over 70%. Development of seismic PSA - Quarter 2 of 2018.

T2-3	An IAEA SEED mission took place in January 2017. Was it a full scope SEED mission? Can you please provide a copy of the SEED mission report or at least the detailed conclusions and the list of recommendations.	NPP in relation to special external hazardous impacts: Item 64 calls the Republic of Belarus for further development of confidence-building measures, in particular by sending to IAEA suggestions on arranging a SEED mission to assess criteria and results of the studies for NPP site selection, as well as NPP construction and operation for the purpose of ensuring its complete safety; In the course of the mission, screening of external impacts, characterization of external impacts, both natural and man-caused, study of the construction site design parameters, site and environmental monitoring and consideration of the lessons learned from the Fukushima NPP accident were performed. According to the published IAEA press release, in their preliminary conclusions the SEED mission team noted that the NPP design parameters take into account external threats typical for the site such as earthquakes, floods and extreme weather conditions, as well as man-caused events. The team noted that the threat monitoring programs to be implemented throughout the NPP life cycle are adequate and properly documented. In addition, measures were taken to meet the challenges related to external events in view of the lessons learned from the Fukushima NPP accident. The mission noted the following good practices: • systematic and comprehensive screening of external threats typical for the site based on the well documented criteria; • obligation of the operating organization to submit to the regulatory authority in advance, prior to the NPP commercial operation, a comprehensive probabilistic safety analysis for both internal and external events as a part of the documentation Report by IAEA SEED mission for the Belarusian NPP is publicly available on Minenergo website (http://minenergo.gov.by/wp- 'ontent/uploads/Report-SEED-mission-Belarus.pdf).
T2-4	What is the power substation "Vilia"? A power plat or a part of it with an exclusive current line to the NPP site? Is thi sline to be applied for both units?	Viliya substation is not a generating power source, it serves for power distribution only. It is included into the Belarusian integrated power system and receives power supply from generating power sources of the Belarusian integrated power system. An individual cable line common for two Units is provided for the Belarusian NPP from Viliya substation. For auxiliary power supply of the Belarusian NPP Units during BDBA, i.e. in case of blackout and start failure of the diesel generators, 110/10 kV emergency standby auxiliary transformer with 16MVA power is provided. It is connected to Viliya NPP with 110 kV cable line laid in the ground and thereby protected from extreme external impacts. The 16 MVA emergency standby auxiliary transformer ensures operation of one channel of the safety systems, which is able to provide Unit safe shutdown and cooldown at each of the two Units.
T2-5	What does the sentence mean: *"If operating personnel decide to use an additional 110/10 kV power source to supply power to essential loads of the unit, the circuit is assembled manually."*	Power feed to the Unit section from the emergency standby transformer is not provided in automatic mode. Circuit breaker of the power feeder from the emergency standby auxiliary transformer is controlled remotely from the central control room (CCR) (performed by operative switch-over by the dispatcher in accordance with the diagram of the NPP backup auxiliary power supply from the standby transformers and the emergency standby auxiliary transformer shown in the drawing (see the answer to G-12).
T2-6	For the Belarusian NPP, loss of external power supply is a design basis condition analyzed in the SAR on the Belorussian NPP. The design provides for the following backup AC power supplies for each NPP Unit (constantly available for use). Could you please clarify this sentence? Are there backup AC power supplies subject to Technical Specs and/or some kind of surveillance requirements?	In this text, backup power supplies are emergency diesel generators (emergency power supply and normal operation power supply systems), as well as substation. Viliya.
T2-7	Can the emergency transformer provide energy to all four safety trains or is it stable wired to one of the four?	Design functioning of PHRS is sufficient for ensuring safety and efficient heat removal from the reactor core. If PHRS does not function/fails, one channel of the safety system is sufficient. The emergency transformer cannot supply power to 4 safety channels - there is no need for that. It can be connected to one (any) channel at each Unit in accordance with the diagram shown in the drawing (see the answer to G-12).
T2-8	Which customers will be supported by the unit DG?	AC power supply: - Coolant removal pump of system KBB; - Controlled leakage pump of system KTA; - Low capacity pumps of system KBA; - Part of pressurizer electric heaters; - High capacity make-up and boron control pump of system KBA; - Main steam isolation valve of system LBA; - Turbine rotor hydraulic jacking pump; - Make-up water pump of system LCU; - Pump of the turbine and generator lubrication system; - Turbine barring gear; - Water supply pump for automatic fire-fighting; - Shut-off and control valves of systems JEF, JEG, JNK, KAA, KAB, KBA, KBB, KBC1, KBC2, KBE, KPK, KPL1, KPL2, KRK, KTA, KUA, KUB, KWA, LAA, LBA, LBG, LBG30, LBJ, LCM, LCS, LCU, LDT, MAJ, MAL30; - Ventilation systems ensuring operation of the above process systems, electrical equipment and I&C equipment; - Lighting. DC power supply and UPS: - Emergency oil pump; - Unit I&C systems (normal operation); - Control current of switchgears; - Relay protection and electrical automation cabinets of the normal operation system; - Equipment of the communication system; - Equipment of fire protection and alarm I&C system; - Equipment of the automated radiation monitoring system; - Emergency and evacuation lighting.
T2-9	Please clarify the seismic category of elements (is this cable line laid in the ground the only element?) from the transformer up to the "Vilia" substation and down the transformer to the two units.	The emergency transformer is connected to Viliya substation with one 110 kV cable line laid in the ground. Two cable lines are laid in the ground from the transformer to 10 kV switchgear - separately for each Unit. Further all cable lines to 10 kV switchgear of Unit 1 are laid separately from cable lines of Unit 2.
T2-10	Are there design provisions allowing to supply an electrical power to affected unit from an operational (back-up) transformer of the other unit?	In accordance with the diagram of the NPP backup auxiliary power supply from the standby transformers and the emergency standby auxiliary transformer shown in the drawing (see the answer to G-12), an affected Unit can be powered from the standby transformers of the other Unit in non-automatic mode - by operative switch-overs from CCR.
T2-11	Could you please clarify if the EPSS for both units (i.e two EPSS) are the only loads for this transformer? Is it intended in any scenario to feed both units through this transformer?	The emergency transformer allows for connection of the emergency power supply system only - one channel of the emergency power supply system of each Unit can be simultaneously connected in accordance with the diagram of the NPP backup auxiliary power supply from the standby transformers and the emergency standby auxiliary transformer shown in the drawing (see the answer to G-12).
T2-12	If operating personnel decide to use an additional 110/10 kV power source to supply power to essential loads of the unit, the circuit is assembled manually. The 10 kV section (including EPSS) was selected in accordance with the NPP emergency response manual; Could you please clarify if this option is already considered in some procedure/SAMG etc of the plant along with the procedure to assemble it manually? Is it trained?	Manual assembly of the circuit means only to switch on 10 kV circuit breakers of sections 01BCC, 02BCA from standby section 00BCJ, which is under voltage in the normal mode. Switch-overs of this type are performed in accordance with the technical operation regulations, safety regulations and local instructions. Personnel permitted to work without supervision must pass theoretical training, training course, practical training, theoretical basic course on NPP with VVER, special course, probation course, probation in service, examination, backing-up, emergency response training. The option of power supply from the emergency standby auxiliary transformer is considered in document "Procedure of actions by the Belarusian NPP personnel in case of NPP auxiliaries blackout BLR.1.E.534.&.&&&&&&.&&&&&.022.YP.0001" and will be implemented in accordance with the switch-over form.

T2-13	Please could you clarify the difference between the "Unit DG with a power of 6300 kW" and the other "4 EPSS DGs with a power of 6300 kW each".	Unit DG is an internal power source of the reliable power supply system for normal operation auxiliary consumers important for safety and integrity of the main equipment. List of consumers - see the answer to T2-8. Each of the four EPSS DGs feeds all the load of the respective channel of the safety channel required from the point of safety in case of loss of external power supply to bring the Unit to the safe shutdown condition and maintain it in the safe condition.
T2-14	Are the EPSS DGs allocated to a respective safety train and, which systems count to the respective safety train feeded by EPSS?	Yes, the EPSS DGs are distributed among the respective safety channels and numbered depending on what channel is powered by EPSS.
T2-15	Which aircarft is the refernce for an airplane crash to be considered in teh hazard calculations for the electrical power supply buildings?	This question is beyond the scope of the stress-tests. The design of the Belarusian NPP provides for protection against 5.7t light aircraft crash at a speed of 100 m/s, which complies with NP-064-05. See the answer to G-41.
T2-16	Does "natural impacts" also includ "man-caused impacts"?	Yes. It is meant that when designing and selecting equipment and building structures, the site conditions and natural and human-induced impacts were considered. In the sentence "The equipment reliability analysis shows that the above-mentioned natural impacts do not lead to accidents" the word "man-caused" was omitted by mistake.
T2-17	Which earthquake category does the diesel fuel warehouse UEJ have?	With regard to the standby diesel power station, the intermediate diesel fuel warehouse refers to seismic category I as per NP-031-01. With regard to the unit diesel power station, the intermediate diesel fuel warehouse refers to seismic category II as per NP-031-01.
T2-18	How the alarms for levels in the supply tank and the storage tank of the EPPSS DG are supported? By the UPS (channel 7)?	Alarms for levels in the supply tank and the storage tank of the EPSS DGs are powered from the EPSS DG control cabinets. EPSS DG control cabinets have two power feeders (with automatic load transfer) from 0.4 kV EPSS sections, and also include 2 batteries designed for 30 minutes operation to provide continuous data transfer.
T2-19	Please clarify the 72 hours operational time of the EPSS DG`s initially defined versus the 53-hours operation defined later in the next paragraph.	Standby diesel power station: For each NPP Unit the diesel fuel stock is located: - in the main warehouse; - in the intermediate warehouse and in the supply tank of DG set of each channel (irreducible fuel stock). The fuel stock in the main warehouse is provided in the amount not less than required for operation of DG set of one channel for each NPP Unit at nominal load for at least 120 hours (5 days). Volume of the irreducible fuel stock stored in the intermediate warehouse and in the supply tank of each DG set is sufficient for operation of DG set of each channel at nominal load for at least 48 hours (2 days). If it becomes impossible to restore the power supply of NPP auxiliaries within two days (and return the DG sets to the standby) in the NPP blackout mode, the design provides for replenishing the main and intermediate warehouses with diesel fuel of the required quality from the regional oil product depots by means of oil tanker trucks. The fuel stock in the supply tank of each DG set is provided for DG set operation at nominal load for at least 5 hours. Unit diesel power station: The fuel stock in the intermediate warehouse is provided for DG set operation at nominal load for at least 24 hours. 72 hours are mentioned due to the requirement for NPP independent operation in case of loss of external power supply. According to the Russian standards, the fuel stock in the DG set structure is designed for 53 hours of DG operation (sup
T2-20	Could you please indicate if the control panel is the one in the Diesels? If the answer is "yes", is it defined in the shifts the task to look after these values?	The DG control panel is meant. According to the operation manual "Diesel generator sets of the emergency power supply system and normal operation reliable power supply system (XJ10, 20,30,40,50)", during the shift inspection the operating personnel of the electrical department monitor the indicated parameters. Values of the monitored parameters are displayed at the local control panel and duplicated in the MCR.
T2-21	What is LCU tank and SG PHRS tank minimum allowed water volume/level during power operation and during refueling (according to operational limits and conditions)	Minimum level for LCU tanks 1. during operation: - LCU01, 04BB001 – 700 m3 in each tank. - LCU02, 03BB001 – total irreducible stock 700 m3. 2. Not rated during refueling. In particular, complete emptying of the tanks is permitted for performing any works or replacing water in the tanks. During normal operation, SG PHRS is in the standby mode. The level in the EHRT is 5.8 m.
T2-22	The fuel storage tank capacities are 100, 8, 50 and 8 m3. Are these minimum allowed capacities according to the operational conditions and limits?	The tank capacities are determined according to RD EO 0052-00 rev.2. 2.
T2-23	The fuel tanks are refilled from tank trucks. In chapter 5.1.2 it is stated that there are no possible sources of external fire and smoke in the two-kilometer zone of the NPP. So fuel truck are not considered as possible source of fire and smoke? Is there always possibility to deliver diesel fuel by road?	According to the design, diesel fuel to refill the diesel fuel tanks is delivered by road only. There is a developed transportation system in the area of the Belarusian NPP, which provides many possible access ways to the NPP site. Additionally, there is a railway track for transportation of goods and people from Oshmyany railway station to the NPP site. From the site, two exits to public roads are provided for personnel access, material and technical support. The network of on-site roads allows for various access ways to buildings and facilities of the site. On-site roads are looped to organize unimpeded and free movement of vehicles and personnel. According to the performed assessments of extreme natural and man-caused impacts on the access ways, their partial damage is possible with preservation of their functions. At seismic impacts above DBE level, there is a probability of damage to the engineering structures on access ways. They are restored by road service. If it is impossible to restore the engineering structures, arranging of personnel and support access over obstacles is provided by alternative ways.
T2-24	How how the additional fuel stored in the diesel warehouse will be transported to the storage tanks of the respective EPSS DG? By stationary pumps? How these pumps will be supported with energy ?	Fuel is transported to the fuel tanks of the standby diesel power station/unit diesel power station from the common-plant diesel fuel storage warehouse by tanker trucks.
T2-25	Could you please explain how the tank in 00UEJ is connected to each DG? Pipelines in four different trains physically separated? Are there valves to isolate each train from the others? Are these valves manual ones or are they electrically feeded from some source (please specify the latter)?	Fuel is transported to the fuel tanks of the standby diesel power station/unit diesel power station from the common-plant diesel fuel storage warehouse by tanker trucks.
T2-26	A flow rate of 204 g/kWh for one DG: could you please clarify: does this value come from SAT testing of the diesels or is it from the supplier?	Specific flow rate of diesel fuel at nominal power for one DG set 207 g/kW x h is adopted based on the technical documentation (specification on DG set) of the potential DG set supplier.

T2-27	Could you please clarify if the regional oil supply is specifically stored for the plant? Is the road considered for transportation available in any type of accident scenario (weather conditions as heavy snow, black ice, flooding, earthquake)?	According to Decree by the Council of Ministers of the Republic of Belarus No.1800 dated 20.11.1998 "On establishing of the republican system of material reserves for emergency response", material reserves of various levels are established, including facility, local, regional, industry and state reserves. Based on the respective documents of entitlement, material reserves of various levels are established, with defined procedure of their establishing, accumulation, storage and use. The Belarusian NPP State Enterprise establishes facility material reserves, including diesel fuel stock intended for the NPP only. Access ways (main and standby) intended for diesel fuel transportation will be maintained in satisfactory condition in case of unfavourable weather conditions by road (utility) services responsible for the respective roads.
T2-28	Could you please clarify what are the loads to be feeded from the UPS during the first two hours?	During the first two hours, power supply shall be provided at least to: pressurizer POSV, BRU-A, MSIV, emergency gas removal system, isolating valves of the containment. The main consumers are: - Cut-off valves; - Some valves of the safety systems; - Valves of fuel pool cooling system FAK; - Valves of component cooling system for essential loads KAA; - Valves of main feedwater piping system LAB; - Valves of steam generator blowdown system LCQ; - Valves of live steam piping system LBA; - Equipment of I&C systems; - Control current of switchgears; - Equipment of fire protection and alarm I&C system; - Equipment of automated radiation monitoring system; - Emergency and evacuation lighting.
T2-29	On page 87 an operational determined leckage of 2.15 m³/h in the primary circuit is mentioned. What are the sources and how much water will be releaased from the respective source?	The analysis takes into account both controlled and uncontrolled primary circuit leakages in the amount of 2.15 m3/h, which corresponds to the maximum possible leakage rate during RP operation at the rated parameters. The specified volume of leakages includes: - leakages through RCPU seals - 4•0,05 m3/h; - leakages through pressurizer POSV - 0,35 m3/h; - sampling - 3•0,5 m3/h; - uncontrolled leakages in the amount of 0,1 m3/h. When the pressure decreases, leakage rate of the primary circuit also decreases (document /1/, Fig. 5.2). Mass yield of the primary circuit coolant through leakages after 72 hours is approximately 41 tons. Boric solution stored in the ECCS hydro accumulators recovers loss of the primary circuit coolant.
T2-30	The SG PHRS is a new safety feature not implemented in a reactor technology before. So, the function is not so well know. Since, the SG PHRS is intended to be the main system in a BDBA case, a detailed description of structure and components (EHRT, LCU, JNB50, respective DG`s) as well as of function, limits and values is still required for the understanding of the system. Beside the needed genarl information , following special questions arose after the first check: How the SG PHRS will be activated (explanation of procedure)? Please could you give further information about the alignments of the SG PHRS system and a description of the function/position of each valve in normal operation/outage and during the different phases of the accident? What is the function of the isolation valves? How do you assure all the valves are opened in the accident and there is no failure of any of them? Could you clarify the meaning of "A solenoid valve is used as a small startup valve". What is the purpose of the solenoid valve? Does it open any of the startup, control and isolating valves? Could you please clarify the concept of "A motor-operated valve is used as a big valve"? Regarding the sentence "mechanical passive opening of the sta	A more detailed information is given in SAR, Chapter 12.
T2-31	*Please confirm that the SG PHRS tanks are designed both for primary system cooling and containment cooling. Could you give further detail about "Heat is removed into the atmosphere by evaporation of water from the SG PHRS tanks"? Where is the atmosphere where the evaporate goes? How is it followed SG PHRS function during the accident? Could you please clarify what is the meaning of "the SG 1-4 PHRS are activated" and "full design capacity"? Regarding the time "and within 80 sec", does it mean 80 sec after time zero in the accident? What is the level in the core when the SG 1-4 PHRS reach the full design capacity? Could you please clarify if "reaching the design parameters" is the same concept mention above "the full design capacity" and this is reached in 80 sec? Could you please explain what the meaning of "pulsating" is? How is this function performed? Is it performed opening and closing valves? What are the valves involved and how are they electrically feeded? How this "pulsating" mode is controlled from the MCR? Is PHRS tanks make-up pipeline routed via annular space? Please clarify why it is not considered any additional failure of one train of the PHRS. Could you please clarify if*	"SG 1-4 PHRS start operation" means that after formation of the signal for PHRS actuation (following failure to start of DG) with a delay of 30 s the PHRS starts reaching the design power. The start-up period during which the PHRS capacity changes from the nominal to maximum design value under the current pressure in the steam generator (Table 1) is 80 s. At the moment when the SG 1-4 PHRS reaches the full design capacity the reactor is completely filled. Change of the coolant level in the reactor chambers is given in document 491-Пр-1977 /1/, Fig. 5.7. The "pulse" supply of boric solution from the ECCS HA is caused by change of pressure both in the primary circuit (pressure decrease due to coolant leakage) and in the ECCS HA (pressure decrease due to their draining). The ECCS HA is a passive system. Boric solution is supplied from the ECCS HA due to opening of the check valves installed on the pipelines connecting the ECCS HA with the primary circuit. For actuation of the check valve external sources of power supply are not required. The valve opens when the relevant pressure drop is reached. Under pressure balance in the ECCS HA and in the primary circuit the check valves close. The specified processes cause the "pulse" supply of boric solution from the ECCS HA. The BDBA analysis is performed with no regard to additional failures. Only those failures are taken into consideration which are consequences of an initial event. For example, due to failure of all di The expert assessment performed proceeding from the available amount of boric solution in the ECCS tanks and rate of its consumption in three days from the beginning of the accident (see document /1/, Fig. 5.1.2.7) and rate of the core res 1. Water from the emergency heat removal tanks (EHRT) is supplied for cooling the heat exchangers of the PHRS containment. In addition, the SG PHRS heat exchangers are submerged in the EHRT water. The EHRT water is heated in the heat exchangers of the PHRS containment and also due to heat transmission from the SG PHRS heat exchangers . When the water is heated it starts boiling and the steam is released into atmosphere via air 2. The core level by the moment when the SG PHRS reaches the full design capacity depends on accident scenario. 3. The 80 sec period (from the moment of valve opening) is a period for reaching the nominal power under small valves opening. This period is required for medium heating in the SG PHRS, for reaching nominal flow rate and steam condensa 4. Justification of adequacy of the capacity for 24/72 hours is performed according to the heat balance equation taking into account a curve of residual energy curve (can be provided). 5. The EHRT level is monitored by the I&C system independently in all modes of operation. 6. Protection against freezing. The performed calculation has shown functioning of containment and SG passive heat removal systems at extremely low temperatures (up to – 60°C for more than 20 days). The design air temperature in the ro 7. Repair of SG PHRS during power operation is not provided. For this reason single failure during repair works is not considered.

T2-32	Regarding the sentence “lack of supply from the ECCS hydro accumulators” could you please clarify how and when do accumulators stop? Could it be a manual action from MCR, or is it an automatic action related with level in accumulators? If accumulators are not isolated what are the consequences that can be produced (N2 ingress in the core?)?	It was accepted during calculations that as a result of the EHRT dehydration, after three days, PHRS operation stops, which leads to increase of the parameters in the secondary circuit up to the setpoints for opening of the pilot-operated safety valves (POSV) of all the SGs. Deterioration of heat removal by the secondary circuit (due to level decrease in the SGs) causes an increase in the primary circuit parameters up to the setpoints for actuation of the pressurizer control POSV. For this reason pressure in the primary circuit exceeds pressure in the ECCS hydro accumulators and, respectively, boric solution is not supplied to the primary circuit. The accumulators are securely disconnected from the reactor by closing of two gate valves arranged in-line. The gate valves are closed automatically following level decrease in the ECCS HA down to 1250 mm from the HA bottom. The accumulators are disconnected to avoid nitrogen ingress into the reactor.
T2-33	Regarding the sentence “start of heating is about 310 000 sec (86 hours)” it is said before that “the PHRS ensures the removal of residual heat within three days”. 86 hours are 3 days and 14 hours to start heating, whilst PHRS ensures cooling for 3 days. Could you please clarify what the 14 additional hours come from?	The PHRS stops in 72 hours after EHRT draining. Further in 14 hours the following occurs: - pressure increase in the secondary circuit up to the setpoints for opening of the pilot-operated safety valves (POSV) of all the SGs leading to draining of all SGs; - - increase of the primary circuit parameters and, as a result, increase of fuel and FR cladding temperature due to deterioration of heat removal by the secondary circuit ;- pressure of the primary circuit reaches the setpoint for opening of the pressurizer POSV;- maximum temperature of fuel element cladding reaches the value close to the value under RP operation at nominal parameters (about 350 ºC). Thus, in 86 hours from the beginning of the accident heating of the FR cladding is started taking into account termination of the PHRS operation in 72 hours from the beginning of the accident.
T2-34	How is “start of heating” related to “dehydration of the FA upper part”? How much time is between both processes?	The beginning of heating is characterized by increase of FR cladding temperature. Dehydration of the FA upper part is characterized by level decrease in the reactor core below the FA heads (coolant level at the level of FA fuel portion). These events demonstrate the tendency of accident transition to a severe stage. Time characteristics of these events are rather close, but can differ depending on the accident mode. For the accident described in 6.2.1 time characteristics are similar to those described in section 6.1.2 (NPP blackout). Specific times of these events are given in the Report on the stress-tests /31/.
T2-35	Could you please clarify if there is any instrumentation available during the accident to identify any of the phenomena mentioned, "start of heating" or "dehydration"?	These processes occur at the BDBA stage (prior to transition of an accident to a severe stage). Loss of coolant in the reactor core is controlled by an emergency level meter. The beginning of heating - increase of coolant temperature - is controlled by the sensors at the core outlet (neutron flux, temperature and level measuring channel).
T2-36	It is said previously in the report that the primary leaks are 2,15 m3/h (see page 87, paragraph 12). How this leak correlates to 41 tons? Could you please clarify if the accumulators capacity includes the leak	The analysis takes into account both controlled and uncontrolled primary circuit leakages in the amount of 2.15 m3/h, which corresponds to the maximum possible leakage rate during RP operation at the rated parameters. The specified volume of leakages includes: - leakages through RCPU seals - 4•0,05 m3/h; - leakages through pressurizer POSV - 0,35 m3/h; - sampling - 3•0,5 m3/h; - uncontrolled leakages in the amount of 0,1 m3/h. When the pressure decreases, leakage rate of the primary circuit also decreases (document /1/, Fig. 5.2). Mass yield of the primary circuit coolant through leakages after 72 hours is approximately 41 tons. Boric solution stored in the ECCS hydro accumulators recovers loss of the primary circuit coolant.
T2-37	Conclusion on Sufficiency of NPP Protection from Loss of Power Supply: According to conclusions “To prevent fuel damage in the spent fuel pool in case of an accident involving the loss of all AC sources at the NPP under conditions of the complete core unloading, it is necessary to supply water to the spent fuel pool at a flow rate of min. 7 kg/s within not more than 41 hours [31]”. From the information given it is not evident that this can be done.	As shown in /31/ capacity of pump JNB50 is 60t/h. Thus, the required makeup of the spent fuel pool is provided.
T2-38	According to the page 88 operation of four PHRS channels decreases pressure in the steam generators in accordance with the PHRS performance parameters….Thus, the PHRS ensures the removal of residual heat within three days. It seems that it is assumed that all four channels are in operation (system is 4*33,3%).	Justification of the events considered for stress tests is performed using the deterministic approach and actual scenarios of these events. Thus, all four PHRS channels can be put in operation.
T2-39	According to the page 88 “The presence of boiler feed water in the SG in case of loss of external power supply and design backup AC power supplies is substantiated by the operating organization in [31].” This is not fully understandable without the reference document.	The document will be submitted to PRT for review within the period from 12.03.2018 to 16.03.2018.
T2-40	Could you clarify is there any possibility to measure level and temperature in the core during the accident?	Under emergency conditions the level and temperature in the reactor core are monitored by the relevant system.
T2-41	It is assumed that in 24 hours from the start of the accident Unit becomes uncontrollable because the reliable power supply batteries are discharged. Could you clarify if it is “unit” or “units”? Could you clarify the meaning of “uncontrollable”?	The phrase is incorrect. In 24 hours the mobile DG is put in operation and power supply of the I&C system is not interrupted.
T2-42	Previously in the report it was said that the PHRS stops after 72 hours. Could you please clarify the possibility of fuel melting and the period of time of 3 days versus 3.5 days”. (The 72 hours time is also considered later in this page when it says: “The analysis results show that in the considered time interval (72 hours)”).	See response to comment T2-33.
T2-43	In previous paragraphs it was defined: “it is required to take measures [PHRS] within not more than 3 days”, and hydro-accumulators finished their inventory at time = 259850s. Why is heating may vary from 13 to 15 days?	See response to comment T2-31.
T2-44	Could you please clarify the meaning of “design means”. Are they defined already for the plant and included in procedures for accident management?	*The comment refers to section 6.1.2 (sh. 89-90). It is stated in the text that in 13-15 days it will be required to restore supply of boric solution using the routine means. The design means are the systems supplying boron to the primary circuit from the safety systems (LP ECCS, HP ECCS and EBIS). The relevant actions are provided in the BDBA Management Guidelines.*

T2-45	Could you clarify why other modes of operation/options are not considered? 3/4 loop inventory during outage, vessel head rising-full core loaded during outage?	Minimum allowable coolant level in the reactor is provided during cooldown for repair works under the unsealed reactor condition and is equal to 600 ± 50 mm over the axis of the reactor "cold" branch pipes. BDBA with loss of heat removal under this state is described in PSAR, section 15.6.1.7 "Long-term (up to 24 h) failure of heat removal by the planned and emergency cooldown systems under uncovered and/or unsealed reactor". The analysis results showed that the time to fuel uncovering under specific accident is minimum 2.4 h from the beginning of the accident. Within this time period to avoid to prevent transition of the accident to a severe stage the personnel must provide boric solution supply to the reactor with minimum rate 10,45 kg/s. In this case water is supplied from the ECCS hydro accumulators postponing the moment of FA heating and coolant loss in the reactor core. Further on it is required to restore power supply of the safety systems removing heat from the reactor core under shutdown. Along with this it is possible to use systems of the neighbouring Unit (See response to comment T3-31).
T2-46	Comment: the worst case scenario is 41 hours for personnel emergency response assuming boiling-off to the FA heads.	Under complete unloading of the spent fuel assemblies for 10 years of operation in the SF pool and under power loss in 41 hours uncovering of the FA heads occurs.
T2-47	Could you clarify if it is one mobile DG set that performs both tasks, restoration of power supply and to ensure water supply, or are there two mobile DG sets?	Recovery of power supply means solution of a wide range of tasks: recovery of external power supply (interaction with a power network operator), recovery of operation ability of, at least, one EPSS diesel generator, preparation of the mobile DG for operation. If it is not possible to recover power supply in the first two aspects within 24 hours the mobile DG (so-called 7-channel DG) must be put in operation supplying power to the consumers (see response to comment T2-51) including those which provide water supply.
T2-48	According to page 90 option 1 "The total time of the spent fuel pool boiling-off to the FA heads from the beginning of the accident will be at least 41 hours" And it is stated that characteristics of the technical means for makeup of the spent fuel pool were selected taking into account the prevention of heavy fuel damage in the spent fuel pool. Statement needs clarification (more explanation)	As shown in /31/ capacity of pump JNB50 is 60t/h. Thus, the required makeup of the spent fuel pool is provided.
T2-49	water level in the fuel pool – 8.7 m (level at fuel storage, Is this value in line with minimum acceptable value in operational conditions and limits?	8,7 m is a nominal level in the fuel pool. Operational limits are given in SAR, Chapter 16.
T2-50	Is assumed that water from the four emergency heat removal tanks is used. This assumption should be justified.	See response to comment T2-38.
T2-51	Which customers will be supported by the "channel 7" of the BDBA equipment? The mobile DG (500 kW) supporting the BDBA channel 7 will be connected in a cabinet outside the UJE building. How it s ensured, that the connection is available in case of external hazards? Could you please clarify what is the worst case scenario considered to transport the mobile DG sets from outdoors to the specific connection point (i.e. heavy snowfall?). Is the cabinet bunkered? How the mobile DG for supporting the channel 7 will be refilled? So the mobile DG set is for: low power high pressure pump JNB50AP001, recharging the batteries, and further operation of the system. Could you please clarify how many loads are fed from the mobile DG set? Could you explained the alignments from the pump to both the PHRS tanks and spent fuel pool? The worst case scenario would it be to lose one train of SG PHRS and perform all the activities in 24 hours?	The main consumers are: - Making-up pump of the emergency heat removal tanks and spent fuel pool ; - Alkali emergency injection pump; - Valves of the emergency gas removal system KTP; - Valves of the system of emergency water use from the reactor internals inspection shafts JNB and JNB90; - Valves of the pressurizing and steam discharge system JEF, JEG; - Valves of the fuel pool cooling system FAK; - Valves of the make-up water system LCU; - Equipment of the hydrogen concentration monitoring system JMU; - BDBA recorders CR; - Equipment of the BDBA I&C system; - Equipment of the automated radiation monitoring system; - Equipment of the ventilation systems; - Equipment of the communication system; - Lighting system Procedure for connection of the mobile DG set will be described in BDBAMG (SAMG). The location of the mobile DG set is considered to be fixed, and there no need to transport the DG set to the connection point. The terminal cabinet is described in the National report.
T2-52	The power calculation for the mobile DG (XKA70) takes into account a current equal to the current of a 10-hour battery discharge (203 A). Is the charging the batteries and supporting the consumers of the channel 7 simultaniously possible? Could you please clarify that the batteries run out after 24 hours? Is it intended to make their life longer depending on the loads needed? How much time is needed. Except for battery recharging an dopereting the punps refilling SG PHRS and SFP, what are the other consumers that need and could be powered by the Mobile DG, according to the design? Is there a procedure to perform the connection of the mobile DG set? Has it been trained (considering among other factors) the time to get the key? Are the mobile DG sets subject to some kind of surveillance requirements/preventive maintenance? What are the provisions for storing at the site of diesel fuel for the Mobile DG, and how long it would supply the needed consumers in case of total blackout? Once the mobile DG set is in place, could you please clarify if there is one person dedicated to the diesel full time? In case of accident in both units, two people will be in place to follow DG performance in the local control panel?	Power of the diesel generator is enough for batteries charging and power supply. Batteries are not fully discharged. There is a certain margin. The 24-hour period is connected with the need to actuate the pumps which can be powered only by diesel generator. Inspection / maintenance requirements are described in the Operational manual.
T2-53	Could you please clarify if problems with ventilation systems have possible consequences already considered in guidelines/procedures etc? Do they jeopardize habitability of the MCR? Could you please indicate what are the cases where 8 or 12 people are in the MCR? Could you please clarify if any I&C could be affected by high temperatures in the MCR?	Consequences of the ventilation failure are described in section 5.1.3/31/. According to Fig. 5.1.3.5 the temperature displayed at the MCP reaches 43 C within 72 hours. Conservatively, the number of personnel in the MCR involved in the analysis is 8 persons. According to the design layout the number of personnel in the MCR is 5 persons. The allowable parameters of the APCS equipment are given in section 5.1.3 / 31/

T2-54	Does the design provide for seismic qualification of category I (SSE) of all systems and components that implement safety functions "Residual heat removal from the reactor core and spent fuel pool" and "Heat removal from the containment" during BDBA (so called "technical means for BDBA management")? What is the seismic qualification of te make-up system for the SG PHRS water tanks and spent fuel pool?	Systems: - residual heat removal from the core- residual heat removal from the spent fuel pool- heat removal from the containment- make-up of the SG PHRS water storage tanks and the spent fuel pool refer to seismic category 1.
T2-55	Among the measures to improve NPP stability in case of power supply loss, the following is defined in the report: "in terms of relevant operational documentation - develop additional sections on the actions of personnel in the event of an accident with complete loss of AC power supply of the NPP with regard to: -strengthening the monitoring of the Unit process parameters; -strengthening the monitoring of the safety-related systems operation;" Can you please elaborate more regarding the meaning of the phrases "strengthening the monitoring of the Unit process parameters" and "strengthening the monitoring of the safety-related systems operation;" and how it is intended to achieve these two tasks?	These requirements are specified in the current emergency documentation in the event-oriented format. Under de-energizing of the auxiliary systems the personnel must enhance monitoring of the process parameters characterizing state of the critical safety function and integrity of physical barriers on the way of radioactive substances emission to the environment. Monitoring of the parameters of the systems which continue their operation must be also enhanced to provide their long-term reliable operation. These measures are provided by the emergency documentation and include the following: increase of walkdown frequency, recording the operation parameters of the safety-related equipment, graphical representation of the recorded parameters for timely detection of negative tendencies and for taking preventive measures.
T2-56	It is reported that: "The main directions of the personnel actions in case of complete loss of the design ultimate heat sinks are as follows:- putting the SG PHRS into operation, monitoring the operation of the system;" Can you please describe in brief what are the personnel actions needed to put SG PHRS into operation and which are the parameters monitored for the system operation?	Under loss of the ultimate heat sink (spray pools, cooling towers) the SG PHRS is put in operation by an operator from the MCR panels. During operation of the SG PHRS the EHRT level and RP parameters are monitored.
T2-57	Which procedure has been developed and has to be applied, if a SBO (loss of external power supply, regular redundant AC power supply and various stationary AC backup power supply) occurs shortly after the start of the reactor refulling (open primiary circuit)?	Scenario with the minimum allowable level in the EHRT is described in SAR, section 15.6.1.7 "Long-term (up to 24 h) failure of heat removal by the planned and emergency cooldown systems under uncovered and/or unsealed reactor". Procedures for this scenario management are given in the BDBA Management Guidelines. The personnel actions are described in section 5.2.2 / 31/.
T2-58	Could you please clarify what are the parameters currently considered as part as the accident management and which ones are going to be implemented?	Section 6.1.5 of the National report contains the recommended measures to improve the NPP stability in the SBO mode. The recommended measures to increase safety level at the Belarusian NPP after targeted reassessment of safety (stress tests) and the measures recommended following the results of the National Report analysis will be implemented stage-by-stage according to the Safety Enhancement Program of the Belarusian NPP. To include these measures in the Program their influence on the NPP safety will be analysed and, depending on the analysis results, priority of their implementation will be determined. The terms of implementation of the measures will be specified by the Safety Enhancement Program of the Belarusian NPP
T2-59	LCU tanks are located in two buildings, UMA and UJE. Could you please clarify the design to arrange the making-up from one building or the other one both to the reactor plant and to the spent fuel pool? What are the alignments and valves to be positioned (which are manual or electrically driven?)?	The make-up system for the EHR tanks and spent fuel pool is described in SAR, section 12. Principle diagram of the make-up system for the emergency heat removal tanks and spent fuel pool is given in stress-tests /31/, section 1.3 (Figure 1.3).
T2-60	In terms of removal of residual heat from the spent fuel pool, it is described in more detail than the part for the removal of residual heat from the reactor plant. Could it be possible to explain in further detail de one related to heat removal from the reactor plant?	Chapter 6.1.5 must contain the following text: "with regard to the heat removal from the reactor plant and spent fuel pool: - to arrange for making-up of LCU tanks from the on-site and off-site sources of water after 72 hours; - to arrange for making-up of the spent fuel pool after 41 hour. This measure can be implemented by connecting non-routine facilities (fire engine with a pump unit having a capacity of 40 liters/s and a head of 100 m) to two process connectors of JNB50 system located on the outside of building UJE (at elevations +0.690 and +0.730 the water is taken from LCU tanks through the pump unit of the fire engine and further through the pipelines of system JNB50 the water is supplied to the spent fuel pool) having flanges with plugs installed on them; - to modify the process flow diagram of the JNB50 system by adding tie-in of a check valve bypass to the make-up line for the emergency heat removal tanks. This solution will allow the operating personnel to make up the spent fuel pool after 41 hours.
T2-61	Regarding "by connecting non-standard facilities (a fire engine with a pump unit …" Could you please clarify the non-standard facilities used? Where are these non-standard facilities electrically connected? Are they seismic? Is it a single pump? How is the single failure considered? Is already decided who would be in charge of running the fire engine with a pump?	Connection of the non-routine facilities (fire engine with a pump unit) is performed via the connector (Bogdanova). There are no electrical connections. The fire engine are not classified by seismic category according to Russian regulatory documentation. The number of pumps are determined according to the number of the used fire engines. Single failure at the DiD 4 level is not considered.
T2-62		
T2-63	Please clarify: is PE system designed for an additional failure + maintenance? Where is the cooling water temperature from 4 to 28°C measured? In which case is 4ºC obtained?	The cooling water system for essential loads (PE) operates in all operating modes of the Unit (including blackout). Design calculation BL-02691s/o "Justification of cooling capacity of the spray pool" executed by "Vedeneyev VNIIG" JSC will be submitted for explanation.

T2-64	Please clarify "through rotating water purification grids": are these grids working all along the accident? Are they seismic design? Where are they connected electrically?	The rotating water purification machines are the elements of the PE system and operate in all modes of operation, including emergency conditions. According to the RF standards a rotating water purification machine refers to: - safety class 3NO as per NP-001-97 (PNAE G-01-011-97); - seismic category I as per NP-031-01. Power supply to a terminal box of the electric motor is performed at elevation above el. "0" of the pump station.
T2-65	Please clarify: "All the water conduits are laid in tunnels" are there four independent tunnels for each redundancy?	Tunnels of the safety systems are provided for pipelines of the cooling water system for essential loads (PE). In the tunnels it is provided to route the power supply cables for each channel of the pump station of essential loads (UQC) from the emergency power supply switchgear located in the standby diesel generator building (UBS). In accordance with the structure of the safety systems, the PE system consists of four channels which are independent in terms of process and electrical connections, as well as in terms of I&C systems.. To perform safety functions in emergency modes with a loss of coolant, it is sufficient to operate two of the four channels with an efficiency of 50% each.
T2-66	Please clarify: consumers from one specific train of the safety systems are connected to a specific channel of the PE system? Or one specific train of the safety systems could be connected to different trains of the PE system?	In accordance with the structure of the safety systems, the PE system consists of four channels which are independent in terms of process and electrical connections, as well as in terms of I&C systems. The system facilities are arranged so that failure of one system channel does not lead to failure of the other channel (via ventilation systems, building structures, transportation routes, cooling water channels and cable communications).
T2-67	Are the underground pipelines (laid in tunnels) of PE system for spray ponds designed to withstand seismic loads of category I (SSE)?	The PE system pipelines routed in the tunnels according to the RF standards refer to: - safety class 3NO as per NP-001-97 (PNAE G-01-011-97);- seismic category I as per NP-031-01. According to the results of the performed calculations the pipelines meet the DBE-strength requirements of PNAE G -7-002-86.
T2-68	Could you please clarify what are the extreme weather conditions considered for this atmospheric heat sink from the point of view of hot weather?	"Analysis of cooling capacity of the spray pool under extreme temperatures" BL-12183 is performed. The purpose of this analysis is to check cooling capacity of the spray pool under extreme temperatures with probability 0.01% (influence of extreme outdoor temperatures on the thermal mode of the spray pools). Maximum temperature plus 38.7 °C and humidity 20% are accepted for the hot period.
T2-69	Could you please clarify if there are any restrictions/special operating measures-maneuvers in the pools in case of low temperatures (i.e bypass of spray)? Could be these restrictions/special operating measures- maneuvers needed during an accident?	Under minimum outdoor temperature minus 41.5 °C and humidity 75% temperature of the water cooled in the spray pool in the rated mode (within a day of operation without nozzle spraying) does not exceed the allowable value of plus 28.0 °C. Due to significant difference between the cooled water temperature and the outdoor temperature, extreme steaming over the open spray pool occurs leading to adverse increase of make-up flow rate. When the cooled water temperature in the spray pool reaches plus 18.5 °C it is recommended to put half of the nozzles in operation.
T2-70	Could you please clarify "10% probability"? Is not the hot five-day period related with data from a certain period of time (years)?	10% probability means that once in 10 years the outdoor temperature reaches this value.
T2-71	Could you clarify "the capacity of each spray pool ensures the operation of the system"? Is it related with the volume stored in the spray pools taking into account losses of water inventory in the hot five-day period of 10% probability + other inventory losses?	The required make-up volume for the spray pools is determined taking into account water loss during evaporation and wind blowing. Make-up of the PE system is performed with chemically treated water supplied from the water treatment building (UGB) through make-up pipelines GHC to the water receivers of the pump stations for essential loads (UQC)
T2-72	Could you please clarify "measures for supply of make-up water must be arranged"? As per later paragraphs it is not related to make-up water to the spray pools but to the SG PHRS? Is that correct?	The additional technical measures according to the Technological Regulations can be implemented for water replenishment in the spray pools by the mobile pumping equipment and for long-term removal of the core residual heat to the ultimate heat sink through the second circuit (SG PHRS) in case of BDBA involving total loss of all AC power supplies, total loss of feed water, as well as a part of the range of accidents with the primary circuit coolant leakage in case of failure of the active safety systems.
T2-73	Could you please clarify the meaning of "shore pump station" and how is it defined ""cold" initial state"? Among the "various operation modes of the reactor plant" what are the cases included that are related to the outage?	The shore pump station is a make-up pump station located on the bank of the river Viliya.
T2-74	According to the design what is the seismic qualification (seismic category) of the additional water piping system GAC supplying make-up water to the cooling towers circulation system?	All structures and main equipment of the system refer to seismic category II as per NP-031-01.
T2-75	Could you please clarify if the transportation of chemical reagents is considered for normal operation only?	Adjustment of water chemistry in the process system tanks is made only during normal operation.
T2-76	Could you please clarify if other operational modes during outage have been considered? (for example: ¾ loop level, vessel head raising, i.e. end of life, full core loaded, open primary circuit).	See response to comment T2-45
T2-77	How are the feed-water pumps cooled in case of LoUHS? Is the cooling emergency power supplied?	Under loss of heat removal and loss of alternative heat removal operation of the feedwater pumps is not provided. Heat is removed from the reactor plant by the SG PHRS.
T2-78	How is the integrity of the main coolant pump seals ensured in case of LoUHS?	The calculations additionally take into account leakage rate of the primary circuit 50 l/h from each RCPS.
T2-79	For the operation of SG PHRS BRU-A and BRU-K must be closed. How it is ensured?	BRU-A and BRU-K must be controlled from the MCP. Along with this the rate of cooldown through the SG PHRS is higher than through the BRU-A. Joint operation of SG PHRS and BRU-A is permitted. Operation of four PHRS channels leads to pressure reduction in the steam generators according to the PHRS operation performances. As a result the BRU-As on the steam lines of all steam generators close and loss of boiler water in steam generators is stopped. According to Table 5.1.2.5 of stress-test /31/ the BRU-A opens in 4,2 s. after actuation of the reactor emergency protection and closes at the 84th s. after actuation of the SG PHRS (SG PHRS is actuated in 30 s. after actuation of the reactor emergency protection)
T2-80	Spent fuel pool cooling: How is the JNB50AP001 pump cooled? Is emergency power supply available for the FAK70 valves? Is there a procedure available for this scenario? What is the design basis temperate of the SFP? Is containment venting necessary to avoid damage of the SFP?	Electric motor of pump JNB50AP001 is provided with air cooling. Valves FAK70 are powered from the BDBA power supply channel 7,8. The FAK system components located in the reactor building are designed for temperature 150°C and pressure 0.4MPa (gage). It is not required to remove air from the protective containment.

T2-81	Could you please clarify the different times to remove heat from the primary circuit using each of the methods described? What is the minimum level reached in the vessel in each case?	In case of loss of cooling water from the condensers of the turbine plant, the process of the Unit cooldown and maintaining in a safe state is performed through the secondary circuit by BRU-A. The SG make-up to ensure the BRU-A operation is provided by emergency feed water pumps of the safety systems. When temperature in the primary circuit decreases up to 150 C the JNG1/JNA safety systems are connected and heat is removed through these systems. Under failure of the spray pool or BRU-A the reactor plant is cooled down by the SG PHRS. The SG PHRS can remove residual heat of the reactor plant in the self-sufficient mode for 72 hours from the beginning of the accident, provided that the water reserves of the 4 emergency heat removal tanks are used. If 3 out of the 4 emergency heat removal tanks are used, the self-sufficient operation for not less than 24 hours is provided. The further operation of the SG PHRS provides make-up of the emergency heat removal tanks by pump JNB50 from the LCU tanks. Along with this the level in the reactor and spent fuel pool does not decrease.
T2-82	Could you please clarify what are the weather conditions (highest and lowest temperature) considered for the ultimate heat sink?	"Analysis of cooling capacity of the spray pool under extreme temperatures" BL-12183 is performed. The purpose of this analysis is to check cooling capacity of the spray pool under extreme temperatures with probability 0.01% (influence of extreme outdoor temperatures on the thermal mode of the spray pools). Maximum temperature plus 38.7 °C and humidity 20% are accepted for the hot period. For the cold period minimum temperature minus 41,5 °C and humidity 75% are accepted.
T2-83	Could you please clarify the "water level in the emergency heat removal tanks"? Could you please clarify how is level monitored during the different accidents described?	Water level in the EHRT is monitored by the sensors during BDBA and BDBA I&C. Information is displayed on the MCR at the BDBA control panel.
T2-84	What is the design protection against extreme external events of the components of the systems for residual heat removal to ultimate heat sink, which are situated outside the protected buildings (e.g. protection of the spray ponds against tornado, etc.)?	To maintain the water volume in the pool in case of tornado each section of the pool is divided into two parts: open and closed. The closed part with a clear width of 12,00 m is located along the pool perimeter. Calculations have been performed by "Vedeneyev VNIIG" JSC. within the Project "Belarusian NPP. Units 1, 2. Justification of cooling capacity of the spray pool" (inv. N BL-02691 s/o). In case of falling items during tornado and damage of the pipelines with nozzles redundancy of the channels is possible for the period of repair works.
T2-85	How are the JDH pumps cooled in case of LoUHS?	Operation of system JDH under loss of the ultimate heat sink is not provided.
T2-86	Could you please clarify, is it a single pump? How does it cope with single failure?	It is the only one pump. Single failure at the DiD 4 level is not considered. Redundancy is provided by two connections for non-routine facilities.
T2-87	What kind of measures are planned to consider a multi-unit accident on the site, such as sharing of resources, emergency response and rescue teams, external support, deliveries, etc.?	See response to comment T3-31. According to the Decree of the Council of Ministers of the Republic of Belarus No. 495 dated April 10, 2001, "On the State System of Prevention and Mitigation of Emergencies", Article 21, the Commissions for Emergency Situations and Emergency control authorities at all levels provide rescue and other emergency actions during mitigation of emergencies. If the scope of the emergency situation exceeds available manpower and resources to localize or mitigate the emergency situation, the commissions seek the assistance of a higher commission for emergency situations. The higher commission for emergency situations coordinates or takes the lead in mitigation of the emergency situation and provides the required assistance. If the available manpower and resources are insufficient, the manpower and resources of the republican authorities and other state-owned organizations subordinate to the Government of the Republic of Belarus are duly engaged".
T2-88	Could you please clarify what are the expected times for dehydration and levels associated?	With the blackout (SBO), the spent fuel pool is heated up to 100°C during 16 hours. The time of the spent fuel pool boiling-off to the FA uncovering is 73 hours. The total time is 89 hours. Under complete power loss, heat removal from the RP core (with operating 3 PHRS and 4 EHRT) stops in 72 hours. Under complete unloading of the spent fuel assemblies for 10 years of operation in the SF pool and under power loss in 41 hours uncovering of the FA heads occurs. When operating at power at the beginning of the reactor campaign (after refuelling), and under power loss, uncovering of the FA heads occurs in 89 hours. For detailed results, please refer to item 6.1.2 and /31/
T2-89	Could you please clarify: all these actions are in the BDBA Management Manual? "- prompt assessment of the equipment condition for the NPP design ultimate heat sinks (PA, PC, PE systems), as well as the availability and operability of the systems and equipment; - preparation for operation of additional technical means to make up the SG PHRS and the spent fuel pool."	The developed symptom-oriented emergency procedure BDBA MG will contain the procedures for monitoring and restoring the critical safety function that are a part of the severe accident management strategies, and, among other measures, include: - prompt assessment of the equipment state for the ultimate heat sinks (PA, PC, PE systems); - defining availability and serviceability of the systems and equipment; - preparation for operation of additional technical means to make up the SG PHRS and the spent fuel pool. The described strategies are a part of the symptom-oriented BDBA MG.
T2-90	What are the parameters monitored? Are there several possibilities considered depending on how could the accident develop? "Monitoring and control are performed from the BDBA panel located in the MCR."	General information is given in item 7.3.9 of the National report. The List of controlled parameters on the BDBA panel is given in item 6.3.9 in the Report on the stress-tests /31/.
T2-91	It is stated that "Monitoring and control are performed from the BDBA panel located in the MCR". Are there design provisions for monitoring and control of the BDBA system performance from the Emergency Control Room or Emergency Response Center?	According to the regulations, ECR is not equipped with the BDBA panel. The SERCP is provided with the RP parameter monitoring means, control is not possible.
T2-92	Table 3.1.2.1: what does it mean *), **) and ***)?	Note to the table is missing. Value of applied designations is as follows: *) – necessity and sufficiency at NO and AOO; **) – necessity and sufficiency at DBA; ***) – necessity and sufficiency at BDBA.
T2-93	There is no information given for LoUHS during shut down operation with closed primary circuit. How is the decay heat removal ensured in this operational state? Are the steam generators and the SG PHRS available? Is there a procedure available for this scenario?	The time of fuel uncovering for the scenario with the closed reactor head exceeds the time for the scenario with the removed reactor head, as water does not boil off under the containment, and is supplied to the SG as steam and is condensed due to PHRS operation. Thus, a more conservative scenario with a removed reactor head is selected for the stress tests.
T2-94	There is no information given for LoUHS during shut down operation with opened primary circuit (reactor vessel head removed). How is the decay heat removal ensured in this operational state? Is there a procedure available for this scenario?	See response to comment T2-45
T2-95	How is the JNB50AP001 pump cooled? Is there a procedure available for this scenario?	Electric motor of pump JNB50AP001 is provided with air cooling.
T2-96	Is emergency power supply available for the valves in the make-up line? Is there a procedure available for this the make-up?	Valves on the EHRT and spent fuel pool make-up line are powered from BDBA power supply channels 7, 8.
T2-97	The overall technological scheme of power supply for equipment important to safety should be provided.	The diagram is attached (see the answer to G-12)

T2-98	To assess provided information in the chapter 6, a description of the location of the cable lines shall be provided, taking into account the power supply for the safety-related systems and the normal operation systems. Are they placed in separate trays? Do the cables have a fire retardant coating?	The layout drawings of the main cable routes are given in attached file T2-98.pdf Cables of the safety systems and normal operation systems are laid in different structures. Bearing structures of cable facilities in the NPP safety systems, as well as enclosing structures separating cable facilities of different safety system channels from each other and from similar normal operation facilities and devices are made of non-combustible materials with fire resistance equal to or over 1.5 h. It is allowed to lay single power cables not referring to the safety systems in cable facilities of the safety systems. For such cables throughout their route the requirements applied are the same as for components of the safety systems. In this case, within one channel of the safety system these cables are laid together with cables of the safety system without separation, and in the rooms of other safety system channels they shall be separated from other cables by enclosing structures with fire resistance equal to or over 1.5 h. Cables of the safety systems are fire resistant (min.30 minutes as per IEC 60331-21(23)), fire retardant (category A as per IEC60332-3-22). With the volume of combustible mass over 7 l for cables laid in groups, special coating is applied to prevent fire propagation.
T2-99	The mid-loop operation shall be analysed (see clause 3.2.6 of "Compilation of recommendations and suggestions Peer review of stress tests performed on European nuclear power plants" and clause 6.2.4 of "Peer review report Stress Test Peer Review Board Stress tests performed on European nuclear power plants"	See response to comment T2-45
T2-100	It is stated that "For the Belarusian NPP, loss of external power supply is a design basis condition analyzed in the SAR on the Belorussian NPP."	It is considered that external power supply can be restored to a maximum extent within several days. Taking into account fuel transportation from the central diesel fuel warehouse (00UEJ), operation of DG of one EPSS channel is supported for 7 days more. In case of start failure of all EPSS DGs it is considered that within 3 days (72 hours) either external power supply will be restored, or at least one DG will be started.
	If the loss of external power supply is a design basis condition for the Belorussian NPP and is analyzed in the SAR on the Belorussian NPP, a time for power supply restoration shall be specified.	
T2-101	It is stated that "- Emergency backup transformer with a power of 16 MVA, seismic category I, voltage 110/10 kV, powered from the "Vilia" substation through a cable line laid in the ground. The power of this transformer was selected so as to supply power to one EPSS (emergency power supply system) channel of each Unit (feeders from 110/10 kV substation are provided for all 10 kV sections of the Unit reliable power supply system…"	110 kV cable line from Viliya substation to the emergency transformer is laid separately from other cables in the ground. Two 10 kV cable lines from the emergency standby auxiliary transformer to 10 kV switchgear are laid separately from each other in the ground. Two 10 kV jumpers from 10 kV switchgear of the emergency standby auxiliary transformer to the backup power supply assemblies are laid in an exposed way on the cable structures together with 10 kV backup power supply cables, but separately from each other. In accordance with the diagram shown in the drawing (see the answer to G-12), backup power supply cables are used for connection to sections of the Unit.
	Is the mentioned cable line laid out separately from other power cables? It is important because inputs from the 110/10 kV substation are provided for all sections of the 10 kV of normal operation systems. The laid out of power supply cables for normal operation systems shall be described for corresponding assessment of presented information.	
T2-102	What type diesel generators are? Are they cooled by air or by service water? Taking into account principle of diversity at least one of diesel generators should by cooled by air.	DG set of the standby diesel power station/unit diesel power station has water/air cooling system (water of cooling systems of the high temperature and low-temperature circuits is cooled with air).
T2-103	It is stated that *"-each DG has its own self-contained auxiliary systems;"*	Cooling systems of the standby diesel power station refer to seismic category I as per NP-031-01 and perform their functions under OBE and DBE loads. Cooling systems of the unit diesel power station refer to seismic category II as per NP-031-01 and perform their functions under OBE loads. Process flow diagrams of the cooling systems of the standby diesel power station/unit diesel power station are given in the Design - Section 5.7.2, Volume 2, Book 4 BLR1.B.130.&.050702.0204&.021.LG.0001.
	For evaluation of DG self-consistency the cooling system of each DG shall be described including analysis of operation possibility during and after external events and in case of malfunctions of service water supply. The technological scheme of DGs cooling system should be provided on purpose to demonstrate if independence and reservation in the cooling trains of each DG are ensured.	
T2-104	It is stated that *"-An additional diesel fuel amount of 1160 m^3 is stored at site in the central diesel fuel warehouse (00UEJ) of (290 m^3 for DG of one EPSS channel of each Unit) to ensure additional stock for 7 days for DG of one EPSS channel of one Unit (this calculation is based on a flow rate of 204 g/kWh for one DG)."*	The common-plant diesel fuel storage warehouse refers to safety class 4 as per NP-001-97, seismic category III as per NP-031-01, therefore it is designed according to the general industrial standards. see response to comment T2-19
	The results of central diesel fuel warehouse (00UEJ) analysis to withstand extreme weather conditions and/or earthquake shall be described. Also it is necessary to present the design characteristics of the central diesel fuel warehouse (00UEJ).	
	It is stated that *"In case of NPP blackout, if the NPP auxiliary power supply is not restored within two days (48 hours) with DG in standby mode, the main and intermediate warehouses are refilled with diesel fuel of the required quality delivered from the regional oil supply points by road."*	Answer to the 1st question: The main warehouse and the central warehouse are the names of the same diesel fuel warehouse at the NPP. For each NPP Unit the diesel fuel stock is located: - in the main warehouse; - in the intermediate warehouse and in the supply tank of DG set of each channel (irreducible fuel stock). The fuel stock in the main warehouse is provided in the amount not less than required for operation of DG set of one channel for each NPP Unit at nominal load for at least 120 hours (5 days). Volume of the irreducible fuel stock stored in the intermediate warehouse and in the supply tank of each DG set is sufficient for operation of DG set of each channel at nominal load for at least 48 hours (2 days). If it becomes impossible to restore the power supply of NPP auxiliaries within two days (and return the DG sets to the standby) in the NPP blackout mode, the design provides for replenishing the main and intermediate warehouses with diesel fuel of the required quality from the regional oil product depots by means of oil tanker trucks. The fuel stock in the supply tank of each DG set is provided for DG set operation at nominal load for at least 5 hours. Unit diesel power station: The fuel stock in the intermediate warehouse is provided for DG set operation at nominal load for at least 24 hours.
	It is necessary to clarify what is the "main and intermediate warehouse", as previously in the Report a "central warehouse" was used for storage of diesel fuel.	

T2-105	What measures/requirements are applied to the regional points of supply of diesel oil to store the required quantity and quality of diesel fuel on site during the accident at the plant?	Answer to the second question: According to Decree by the Council of Ministers of the Republic of Belarus No.1800 dated 20.11.1998 "On establishing of the republican system of material reserves for emergency response", material reserves of various levels are established
T2-106	In section 6.1.2 it is mentioned: "The facilities designed for electrical equipment installation meet the requirements for ensuring their safety and operability in accordance with their classification and ensure protection against possible natural and man-caused impacts in the NPP area. The technical means are resistant to impacts caused by earthquakes and flooding". But no information about the qualification of equipment (capability of equipment to perform their functions at high / low temperatures, high humidity, increased pressure) is presented. The qualification of equipment shall be discussed.	The initial technical requirements for the equipment include requirements for ambient conditions (in continuous normal mode and in emergency modes) to be withstood by the equipment - temperature, pressure, humidity, radiation levels (where applicable), seismic impacts, treatment with decontaminating solutions. These requirements and methods to check their fulfilment after equipment supplier selection are specified in the technical assignments developed by the equipment manufacturers. At the equipment acceptance stage fulfilment of these requirements is checked by the acceptance commission.
T2-107	It is stated that *"The facilities designed for electrical equipment installation meet the requirements for ensuring their safety and operability in accordance with their classification and ensure protection against possible natural and man-caused impacts in the NPP area. The technical means are resistant to impacts caused by earthquakes and flooding."* 1. Please explain how emergency diesels generators and safety trains are constructed and located. 2 Are the emergency diesels generators and the trains physically separated? 3. Is "bunkered design" design used? The appropriate schemes and on-site pictures would be anticipated. 4. What requirements for fire resistance (class) are applied for the doors of electrical facilities?	Layout of building UBS is shown in attached file T2-107.pdf The DG and safety channels are physically separated. Fire resistance of the enclosing civil structures separating adjacent channels is 1.5 hours. Doors in electrical rooms have the following fire resistance: - min.0.75 h in normal operation rooms; - 1.5 h in rooms of the safety systems.
T2-108	It is stated that *"The condition of the Unit at the initial stage of the accident is characterized by:* *<…3. subcritical state of the reactor;>"* It is necessary to describe/explain why the subcritical state of the spent fuel pool was not taken into account in this analysis.	Fuel in the spent fuel pool is in subcritical state during operation. Therefore, any initiating event (accident) starts from the subcritical state in the spent fuel pool.
T2-109	During „stress tests" many countries operating pressurised water reactors as a problem indicated overheating of RCP seals, due to which additional loss of coolant is possible during the accident. Was this issue analysed and taken into account?	The analysis takes into account both controlled and uncontrolled primary circuit leakages in the amount of 2.15 m3/h, which corresponds to the maximum possible leakage rate during RP operation at the rated parameters. The specified volume of leakages includes: - leakages through RCPU seals - 4•0,05 m3/h; - leakages through pressurizer POSV - 0,35 m3/h; - sampling - 3•0,5 m3/h; - uncontrolled leakages in the amount of 0,1 m3/h. When the pressure decreases, leakage rate of the primary circuit also decreases (document /1/, Fig. 5.2). Mass yield of the primary circuit coolant through leakages after 72 hours is approximately 41 tons. Boric solution stored in the ECCS hydro accumulators recovers loss of the primary circuit coolant.
T2-110	It is stated that "*It is assumed that in 24 hours from the start of the accident Unit becomes uncontrollable because the reliable power supply batteries are discharged….* *Thus, during the accident design period (about 3.5 days), the maximum pressure values of the primary and secondary circuits are not reached, the acceptance criterion is met, the fuel pellets do not melt even locally (the temperature is less than 2540 °C for spent fuel and less than 2840° C for fresh fuel).* *Analysis of the blackout accident development in the course of three days demonstrates the following:"* What measures (organizational and technical) does the operator use to monitor the progress of the accident after 24 hours when the batteries are discharged?	After 24 hours, 500 kW mobile DG can be brought from the storage and connected to provide charging of the batteries and power supply to the required loads.
T2-111	It is stated that ,,Basic directions of the personnel actions in case of complete AC loss: - reactor plant transfer to and maintaining in the safe condition in accordance with the requirements of the Process Regulations, Instructions for the Reactor Plant Emergency Response, BDBA Management Guidelines, Severe Accident Management Guidelines; - prompt assessment of the condition of the NPP power supply equipment (including emergency equipment), as well as availability and operability of systems and equipment; - arranging for priority (urgent) works on restoration of power supply, including putting a mobile DG set into operation; - putting a mobile DG set into operation to ensure water supply to the PHRS tanks and spent fuel pool; - implementation of the plan "Measures to Protect Personnel in the Event of an Emergency at the Belarusian NPP" (if necessary)."	In case of initiating event with complete AC loss, the documentation specifies the following sequence of actions: - if initiating event with complete AC loss is diagnosed, actions of the personnel are specified by event-oriented BDBAMG (1.3-1.0); - if criteria for transfer to symptom-oriented BDBAMG are met (the operating personnel failed to determine an initiating event, failed to determine which event-oriented procedure shall be applied for accident management, overlapping of initiating events occurred and the operating personnel failed to determine which event-oriented procedure shall be applied first of all, application of the event-oriented procedures does not lead to expected results), the personnel start actions following the above-mentioned BDBAMG; - if criteria for transfer to SAMG are met – temperature at the core outlet exceeds 650 ºC – the personnel start actions according to SAMG. The actions indicated below are performed, if required, in the course of emergency response in accordance with the emergency procedure applied: - prompt assessment of the NPP power supply equipment condition (including emergency equipment), as well as availability and operability of systems and equipment; - arranging for priority (urgent) works on restoration of power supply, including putting a mobile DG set into operation; - putting a mobile DG set into operation to provide water supply to the PHRS tanks and spent fuel pool. Action plan is introduced on the basis of the respective criteria determined by amount of release of radioactive products and RP state. The initiating events are "Emergency readiness" and "Emergency situation" status at the NPP.

	It is unclear in what sequence the actions of the personnel in case of complete AC loss will be taken: simultaneously or one after another. Please explain, why the plan shall be implemented (please provide triggers for launching the plan).	
T2-112	In section 6.1.3 the passive heat removal systems SG PHRS and containment PHRS are mentioned. These systems are capable to maintain reactor unit in safe mode even if all active systems failed. It is written, that these systems consists of four independent channels and the efficiency of one channel is 33.3%. Applying single failure criterion we can assume failure of one single channel – in this case we can trust only 3 channels. But in the section 6.2.3 it is written: "the SG PHRS can remove residual heat of the reactor plant in the self-sufficient mode for 72 hours from the beginning of the accident, provided that the water reserves of the 4 emergency heat removal tanks are used. If 3 out of the 4 emergency heat removal tanks are used, the self-sufficient operation for not less than 24 hours is provided". Thus, the reduced number of channels drastically decreases the time of self-sufficient operation. It is necessary to describe, how the reliable operation of SG PHRS and containment PHRS will be ensured. Only 24 hours period of self-sufficient operation should be assumed for these systems in the analysis. Because all details regarding passive systems are referred to the reference [31], this report "Report on the conduct of a targeted reassessment of safety (stress tests) of the Belarusian NPP" BL-11752" shall be presented for the international nuclear safety community.	Justification of the events considered for stress tests is performed using the deterministic approach and actual scenarios of these events. Thus, all four PHRS channels can be put in operation. This condition is met only in case of water availability in the 4th EHRT. For all other cases, make-up is required.
T2-113	*"The Unit condition at the initial stage of the accident is characterized by:* *- availability of power supply from UPS of the system for power supply to the BDBA monitoring and management equipment (channel 7). The battery capacity is 2030 A*h. Power from UPS is designed for 24 hours without recharge of the batteries (with no regard to the operation of communication systems) constituting a part of the UPS. Connection of a mobile DG set (power 500 kW) within 24 hours to the switchgear of channel 7 – the cabinet (seismic category I according to NP-031-01, dust and moisture proof design – IP54, UHL1, hufter-proof, with a lock) located on the outer wall of building UJE at el.+1.400. The power calculation for the mobile DG (XKA70) takes into account a current equal to the current of a 10-hour battery discharge (203 A). At this current the fully discharged battery (at the estimated discharge time of 93-95 hours) will be charged to a full capacity of 2030 A*h in 10 hours. As DG is planned to be connected for a time less than that required for a full battery discharge, the time of recovery to the full capacity will be significantly less than 10 hours and will be determined by the discharge mode of the battery;"* For appropriate evaluation it is additionally necessary to present the description of the situation when it is not possible to recharge the batteries. Are there other ways to supply power for safety-related systems without batteries then BDBA occur? Does Belarusian NPP have an additional list of power consumers and power supply schemes during BDBA?	In the NPP design, BDBA is understood as a situation with loss of external power supply and start failure of EPSS (safety systems) DGs. In this case, provided that power supply from Viliya substation is still possible the safety system can be powered through the emergency transformer. If it is impossible, the required power consumers will be powered for 24 hours from the batteries, and after these 24 hours - from a mobile DG. A case of mobile DG failure was not considered.
T2-114	The situation in spent fuel pools is not analysed.	The information is given in item 6.1.2 of the National Report.
T2-115	Was arrangements for black start of co-located or nearby gas or hydroplants analysed as possible source of energy supply? What results of analysis and appropriate possibilities are?	An option of using an external generating source is considered in PSAR i. 8.1.2.2 "Reliability of NPP auxiliary power supply in case of failure of its own sources". If an independent power source not related to the power system is unavailable, in this situation the Units must be shut down. They will be put into operation after voltage supply to 330 kV switchgear through 330 kV overhead line. For start-up of one Unit of the Belarusian NPP, a total power of 91 MW shall be supplied to the auxiliaries. For this purpose, any of seven 330 kV overhead lines outgoing from the Belarusian NPP can be used. Taking into account a large scope of the Belarusian NPP auxiliaries, first of all stable operation of individual power centers with their own generating capacities shall be provided. The main large generating power sources of the Belarusian integrated power system are Lukomlskaya and Berezovskaya regional hydro-electric power plants, Minsk combined heat and power plants No.4 and No.5. After that voltage is supplied to 330 kV buses of the Belarusian NPP though one of the overhead lines from one of those generating power centers or their combination. Automation of the NPP black start is unallowable due to complexity and uncertainty of the emergency situation in the system.
T2-116	It is stated: "2) temperature in the MCR will not exceed 43 °C during 72 hours." It is doubtful that personnel will be able to work in such ambient conditions, especially wearing PPE. The issue shall be clarified when and how long time the personnel is going to work in MCR in accident case.	Same as for T2-53

T2-117	In the section 6.1.5 "Measures to Improve the NPP Stability in case of Power Supply Loss" the organizational and technical measures are proposed to mitigate the consequences of accidents with a complete loss of power supply. The guaranteed confirmation, that these measures will be implemented before the start of operation of NPP shall be presented.	See the answer to G-37.
T2-118	„Based on the information provided in the report [31], it can be concluded that the means available in the NPP design are sufficient, adequate and stable to protect against loss of power supply, including impacts caused by earthquakes and floods." Please note that this conclusion does not provide for any evidence of sufficiency, adequacy and stability to protect against loss of power supply, including impacts caused by earthquakes and floods. Please elaborate the mentioned means.	According to section 8.3.2 of the National Report, pump JNB50 is located above the zero elevation, therefore it is not exposed to flooding. The pump refers to seismic category I and located in the building of seismic category I. Thus, all conclusions on seismic resistance margins apply to this system (pump). The power is supplied from BDBA power supply channels 7 and 8 with a possibility of mobile DG set connection (the terminal cabinet is located above the zero elevation and refers to seismic category I). As for using means of the adjacent Unit - see the answer to T3-31.
T2-119	It is stated that *"To mitigate the consequences of accidents with a complete loss of power supply, the following organizational and technical measures are provided:…* *- in terms of organizational measures for preparation of operation and commissioning of an emergency standby auxiliary transformer with a power of 16 MVA 110/10 kV...;* *- in terms of organizational measures to allow for power supply from the neighboring Unit (if possible) through 10 kV assemblies of 330/10 kV standby transformers connected together with cable jumpers, it is required to develop appropriate operational instructions and sections of emergency procedures for its use at full loss of AC power supply."* The presence of one backup transformer per unit with a multi-channel power supply system for consumers can lead to a significant and unjustified loss of time when manually switching in the event of an emergency.	See the answer to T2-12.
T2-120	The overall technological scheme of cooling water for essential loads system should be provided.	The diagram will be submitted to PRT within the period of 12-16.03.2018.
T2-121	Taking into account information provided here and in Table 2.3.3.1 the redundancy of cooling water for essential loads system is not clear.	The cooling water system for essential loads (PE) operates in all operating modes of the Unit (including blackout), except for the mode with loss of external power supply, design backup AC power supplies and various fixed backup AC power supplies. As initial design data determining the required characteristics and parameters of the cooling water system for essential loads, the following criteria and requirements are applied to the system: - in the modes of normal operation at the reactor power operation, the PE system must ensure heat removal from the consumers through two channels PE 10 (20) and PE 30 (40) at a cooling water temperature from +4 to +28°C; - in the modes of planned cooldown of the primary circuit, the PE system must ensure heat removal from the consumers of intermediate component cooling circuit KAA through two channels, for example PE10 and PE30, and from the consumers of buildings UJA, UKA - through one channel, for example PE20, with a cooling water temperature not exceeding +28°C; - in the modes of design basis accidents, the system must provide heat removal from the consumers of safety systems through any two channels at a cooling water temperature of +31°C. In accordance with the structure of the safety systems, the PE system consists of four channels which are independent in terms of process and electrical connections, as well as in terms of I&C systems. To perform safety functions in emergency modes with a loss of coolant, it is sufficier
	Please provide the information about capacities of the system's channels and spray pools. How long one spray pool can ensure cooling of reactors and spent nuclear fuel pools of the both units?	For the PE system of each Unit, two spray cooling pools are provided: one spray cooling pool per two channels. Accordingly, the spray cooling pool is divided into two sections. In emergency modes heat removal can be carried out by any two channels of the PE system. If they are connected to one spray cooling pool, then one spray pool is sufficient for heat removal. The design characteristics of the spray cooling pool (capacity, overall dimensions, type and nozzle arrangement) are determined by the thermal hydraulic analysis of all operation modes of the system, based on the necessity to ensure the cooldown of the reactor plant in the mode of the maximum design basis accident at a temperature of cooling water supplied to the reactor compartment not higher than + 31°C. The volume of water in one spray cooling pool is 20000 m^3; the volume of water in the pipelines of one channel is 500 m^3. The volume of the spray pool ensures operation of two channels of one Unit for a long period of time without the need for making up (longer than 8 days).
T2-122	It is stated "The equipment and pipelines of the systems for heat removal to the ultimate heat sink refer to seismic category I and fulfill their functions in the event of an earthquake up to the level of the safe shutdown earthquake (SSE)."	The quoted statement applies to the components of system PE; the cooling tower is a cooler of system PA and does not belong to seismic category I.
	1. does it mean that cooling tower are of seismic category I?	The cooling tower does not refer to seismic category I as per NP-031-01.
	2. What is seismic qualification of spray pools?	The spray cooling pools belong to seismic category I as per NP-031-01.
T2-123	It is stated that *"The main ultimate heat sink in the normal operation mode is cooling water towers."* This system is not analyzed in Chapter 6.2.1. "Design Measures and Means to Prevent Loss of Ultimate Heat Sink, Resistance of Provided Measures and Means to Earthquakes and Flooding"	Evaporative cooling towers are designed to cool down the circulating water of the turbine condensers, auxiliary equipment and chillers. Subsection 6.2.1 considers operation of the safety-related systems The main system for removal of heat to the ultimate heat sink is not considered in the stress-tests and the National Report because it does not belong to the systems important for safety and it is used only in the normal operation modes and does not affect the reactor plant safety. In the emergency modes heat is removed from the reactor plant to the ultimate heat sink by the systems especially intended for this (these systems are detailed in Chapter 6.2.1.

T2-124	It is stated that ,,Also, rooms of the Units allow for storage of chemical reagents for water chemistry adjustment for tanks of the process systems. Therefore, the need for chemical reagents can be promptly satisfied by transporting them from one Unit to the other." As item 8 of the reference [23] "Norms and regulations for ensuring nuclear and radiation safety "Requirements for carrying out stress tests (targeted reassessment of safety) of the nuclear power plant", approved by the resolution of the Ministry of Emergency Situations of the Republic of Belarus dated 12.04.2017 No. 12" provides for the requirement to assess the simultaneous impact to all reactors and spent nuclear fuel pools located at NPP site, therefore, measures to assure the delivery of chemical reagents from other locations shall be foreseen.	Adjustment of water chemistry in the process system tanks is made only during normal operation. In the event of a simultaneous impact on all of the reactors and spent fuel pools at the NPP site, adjustment of water chemistry in the process system tanks is not required.
T2-125	Please specify the time of fuel damage in the Core and in the Spent fuel pools to understand time limits for recovery functions of heat sink.	The scenario when the function of ultimate heat sink is lost can be considered as an equivalent to the NPP blackout (6.1.2). The same time limits can be adopted as in Section 6.1
T2-126	It is stated that ,,The main directions of the personnel actions in case of complete loss of the design ultimate heat sinks are as follows: - reactor plant transfer to and maintaining in the safe condition in accordance with the requirements of the Process Regulations, the Reactor Plant Emergency Response Manual, the BDBA Management Manual. - putting the SG PHRS into operation, monitoring the operation of the system; - prompt assessment of the equipment condition for the NPP design ultimate heat sinks (PA, PC, PE systems), as well as the availability and operability of the systems and equipment; - preparation for operation of additional technical means for SG and PHRS making-up; - arranging for priority (urgent) works to resume the operation of the NPP ultimate heat sink systems (PA, PC, PE systems); - implementation of the Action Plan for personnel protection in the event of an accident at the Belarusian NPP (if required)." It is unclear in what sequence the actions of the personnel in case of complete AC loss will be taken: simultaneously or one after another. Please explain, why the plan shall be implemented (please provide triggers for launching the plan).	The detailed description of the measures and the sequence if their implementation is given in the BDBA Management Guidelines / Severe Accident Management Guidelines. Section 6.2.3 presents the main lines of actions of the personnel in the event of the described accident. Specific actions for accomplishing the main objectives are described in the emergency response documentation (BDBA Management Guidelines), in the Action Plan for Personnel Protection, Emergency Response Procedure in the Event of Floods, Destructions, Spills of Chemically Hazardous Materials, Icing in Rooms or on Equipment Affecting Safe Operation of the Belarusian NPP Facilities. Since the personnel of different NPP divisions are responsible for specific actions (the operating personnel, the personnel of the emergency response teams and groups), the main lines of actions are carried out in parallel. The Action Plan for Personnel Protection is implemented, when required, if the safe operation conditions and/or limits have been exceeded. When the function of ultimate heat sink is completely lost the safe operation conditions and/or limits are exceeded as per the number of the safety system channels that preserve operability, which is an initiating event for the Plan implementation.
T2-127	In the section 6.2.5 "Measures to Improve the NPP Stability in case of a Loss of the Ultimate Heat Sink" it is mentioned, that "to improve the NPP stability, the measures *are proposed* in regard to the making-up of the LCU tanks and the spent fuel pool". Also it is written "to maintain the controlled state after BDBA for more than 72 hours in case of loss of the ultimate heat sink at two NPP Units at the same time, the respective measures will be proposed". The measures for the making-up of the LCU tanks and the spent fuel pools in case of loss of the ultimate heat sink at two NPP Units at the same time shall be developed and implemented before the start of operation of NPP.	The measures shall be developed and implemented before the start of operation of the NPP.
T2-128	It is stated that *"Residual heat is removed from the reactor plant by the SG PHRS within not less than 72 hours."* The statement contradicts information presented in Chapter 2.3.3. (page 38): "The selected system design ensures its fully off-line operation without the operator intervention for at least 24 hours in accidents resulting in complete blackout and the SG feed water failure." and to information presented in Chapter 6.2.3. (page 99): *"If 3 out of the 4 emergency heat removal tanks are used, the self-sufficient operation for not less than 24 hours is provided."*	Justification of the events considered for stress tests is performed using the deterministic approach and actual scenarios of these events. Thus, all four PHRS channels can be put in operation.
T2-129	What is required time of autonomous operation of mobile diesel generators?	This is the time necessary to comply with the requirement for ensuring independent operation of the power unit in case of the auxiliary AC power supply: 72 hours (including operation of the storage batteries during the first 24 hours). It is considered that within this period of time the external power supply will be restored.
T2-130	It is stated that *"Monitoring and control are performed from the BDBA panel located in the MCR."* How the presented information is related to the Mobile DG. Generally, mobile DG is not operated from MCR. What parameters can be monitored and controlled in the MCR during the BDBA? Is it possible to control a level and temperature of water in the Spent fuel pools during the "Station Black out"?	1. The mobile DG is controlled locally by the operator. From the MCR the parameters are monitored, on the basis of which efficiency of operation of the equipment powered from the mobile DG is evaluated. 2. The information dealing with monitoring can be found in Section 7.1.3.3 of the National Report 3.Controlling - restoring of FAK or (JNB50 ensuring operation of SG PHRS).
T2-131	Was the possibility to use a water engine of a fire truck like an additional water supply source for spent fuel filling during an emergency analyzed?	The possibility to use a water engine of a fire truck is considered as a backup source of water.
T2-132	In the section 6.3.3 "Measures to Improve the NPP stability in Case of Loss of the Ultimate Heat Sink in Combination with the NPP Blackout" it is mentioned, that the PHRS tanks and the spent fuel pool are make up by a low-power high-pressure pump JNB50AP001 of the make-up system for the PHRS tanks". This shows how important is this pump JNB50AP001 – because this pump is necessary after 41 hours (for spent fuel pool make-up) and 72 hours (for PHRS tanks make-up) after NPP blackout.	The make-up system for the emergency heat removal tanks and spent fuel pool is describedin Section 1.3 of the Report /31/ there is a principle diagram of the make-up system for the emergency heat removal tanks and spent fuel pool (Figure 1.3). The components of the system for making up the emergency heat removal tanks and spent fuel pool belong to seismic category I.

T2-132	The connection of pump and water sources (tanks of the LCU system and the sump tanks of the containment) to PHRS tanks and the spent fuel pool shall be presented in more details. The justification of reliability of this system shall be justified. The guaranteed confirmation, that measures for the improvement of NPP stability in case of loss of power supply and the ultimate heat sink (implementation of two mobile DGs (one DG per NPP Unit)) will be implemented before the start of operation of NPP shall be presented.	
T3-1	The Emergency Operating Procedures (EOP), Beyond Design Basis Accident Management Guidelines (BDBAMG) and Severe Accident Management Guidelines (SAMG) are stated to be under development. Can you please outline the ongoing and future efforts and activities up to the completion and implementation of a severe accident management program for the Belarusian NPP and elaborate on the following topics, in particular : what is the exact scope of each document and how the transition between EOP, BDBAMG and SAMGs is implemented? What are the foreseen milestones for these documents and who is supposed to approve these documents before releasing them? Are there interfaces between these various procedures and potential link to other procedures at the governmental level? Will some of these documents being available for the consideration of the peer review team during the stress test visit?	The package of emergency response instructions in the format of the symptom-oriented emergency procedures shall be developed in compliance with the technical assignment approved by the Belarusian NPP. When developing, the IAEA requirements for the contents of the documents "NS-G-2.12 Severe Accident Management Programmes for Nuclear Power Plants" and RB-102-15 "Safety guide on the use of atomic energy", "Recommendations for the structure and contents of the BDBA Management Guidelines and Severe Accident Management Guidelines" will be taken into account. According to TA, the scope of Emergency Operation Procedure. The BDBA Management Guidelines and Severe Accident Management Guidelines must apply to all the operational states of the reactor plant, including accidents at the shutdown reactor. The symptom-oriented emergency procedures must be developed as follows: development of draft Emergency Operation Procedure, BDBA Management Guidelines and Severe Accident Management Guidelines; development of analytical substantiation for Emergency Operation Procedure, BDBA Management Guidelines and Severe Accident Management Guidelines; specifying the draft documents based on the data obtained when performing the analytical substantiation; approval of the documentation by the General Contractor, the General Designer of the reactor plant, The procedures of various manuals will have clear limits to applicability according to the purpose and the state of the power unit equipment, with the specified possible places and steps for transition from one to another in case of changed acci
T3-2	It is reported that "During the first 24 hours following transition of an accident into a severe stage, the automated controls help the operator perform a minimum amount of actions to provide integrity of the container.". Please specify what the automated controls and the minimum amount of actions include. The reference to the 24 hour time interval is confusing at this point and possibly inconsistent with chapter 6. More specifically, the 24-hour interval is mentioned in chapter 6 of the report as the maximum time capacity of the batteries in case of a SBO. In the same chapter it is also stated that pass to the severe accident phase occurs later, in particular after 72 hours i.e. after the water reserves of the PHRS tanks is depleted (see end of par. 6.3.1). Can you please elaborate on the meaning of the terms severe stage/phase at these points of the report?	Provided minimizing response within the first 24 hours is the requirement of the IAEA and EUR. In this case, this requirement is defined as unconditional for implementation when developing BDBA Management Guidelines / Severe Accident Management Guidelines (for BDBA and severe accident (SA)). According to the calculation results (section 6) - during the first 24 hours, no personnel actions are required to manage the accident in the RP or SF pool (SG PHRS, containment PHRS operates, there is sufficient water inventory in the SF pool). Then, actions to make-up EHR tanks and SF pool are required. When the transition of the RP accident to a severe stage occurs, the passive safety systems that do not require active components for their operation, are activated, and their I&C enables the operator only to monitor the behaviour and its compliance with the selected accident management strategy. The passive safety systems are SG PHRS, containment PHRS, the system for hydrogen removal from the containment (with the hydrogen concentration monitoring system). The operator's actions aimed at maintaining the integrity of the containment can be reduced to monitoring the passive safety systems during the first 24 hours, using means of monitoring and control of the BDBA panels in the MCR of the power unit.
T3-3	The actions for protection of the personnel in case of an accident should be better described and prioritized : in that sense how is the "action Plan for Protection of the personnel in case of Accident" interfaced with BDBAMGs and SAMGs as well as in-house plan emergency Plan ? what is the role of these documents in the different stages of the licensing process , What are the exact technological actions dealing with protection of personnels ? Is there a pyramidal approach of several documents going into details ?	The structure of the procedures for NPPs with VVER reactor plant implies an approach that allows for the transition from one procedure to another (each procedure has the criteria for entering and leaving it). Information on personnel protection in the event of an accident is specified in the plan "Measures to Protect Personnel in the Event of an Accident at the Belarusian NPP" (the in-house emergency plan). The plan "Measures to Protect Personnel in the Event of an Accident at the Belarusian NPP" is developed taking into account the data contained in the BDBA Management Guidelines and Severe Accident Management Guidelines. The "pyramidal approach" for detailing several documents is performed that is evident from the actions of the personnel when announcing of the states "Emergency preparedness" and "Emergency situation" at the NPP according to the "Procedure of declaration of emergency situation, rapid transfer of information in case of nuclear or radiation-hazard situation at the NPP" approved by the Decree of the Ministry of emergency situations of the Republic of Belarus, i.e. the plan "Measures to Protect Personnel in the Event of an Accident at the Belarusian NPP" and the "Procedure of declaration of emergency situation, rapid transfer of information in case of nuclear or radiation-hazard situation at the NPP" comprise the Emergency Action Levels (EAL) – a The "in-house emergency plan" is coordinated with the "Plan of protective measures in case of a radiation accident at the Belarusian Nuclear Power Plant" (external emergency plan)" developed by the Ministry of emergency situations of the R The plan "Measures to Protect Personnel in the Event of an Accident at the Belarusian NPP" is part of the documents that prove providing of nuclear and radiation safety and that belong to the List of documents for appealing to Gosatomnadzo
T3-4	It is reported that "In case the radiation background is higher than the design values, it may be concluded that the containment integrity is under threat or already damaged, thus requiring immediate measures to limit the release and spread of the radiation substances.". Please provide more details about the possible immediate measures to limit the release and spread of the radiation substances?	The main measures for localizing releases and preventing radiation particles spreading beyond the containment are maintaining operability and integrity of the containment, and in case of failed integrity of the containment, the following measures are provided for: containment localization; reducing pressure in the containment; reducing concentration and removal of hydrogen in the containment; leakage localization in the containment by treatment at the filters of the ventilation system of the annulus. The design of the Belarusian NPP provides for the following safety systems: The system of isolation valves located on the pipelines crossing the containment; the sprinkler system (JMN); the hydrogen removal system (JMT); the ventilation system of the annulus (KLC11-41).

T3-5	Site radiation monitoring should be better described, including number of radiation sensors, how they are checked, how frequently?	In case of a radiation accident, monitoring of the radiation situation is performed by the following resources of the radiation monitoring system of the Belarusian NPP: - measuring channels of the automated radiation monitoring system (MC ARMS); - mobile and portable means of radiation monitoring (MP MRM); - independent software & hardware complex of emergency radiation monitoring (ISHC ERM). As part of the measuring channels of the ARMS, detecting devices measuring the ambient gamma dose equivalent are located on the main buildings of the Belarusian NPP site along the path of possible propagation of radioactive contamination in the northern, southern, eastern and western directions. In addition, radiation in-process monitoring is performed by detecting devices monitoring the volumetric activity of water in the spray pools, groundwater, blowdown water of cooling towers, steam-and-gas relief from the turbine ejectors. The mobile and portable means of radiation monitoring perform radiation reconnaissance in the reactor and turbine buildings, along the perimeter of the damaged power unit and within the site. The reconnaissance aims to perform radiological assessment in the accident area (primarily following the personnel location and movement), and to specify the scope of radioactive contamination. The kit of portable and mobile devices includes dosimeter-radiometers of alpha-, beta-, gamma- and neutron radiation, mobile units for measuring the volumetric activity of aerosols and iodine. Radiation reconnaissance is also performed withi The independent software & hardware complex of emergency radiation monitoring comprises the means of radiation monitoring in the buildings of NPP SERCP, T SERCP and shelters. According to the in-house emergency plan in the event of an accident at the Belarusian NPP site, the organization and technical measures are introduced to localize the source and to mitigate the accident consequences, followed by radiation Testing operability, including metrological certification of equipment of the automated systems is performed as follows: – in the automatic mode - diagnostics of inner malfunction (self-testing of the bottom-level equipment); – in the automated mode - advanced remote diagnostics (computer-aided diagnostics with the displayed fault signal to the upper level for the operator); – in the manual mode, involving additional means and instrumentation (inspections of equipment when walking around, and maintenance operations according to the procedures established at the Belarusian NPP); – periodic functional testing; – metrological verification of the equipment. For mobile and portable means, the operability (including metrological certification) is checked by inspections and testing according to the operation manuals for the specific equipment and the relevant procedures at the Belarusian NPP. The frequency of operability testing and metrological certification of the equipment is determined by the operation manuals, verification procedures, functional test schedules, verification schedules and relevant procedures at the Belarusian NP
T3-6	Could you please provide some brief information about the general concept for processing the large amount of the resulting liquid radioactive waste in case of a severe accident?	Collection of the large amount of emergency radioactive waters is provided by sufficient capacity of the KPF and KPK systems reservoirs. These waters are processed by the standard equipment of the KBF and KPF systems. Final processing of liquid radioactive waste is performed in the LRW solidification plant of system KPC. The LRW solidification plant is designed for cementation of the vat residue and drying of spent sorbents with subsequent loading in reinforced concrete non-returnable containers. In the mode of normal operation of the power unit, LRWSP functions discontinuously, as waste are collected. Under normal operation of the power unit, 25 cub.m/year of the vat residue is expected. As a result of processing, 14 non-returnable containers with vat residue cement compound are formed per year (and 8 non-returnable containers with dry resin). The maximum expected number of non-returnable containers taking into account possible emergency situations is 38 non-returnable containers per year (PSAR 10.4.2.1). The maximum output of the LRWSP when concentrating the vat residue is 0.5 cub.m/h; the output per the final product (vat residue cement compound) is 3 non-returnable containers per day. Thus, the LRWSP is set with multiple capacity ma
T3-7	Are severe accidents taking place in the spent fuel pool considered in procedures and guidelines for accident management?	The main purpose to manage the accident in the SF pool is to provide the spent fuel pool makeup; the procedures of the spent fuel pool makeup from JNB50 are described in the National report. The procedures will be described in the symptom-oriented Severe Accident Management Guidelines.
T3-8	What are the means for forecast of potential radiation consequences available for Emergency Response Supervisor?	With the aim of assessing the situation, predicting the possibility of radiation consequences and elaboration of the proposals for normalizing the situation, the RECASS NT decision support system (DSS) developed by the Federal State Budgetary Enterprise "Research and Production Association 'Typhoon' " (RF), is used. The RECASS NT system serves to assess the situation and to forecast the consequences of the accidental environment pollution (as a result of releases, discharges, leakage, explosions and fires), and to develop the recommendations, as required, on the protective measures for the population within the accident area. The purpose of the system: - providing continuous collection of operational data on the state and level of environmental pollution, meteorological information, organization of loading, storing and archiving of the incoming data; - operative analysis of the situation and forecast of pollution spreading in case of accidental releases into the environment; - calculation of radiation exposure for the population; - development of recommendations for performing the protective measures for the population in case of emergency situations; - simultaneous multiple user access to operational and calculated data; - the possibility of making calculations in case of training.
T3-9	Who are the personnel in the Commission for Emergency Situations of NPP (NPP CES)? Please, clarify also who has (or it is planned to have) the responsibility for decision making in SAM. Is establishment of an Emergency Response Organization anticipated within the operator to take charge of the response (such an organization is not explicitly mentioned in the report)?	According to the Decree of the Council of Ministers of the Republic of Belarus No. 495 dated April 10, 2001, "On the State System of Prevention and Mitigation of Emergencies", and the Order of the General Director of the State enterprise No. 146 dated September 15, 2017, the Commission for emergency situations at the NPP (CES NPP) was established at the enterprise. The Regulations of the CES NPP, functional responsibilities of the CES NPP members were approved. The CES NPP operational authority is established, the warning and gathering procedure is determined. The Commission for emergency situations at the NPP is composed of the officials responsible or making decisions on emergency planning, response and providing emergency response actions. 1.3 The emergency response supervisor (ERS) at the Belarusian NPP site (Sanitary Protection Zone - SPZ) is the NPP director or, in his absence, his deputy. The CES of the Belarusian NPP prepares, and ERS (Belarusian NPP director) makes a decision on performing the rescue and other emergency actions. The ERS work on managing the rescue and other emergency actions depends on the time available, the evolving situation, and can be performed by the method of sequential or parallel procedures, and by combining the procedures. The method of sequential procedures is used provided that sufficient time is available for managing the rescue and other emergency actions performed consistently. The ERS makes a complete decision, then the Belarusian NPP divisions re The method of parallel procedures is used provided that there is lack of time to prepare the Belarusian NPP teams for performing the given tasks (usually in emergency circumstances). At the same time, the ERS determines the plan, reports As it is required to activate the Belarusian NPP teams, to deploy and to perform rescue and other emergency actions as soon as possible, the decision is made by the method of parallel procedures when assessing the situation. When assessing the situation, the following is identified: the dose rate of external radiation on the routes of the teams and in the areas of the rescue and other emergency actions performed; nature and levels of radioactive contamination of the area, buildings and structures of the nuclear power plant: extent of destruction of buildings and structures; presence of rubbles, their nature and size; possible centres of secondary damaging factors (fires, contamination with hazardous chemical substances); presence, location and possible health data of the injured people; ways of personnel and injured people protection. The ERS makes a decision on performing the rescue and other emergency actions based on assessment of the situation, calculations and conclusions. The ERS must specify the following in the decision on performing the rescue and other emergency actions under the radiation accident occurred: the plan of action, the tasks to the divisions, teams, manpower and resources; the main issues The decision is based on the plan of actions defining the following: object of concentration of the main efforts; the number of shifts and grouping of the teams by shifts, stand-by manpower and resources structure (if it is established); the orde The emergency response supervisor (ERS) must make a decision and order using the facility plan view, map or diagram. The solution and tasks are subsequently specified at the site (facility) of the actions. The ERS's decision on performing the rescue and other emergency actions must be graphically presented in the plan view (map) provided with brief description. The made ERS's decision must be approved by the relevant order. According to the standard Severe Accident Management Guidelines (SAMG) that will be taken into account when developing the NPP Severe Accident Management Guidelines, the development of strategies for managing the accident transfe

T3-10	Please provide more details regarding the relevant assistance from State organizations (e.g. Republican Special Operations Detachment of the Ministry of Emergency Situations of the Republic of Belarus) in relation to SAM.	For the purpose of supporting the decision-making on performing the protective and other emergency response measures, assessing the evolving situation, a system of local emergency response centres is established; the system consists of: the Belarusian NPP emergency response centre; the local emergency response centre of the Ministry of energy, Ministry of emergency situations, Ministry of Natural Resources and Environmental Protection, Ministry of Health, Ministry of Internal Affairs, State Security Committee; expert Research and Development Center of the National Academy of Sciences of the Republic of Belarus. The Belarusian NPP concluded a cooperation agreement with the regional emergency response centre (RERC) WANO Moscow Center dated 01.03.2016 Registration No. G14-2016 (Moscow, Ferganskaya str., 25). The RERC serves to provide expert (advisory) and engineering support in the event of an accident within the NPP site, a general-plant accident at the NPP with the VVER reactor plant of the WANO Moscow Center, and distribution of information on safety-related events at the NPP to its members. It forms a single information space to provide the NPP ERT response in case of a request for expert (advisory) engineering support by the NPP. The NPP participates in the activities of the RERC of the WANO Moscow Center per Level 3 according to the "Regulations on the Regional emergency response centre for NPPs with VVER reactors of the WANO Moscow Center". Registratio
T3-11	It is reported that the Belarussian NPP has a training centre equipped with simulators and training materials for training and exercising personnel in emergency situations. Please explain the status of the training centre and the status of operator training. (section 7.1.3.4 page 114 also notes that "The common plant set of anti-emergency training for operational personnel" is under development" – please clarify its status).	The personnel is trained according to the training schedule. "The common-plant set of programs of emergency response training for the operating personnel" will be developed by 01.05.2018. According to the General Contract for the Belarusian NPP construction, personnel must be trained to operate two NPP power units. Training is performed according to General training schedule for the operating personnel of the Belarusian NPP for 2015-2018 and the annual training schedules developed based on the General training schedule. According to the approved schedule date, the employees of the Belarusian NPP will be on probation at the sixth power unit of Novovoronezh NPP-2 of the Russian Federation. Under the General Contract, the Novovoronezh Training Center of JSC Atomtekhenergo (NV TC ATE), according to the schedule for the development of training aids, submitted for consideration and approval the following training aids adopted to the Belarusian NPP project: 10 training courses with a set of topic plans and manuals (out of 19 planned), 60 administrative guidelines (100% of the planned), 81 computer training systems with training scenarios (50% of the planned), 168 posters (37% of the planned). Currently, this documentation is validated with the participation of specialists of the Training center (TC) and the departments. Under the General Contract, since September, 2017, the training staff of the NV TC ATE provide practical training of the operating personnel of the main control room at the workstations of the Turbine control lead engineer, Reactor control lea The experts of the NPP departments were trained by individual training programs to perform their duties (participation in acceptance inspections, input control of equipment, analysis of design, shop and working documentation, supervision of e Now, a local simulator of I&C system of the water treatment plant (WTP I&C) has been manufactured and is subject to independent and comprehensive shop testing by the manufacturer. Till the end of 2018, FP I&C and WTP I&C local simulators will be put into trial operation; the simulators will be used for personnel training at the Belarusian NPP. The TC developed the following: training aids and a stand for practical training of the personnel of the thermal instrumentation & control shop (TICS), the course "Automation fundamentals"; training aids in various activities (labour protection, industrial, radiation and nuclear safety, safety culture, selective review of radiation and nuclear accidents, methods for preventing staff errors, etc.). Primary training and advanced training of managers and experts of the Belarusian NPP in the educational institutions of the Republic of Belarus are provided. When training for obtaining certificates of Gosatomnadzor for performing activities when using atomic energy, the following personnel was subject to training and a primary examination in the commission of the State Educational Institution "RI Personnel directing the operating organization; personnel providing physical protection of the nuclear facilities, nuclear materials, spent nuclear materials, operation radioactive waste; personnel, providing accounting and control, collection, pro The procedure for testing the theoretical knowledge and practical skills of the operating personnel is developed and introduced, i.e. "Guidelines on testing theoretical knowledge and practical skills of performing the technological process for obt Activities on preparation for acquisition of license for works execution in the field of the nuclear energy use are conducted according to the plan approved and agreed with the Gosatomnadzor.
T3-12	What kind of organization will be in place in case of an alert and what are the criteria for activation in an urgency situation? How is information sharing organized especially through on-site and off-site Emergency Plans ? Please give further details on off-site Emergency Plan ?	With account of the possible situation aggravation, certain man forces and means of the State Service for Emergency Situations of Grodno, Vitebsk and Minsk regions will be required to prepare for civil defense activities in the area (distance) of advanced planning. In the case of radiation accidents, the state bodies subordinated to the President of the Republic of Belarus and to the Government of the Republic of Belarus, to the regional government (Minsk) solve the following tasks: Ministry of Internal Affairs: - participation in public announcement of the accident and conduction of emergency and rescue operations and other urgent works; - participation in the operations to block the contaminated area; - control over access to the contaminated areas, over people's travels through the established borders of contaminated areas; - enforcement of public order and provision of public safety of temporary resettled population at collection points, at interim resettling points, at resettling points; - provision in collaboration with the military road inspection of safety and regulation of the road traffic during evacuation, regulation of travels on routes, crossings, contaminated area for the benefit of top-priority travel of man forces of the State Service for Emergency Situations; - convoying evacuation motorcades; - organizing, keeping registration and targeted information work at the resettled points, keeping record of the resettled populati Ministry of Housing in collaboration with the local executive and administrative bodies: - provision of community services to the population;- decontamination of public and domestic waste waters, construction (establishment) and organization o Ministry of Health: - organization and provision of the medical aid to the NPP staff and population suffered from the NPP accident; - preclusion of radioactive iodine accumulation in thyroid gland by taking iodine; - coordination of works on tra Ministry of Foreign Affairs:- submittal to the foreign countries that suffered from or likely to suffer from the radiation accident of information according to clause b Art. 2 of Convention on Prompt Notification of a Nuclear Accident. Ministry of Information: - assistance to the republican state bodies in prompt public announcement via the mass media of the population and administrative bodies on the radioactive accident, progress of consequences elimination and behaviors rules; - assistance to the republican state bodies in dissemination via the mass media of information on the inhabited areas contamination, on hazard level on the neighboring areas, on safety measures when taking meals, on ways of food cooking a Ministry of Forests - radiation control over forests pollutions; - organization and combating forest fires on the contaminated area; - conservation (decontamination) of forests polluted with radioactive substances; - development of proposals for the forests use in conditions of the radioactive contamination. Ministry of Defense - control over radiation in the deployment areas of the Military Forces of the Republic of Belarus, participation in radiation prospection in the contaminated areas; participation in the common announcement system of the administrative bodies a - provisions for transport aircrafts participation in delivery of the forces, means and materials to the areas of operations aimed at the accident elimination; - participation in decontamination of roads, buildings and machinery; - conduction of emergency and restoration works on the railroad, tracks and other railroad facilities; - participation in maintaining the restricted access to the territory, ingress from it; participation in other activities on the accident elimination according the legislation of Belarus.
T3-13	How is the return to a safe status of the NPP organized?	BDBAMG and SAMG actions ensure the NPP transfer to controlled and operated condition (provision of long-term heat removal from fuel (in the molten core, reactor plant and spent fuel pool), absence of radiation emissions, reduction of media temperature down to acceptable values in the containment, reactor plant and spent fuel pool). Actions aimed at transferring the NPP to a safe state are determined on the basis of the final state reached by BDBAMG and SAMG actions. Specific plans for the NPP dismantling after the severe accident are established based on the results of the analyses of the accident consequences, fuel condition and radiation effects on personnel, population and environment including the analyses of the fuel damage degree, equipment integrity and operability, necessity for decontamination of the radioactive contamination of equipment and adjacent territory. Short-term and long-term programs including measures for elimination of severe accident consequences and transfer of the NPP to a safe state are developed on the basis of this analysis and actual post-accident state of the NPP.

T3-14	What simulation tools for severe accidents are available in the training centre and what is the range of scenarii covered by the simulator (BDBA, SA, Low Power, shutdown, etc…) ?	Training centre of the Belarusian NPP corresponds to the systems and equipment of the prototype power unit and ensures simulation of all operation modes of the NPP in real-time scale (normal operation mode, transient mode, anticipated operational occurrences, DBA and BDBA modes up to simulation limits). Scope of simulation ensures practical training of operators, acquisition of professional knowledge and skills to control MCR/ECR and required for the power unit safe operation in normal operation mode, anticipated operational occurrences, emergencies, DBA and BDBA modes. List of simulated modes is determined on the basis of design and operation documentation of the prototype power unit and considering experience in the NPP operation. Specifics of the simulated modes form the basis for specification of the required scope of simulation, list of the simulated failures, simulation limits and required characteristics of technical facilities. To exercise the BDBA management measures the training simulator simulates the processes which can lead to fuel, core and reactor vessel damage. The considered hypothetical accidents are simulated within the entire time interval from an initial event up to a heavy stage of an accident. Severe accidents are simulated in the following scope: - FR heating up to the temperatures exceeding the design limits;- FR cladding oxidation and hydrogen generation;- core materials melting with fission products emission;- reactor core and reactor internals destruction;- heating up and destructi All safety-related systems controlled from the MCP / ECP and the systems having influence on transition processes in the reactor plant are subject to simulation at a full-scale simulator. Boundary values of the design parameters setting the simulation limits are determined taking into account the limits of fuel damage confirmed by the design justification and the limit design parameters of the main equipment specified in the te The full-scale simulator provides the possibility to simulate all transition and emergency operation modes of a prototype power unit including the BDBA mode till the moment of reaching the stable operation state according to the design require The list of the beyond design-basis accidents simulated at the full-scale simulator: - failure of all AC power sources for 24 hours accompanied by failure of the reactor emergency protection;- loss-of-coolant accident due to rapture of the primary circuit pipeline DN 850 with total loss of all AC power supply sources (more than - loss of AC non-emergency power supply (2nd category of reliability) of the auxiliary fixed equipment (NPP blackout);- MSIV spurious closing;- uncontrollable removal of one absorbing rod or group of absorbing rods at minimum controllable po
T3-15	Are computational aids provided as part of the SAM?	For forecast analysis of radiation consequences the relevant software is provided. See response T3-8
T3-16	With respect to mobile equipment it is reported that two mobile 500 kW DGs (one per unit) will be available. Please clarify where the mobile DGs are stored, ie on or off site. Please clarify how many mobile DGs in total are provided for emergency power supply for each unit, including those for the spent fuel pools.	The project provides one mobile diesel generator. To improve resistance of the NPPP to external hazards it is planned to increase the number of mobile diesel generators (two diesel generators: one generator per each power unit). In addition, see the answer to G-11.
T3-17	It is stated that in order to maintain the SG PHRS function it will be necessary to periodically make-up the LCU. Please clarify how this is achieved.	see T-3-68.
T3-18	In addition to the DGs various fire fighting trucks are identified, however it is not clear what other mobile equipment, if any, will be available. Please clarify whether any other mobile equipment is required and where it will be stored, or any additionnal fixed power sources ? A description of the transportation means as well as availablity of roads is ensured to transport them.	In case it becomes necessary to prepare transportation routes and manoeuvres of resources and manpower of NPP, the in-house emergency plant envisages to use the resources of the emergency group of transport and motorization of the Belarusian NPP; this group is staffed and equipped on the basis of the NPP vehicle fleet company. This group will be equipped with the following vehicles and motorization: buses, motorcars, commercial and special vehicles, excavators, bulldozers, mobile cranes, dump trucks, mobile compressors, mobile 0,4 kW electric power stations, decontamination and special treatment vehicles. To link motor road network of the Belarusian NPP with the public motor roads, the access road connecting NPP with national road P-45 was paved: Goza-NPP-Ostrovets. Length of the access road is 1.85 km. The NPP site is accessed also from the side of the construction and installation base via the road that connect the construction and installation base with national road P-45: Goza-NPP-Ostrovets. This way, the access to the NPP site is provided from three directions: Via national road P-45 from the eastern side, then via the motor road Goza-Ostrovets and main access road; via national road P-45 from the western side, then via the motor road Goza-Ostrovets and main access road; Via national road P-45 from the southern side, then via the second entrance nearby the second Unit. To service operations of NPP, including for fire fighting necessities, the territory of the NPP site has a network of on-site motor roads and access roads to buildings and facilities. The design provides no additional equipment to the DGs on the NPP site.
T3-19	What kind of agreements with external support forces (fire men, hospitals, etc) already exist for assistance to the plant in case of emergencies? Elaborate on Unit 2 of fire fighters and its role on supplying water to emergency systems ? Which kind of training plan is developped in accordance with these missions ?	In case of an emergency situation, external assistance shall be rendered in frames of the State System of Prevention and Mitigation of Natural and Human-Induces Emergencies, and concluding any additional agreements is not required. Technical capabilities of the structures and departments of the Ministry of Emergency Situations of the Republic of Belarus allow to arrange water supply to the emergency power systems. Structures and departments of the Ministry of Emergency Situations carry out the table-top exercises regarding the water supply practices to the fire fighting units.
T3-20	For the on-site activities in the case of a severe accident such as connection of the DGs is there evaluation of radiological situation available as needed for assessment of feasibility of accident management actions?	Accidents management strategy of the project is based on performance of all required actions to transfer the power unit to the controlled state remotely from the MCP/ ECP. Capability to act outside MCR/ECR for mitigation of the accident consequences, rescue of the staff or prevention of catastrophic consequences is defined on the basis of the actual radiation situation on the site following the radiation reconnaissance. In case the containment preserves its confining properties, the personnel may have a short-term access to the site with use of personal protective equipment for skin and respiratory organs, and inside a specially equipped vehicles. The movement routes and relevant activity procedures are defined on the basis of the actual radiation situation and planned radiation exposures. In accordance with the "Plan of measures protecting the Belarusian NPP personnel in case of radiation accident (internal emergency plan)" BLR1.E.534.&.&&&&&&.&&&&&.000.YN.0001, after identification of all hazardous factors permits are being issued to the personnel with a strict adherence to the special control procedures, including: staying time estimation on the basis of permissible personnel exposure limits, identification of supplementary personal protective equipmer In the course of implementation of emergency activities a relevant monitoring is performed with a purpose to define a time of job accomplishment because of a changed radiation situation, and, consequently, prevention of over-exposure of th

T3-21	Description of the shelter for the personnel: communication means, autonomy, habitability, workforces available, number of people etc should be provided, as well as on-site means (food, water, etc…) to maintain it operative.	Shelters of the Belarusian NPP provide protection of the staff against the adverse factors of natural and human-induced emergency situation, including: extremely hazardous chemical substances, radioactive and ionizing effects caused by these substances, high temperatures and combustion products released during fires and collapsing buildings following explosions and earthquakes. Shelters of the Belarusian NPP must be ready to immediately accommodate those who seek protection. In the everyday routine operations previsions are made to ensure integrity and technical readiness of civil structures and equipment of the civil protection shelter. Following annunciation of the "Emergency preparedness" or introduction of a enhanced preparedness mode, shelter maintenance staff are brought into readiness, open the shelter entrance doors, check the habitability systems and wait to immediately receive the shelter-seekers. The territory of the Belarusian NPP site houses: NPP SERCP for 100 sheltered persons. It is equipped with three ventilation modes: Clean ventilation, ventilation with filtering and air regeneration mode (complete isolation). Civil defence shelter for 1200 sheltered persons: standalone shelter with a receiving capacity of 1200 persons. It is equipped with three ventilation modes: Clean ventilation, ventilation with filtering and air regeneration mode (complete isolat Civil defence shelter for 600 sheltered persons: standalone shelter with a receiving capacity of 600 persons. It is equipped with three ventilation modes: Clean ventilation, ventilation with filtering and air regeneration mode (complete isolation In the city of Ostrovets there is a T SERCPon for 100 persons. It is equipped with three ventilation modes: Clean ventilation, ventilation with filtering and air regeneration mode (complete isolation). All the protective facilities are equipped with independent electrical power sources (DG), water supply (devoted wells), water disposal facilities. The habitability capabilities of the shelters enable to stay inside up to 5 (five) days.
T3-22	The radiation exposure evaluation criteria should be described in terms of the target values and how these values have been specified.	Section 7.1.2 states that the Emergency Action Levels are developed by the Management of the Belarusian NPP in accordance with the approaches set forth in the IAEA GSR documents Part 7 "Preparedness and Response for a Nuclear or Radiological Emergency", GSG-2 "Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency". General response criteria set forth by the regulatory requirements of the Republic of Belarus comply with the above IAEA requirements. GSR Part 7 , GSG-2. NPP meets safety requirements provided its radiation effect on the personnel, public and environment in normal operation conditions and during design basis accidents does not exceed established exposures of personnel and public as well as normative-based emissions and radioactive substance content in the environment, and limits this impact during beyond design-basis accidents, too. NPP safety must be ensured by consistent implementation of the defense-in-depth protection based on application of physical barrier system across the path of propagation of ionizing radiation and radioactive substances into the environment and system of engineering and organizational measures to protect the barrier and maintain their performance as well as for protection of personnel, public and the environment. The main objective of safety assurance at all NPP life cycle stages is implementation of efficient actions aimed at prevention of accidents and protection of the personnel and public by means of prevention of emission of radioactive product NPP is considered to be safe provided : -its radiation effect on the personnel, public and environment in normal operation conditions and during design-basis accidents does not exceed established values; -during BDBA the radiation effect is limited by reasonable values. The acceptable values of radiation exposure during BDBA are specified: 1 Belarusian NPP. Technical Assignment for NPP in the Republic of Belarus BLR1.B.130.&.&&&&&&.&&01&.000.MB.0001 Book 1 1.4.4.3 Design Basis and Beyond-Design Basis Accidents. In combination with the probability target values, BDBA consequences with severe core damage shall be mitigated to protect population/personnel and environment. During BDBA with severe fuel damage (release probability 10-7 1/year per reactor) beyond the established protection measures zone introduction of the population protection plans is not required. Equivalent exposure doses for a limited part of the population (critical group) at the boundaries of the protection measures zone and beyond shall not exceed 5 mSv for the whole body and 50 mSv for individual organs for the first year after the Attachment to T3-22 Appendix 20 to the Hygienic Standard "Radiation Exposure Evaluation Criteria" (Table 1) and 2
T3-23	What are the general response criteria to prevent deterministic effects and reduce the risk of stochastic effects in emergencies?	See Attachment to T3-23 Appendix 20 to the Hygienic Standard "Radiation Exposure Evaluation Criteria" (Table 1 and 2)
T3-24	How is functioning of the communication system ensured and is there a redundancy to make sure that communication will remain available with the NPP site (e.g. satellite supported systems) and for how long? Please describe the internal communication systems (6) and the redundancy and provide more information about the assessment of the reliability of the external communication systems, in particular in case of extensive destruction of infrastructure and a prolonged loss of power supply	The information is provided in /31/, pages 309-313.
T3-25	It is noted that there is an Emergency Control Room (ECR) however it is not clear what instrumentation is available and what systems can be operated from the ECR. Please provide further clarification as well as provisions taken to ensure its ability to be operative.	The ECR shall meet, in particular, the requirements of i.4.4.3 (NP-001-97), i. 2.4.17, 2.4.19 (NP-082-07). Unit is controlled from the MCR/ECR through the operator's automated workstation (WS) or from the segmented control panels. A set of control, monitoring and alarm elements located on the segmented panels or displayed on the WS screen is specified in the process assignment for automation, and based on this technical assignment a number of workstations and segmented panels to be accommodated in the MCR/ECR is determined. The ECR accommodates a workstation for ECR lead control engineer, sections of operational dispatch control equipment similar to the sections in the MCR and fire monitoring stations. Simultaneous control from the MCR and ECR is impossible. When Unit state control and monitoring is transferred to the ECR, signals from the MCR are blocked. The control transfer key is located in the ECR, the circuit solutions exclude false transfer in case of single failures in the command generation circuits. Arrangement of the MCR and ECR in relation to each other and the design solutions exclude loss of the MCR and ECR for a common reason. Survivability requirements for the MCR/ECR (ventilation, power supply, etc.) are similar.
T3-26	Regarding habitability of MCR and ECR, only conditions in design basis accidents are mentioned. Could habitability conditions in these places in case of severe accidents described? What are hardware provisions for ensuring habitability? What are the milestones to plan such operative guarantee ?	See the answer to T2-53.

<table>
<tr><td>T3-27</td><td>In the event of an accident to reduce personnel exposures when moving across the site it is noted that special vehicles are provided. Please clarify what these special vehicles are.</td><td>The means of the Belarusian NPP for response to radiation accidents do not include special vehicles. When required, special vehicles can be provided by other organizations involved in emergency response operations. In accordance with the republican plan for mobilization of resources and manpower for mitigation of an emergency situation, all the personnel and special-purpose equipment from the neighbouring units of the Ministry of Emergency Situations of the Republic of Belarus may be involved. The emergency response team of fire-fighting and rescue unit-2 has an emergency and rescue vehicle in configuration of chemical and radiation reconnaissance ERV (Mercedes). More detailed information on possibility to involve special means in case of emergency is given in i.6.1.2 /31/.</td></tr>
<tr><td>T3-28</td><td>It is noted that in the event of destruction of the MCR and ECR accident management activities can be carried out from the power plant's shielded emergency control posts (NPP & T SERCPs). Please clarify whether plant parameters are available in the SERCPs. Do the SERCPs already exist? Please provide some more information about the operations and functions that can be performed from the off-site control post. Are communication and information systems for this control post reliable?</td><td>1. During functioning (operability) of the APCS and APCS - NPP SERCP communication channels, Unit operation parameters are accessible for personnel of the emergency center and NPP ERT (emergency response team).
After a failure of the APCS and communication channels, the on-line archive of Units operation parameters is accessible in the NPP ERT.
2. The SERCP is under construction.
3. NPP & T (town) SERCPs do not perform functions of automatic or automated control for Units or emergency response management. Facilities of NPP & T SERCP emergency center are designed for providing personnel of the emergency center and NPP ERT with information for elaboration of solutions on emergency response management.
4. Reliability of the communication systems is sufficient for performing the functions indicated in i.3. All NPP operation parameters are accessible in the NPP & T SERCPs. T SERCP is equipped with independent process systems of life support in chemical and radiation environment conditions; it is also equipped with information systems, software & hardware systems and communication equipment, data transmission system; provided with necessary technical documentation and office equipment required for operations of the emergency response team. T SERCP capabilities are as follows: - radiation monitoring of process e
- radiation monitoring of 12.9 km radiation control area of the Belarusian NPP;
- analysis and forecast of radiation situation development based on processing of earlier prepared data (characteristics of facilities, list of critical parameters, personnel, population, data base, available possibilities for planned civil defence engi
- process monitoring of the Belarusian NPP equipment and systems status;
- environmental monitoring of the Belarusian NPP area;
- fire monitoring of the Belarusian NPP premises and area;
- on-line receipt of data on meteorological conditions of the Belarusian NPP area;
- maintaining of radiation and process monitoring data archives, their analysis;
- assessment of estimated current, cumulative and forecast radiation impact on personnel of the Belarusian NPP, population and environment ;
- preparation of draft orders to be issued by the emergency response supervisor and their follow-up;
- preparation of schedules, plans and required instructions for arranging the operations duty in the crisis centers of the Belarusian NPP at all emergency response stages;
- ensuring evacuation measures;
- ensuring assessment of the logistics support status during emergency response;
- ensuring sound recording of telephone talks;
- ensuring radiation exposure survey of the personnel in the crisis center;
- ensuring radiation monitoring of crisis center premises and performing other tasks.
The communication system covers all the scope of T SERCP activity and has a reserve. NPP & T SERCPs will be commissioned before delivery of nuclear fuel.</td></tr>
<tr><td>T3-29</td><td>Under which BDBA conditions have the suitability and availability of the instrumentation for monitoring of plant parameters been assessed? Are sensors for monitoring plant parameters in case of severe accidents independent for monitoring in case of design basis accidents?</td><td>Conditions for sensors during BDBA in the containment are presented in the table.
Sensors for thermal monitoring do not operate independently during DBA, they operate within I&C systems: EP-ESFAS (Emergency Protection - Engineered Safety Feature Actuation System) and MCDS (Monitoring, Control & Diagnostics System) designed for receipt and processing of signals from sensors and their transfer to the unit upper level control system and operational dispatch control equipment. The unit upper level control system and operational dispatch control equipment provide information on monitors and instruments during DBA.

Таблица Д.3 - Параметры окружающей среды в гермообъеме при ЗПА
Table D.3 - Ambient parameters in the containment during BDBA
<table>
<tr><th rowspan="2">Parameter name</th><th>Value</th></tr>
<tr><th>1.5 BDBA mode</th></tr>
<tr><td>1 Temperature, °C</td><td>up to 150
up to 207 (5h)
up to 250 (1h)</td></tr>
<tr><td>2 Absolute pressure, MPa,</td><td>up to 0,5</td></tr>
<tr><td>3 Relative humidity, %, max.</td><td>steam-gas mixture</td></tr>
<tr><td>4 Volumetric activity, Bq/l, max.</td><td>$5 \cdot 10^{11}$</td></tr>
<tr><td>5 Absorbed dose rate, Gy/h, max.</td><td>$2 \cdot 10^{4*}$</td></tr>
<tr><td>6 Mode existence time, h, max.</td><td>72</td></tr>
<tr><td>7 Design frequency of mode occurrence:</td><td>once in service life</td></tr>
<tr><td>8 Post-accident temperature range, °C</td><td>from 20 to 60</td></tr>
<tr><td>9 Post-accident absolute pressure range, MPa</td><td>from 0.09 to 0.12</td></tr>
<tr><td>10 Time of existence of the listed parameters after the accident, days, max.</td><td>300</td></tr>
<tr><td colspan="2">*.) Absorbed dose rate is specified with account for changes in radiation parameters during the accident and post-accident period.</td></tr>
<tr><td>пара-метров после аварии, суток, не более</td><td>300</td></tr>
<tr><td colspan="2">*) мощность поглощенной дозы приведена с учетом изменения радиационных параметров в течение аварии и послеаварийный период.</td></tr>
</table></td></tr>
</table>

T3-30	In case of non-availability of on-site mobile generators in combination with an extensive destruction of the area infrastructure are there any arrangements (e.g. heavy equipment to clear and open the roads, off-site human resources and responsibility allocation, transport of equipment from other regions in the country by air) in order to ensure transport of mobile DGs and fuel to the plant?	All engineering support measures for heavy equipment delivery, route clearance, mobilization of human resources, etc. are provided by the plan for mobilization of human resources for mitigation of an emergency situation within the external emergency plan.
T3-31	Is there any interconnection between the units allowing mutual help in case of emergency?	Two options of mutual help between the Units in case of emergency are specified below. 10 kV double-section switchgear (10 kV reliable power supply sections BDA, BDC) is provided for power supply to consumers of the normal operation reliable power supply system. Reliable power supply sections BDA, BDC are connected through cable lines to 10 kV normal operation sections BBA and BBC respectively. In order to provide power supply to the consumers in case of loss of voltage in both sections from the normal operation sections, 6300 kW self-contained DG set XKA50 (assembly BDS) is provided, which can be connected to any of the sections or to both sections BDA and BDC simultaneously. The load is accepted by the DG stepwise by automatic action. A cable jumper is provided between power supply assemblies BDS of the two Units for power supply to an assigned load in case of start failure of DG of the normal operation reliable power supply system for one of the Units. The cable jumper is connected manually. For response to an emergency at the NPP related to complete loss of power supply to the auxiliary sections, including loss of power supply of EPSS from DGs, emergency standby auxiliary transformer 00BCT30 with 110/10 kV voltage and 16 MV·A power is provided at the Belarusian NPP. The emergency standby auxiliary transformer is connected through 110 kV cable line to the external power source - Viliya 110/10 kV substation. 10 kV e To involve operating, maintenance, administrative and technical personnel of the plant for additional help, NPP shift supervisor is entitled and able to call such personnel to the site using prompt delivery ways. The emergency response su Further decisions on involving the required personnel and equipment in the works at site are made by the Commission for Emergency Situations.
T3-32	Has each unit its own emergency (crisis) centre, or there is a common centre for both units? Is the space and equipment sufficient and appropriate for management parallel accidents on both units?7.1.4	There is a common crisis center for both Units. For accommodation of officials from the management body (Commission for Emergency Situations, emergency response team) in the shielded emergency response control posts (NPP & T (town) SERCPs), 50 working places are established in the Belarusian NPP area and 50 working places in the town of Ostrovets. NPP & T SERCPs are on-site and off-site crisis centers of the Belarusian NPP. The crisis (emergency) centers are stationary information & control centers of anti-emergency planning and emergency response, which have system-based and organizational links with each other. In terms of equipment, NPP & T SERCPs are identical to each other.
T3-33	Can you please provide some more information about the difference between "instructions for accident mitigation" and "guidelines on management of beyond design basis and severe accidents"?	Information on emergency management procedures and guidelines is provided in i.7.1.1. of the National Report.
T3-34	Development of an accident management program is a rather complicated and demanding task with significant resources required. Will SAMGs for the Belarusian NPP be developed taking advantage of the experience from other NPPs or available generic SAMGs? What will be the role and use of PSA (in particular level 2) in the process of EOPs and SAMGs development?	SAMGs for the Belarusian NPP will be developed taking advantage of the experience in developing similar documents for other NPPs constructed on the basis of the Russian design both in Russia and abroad. Technical Assignment for SAMG development states that it is necessary to use the existing typical SAMG for VVER-1000 Unit with account for the design features of the Belarusian NPP Units. For development of analytical substantiation for BDBAMG/SAMG, PSA-2 results are taken into account.
T3-35	Debris removal is stated to be carried out by "available means". Please clarify what equipment is available for debris removal.	For removing destroyed civil structures and debris to clear passages and emergency exits, emergency teams use available means at hand (spades, crowbars, hand winches, cutting equipment, welding equipment, etc.), as well as machines: cranes, scrapers, bulldozers, trucks, bucket loaders, hammer drills, etc. specified in the plan for mobilization of resources and manpower for mitigation of an emergency situation within the external emergency plan.
T3-36	What instrumentation is provided to monitor operation of the POSV?	The design provides for instrumentation for the pressurizer POSV. If this instrumentation is inoperable under BDBA conditions, the POSV operation can be monitored based on pressure in the primary circuit and in the containment.
T3-37	Has the total time before core uncover been assessed in different accident sequences-scenarios analysis?	Results of the performed analysis, as well as time allowance prior to heating are given in sections 5 and 6 of the National Report.
T3-38	According to the report, a single criterion, i.e. a core exit temperature equal to 650 ^{o}C, is provided for transition to SAMG. It is also mentioned that this criterion is based on preliminary results. Can you please specify what analysis or other technical basis this value is based on, and what further analysis is envisaged to finalize the criterion? What is the degree of the destruction of the core at this temperature (see also point 3 at top of p.124 of the report)?	This criterion was taken from reference project NPP-2006 (LNPP-2) and Novovoronezh-2 NPP. This value is equivalent to a temperature of 1200 oC (maximum temperature of fuel element cladding) for most of the scenarios, which means accident transfer to the severe stage according to the Russian regulatory documents. A particular value for starting severe accident management actions is specified during development and calculation substantiation of BDBAMG and SAMG. Preliminarily, the temperature of accident transfer to the severe stage is assumed as 650 C. At FSAR stage the variants calculations will be performed. Based on their results the final temperature value will be determined, starting from which water supply to the core is prohibited. This temperature characterizes accident transfer to the severe stage. In these calculations the temperature of water supply to the core will vary. The water supply temperature selected on the basis of these calculations shall guarantee: recovery of the primary circuit mass, stable cooling of the core (absence of local hot spots), as well as restriction of hydrogen release to the leakage (restriction of its generation in the course of the zirconium-steam reaction) and prevention of formation of explosive hydrogen-steam-air mixture concentrations in the co
T3-39	With regards to the core catcher it is stated that water is supplied to the surface of the molten material by passive methods after inversion of the molten material. Please explain how this is achieved.	When the thermal protection of the vessel flange is heated up to 650 C, a temperature-sensitive element (fuse-based) in the water supply valve is actuated, and water is supplied from the molten core catcher shaft. Water supply timing is substantiated by calculations. Apart from that, water can be supplied to the surface of the molten core material from the reactor internals inspection shaft located above the molten core catcher. During BDBA with reactor core melting and molten core material release outside the reactor vessel, the system of emergency water supply from the reactor internals inspection shaft performs the following functions: - fills the molten core catcher with water from the sump tanks or from elevation 0.00 depending on an accident development scenario; - supplies borated water from the reactor internals inspection shaft to the molten core catcher vessel. To perform the above functions, manual control of the system electrically-driven valves from the BDBA segmented panel located in the MCR is provided. Valves of system JNB90 are controlled as follows:

T3-40	With regards to the Containment PHRS it is noted that water reserves in the emergency heat removal tanks are designed for a period of 24 hours after the beginning of the accident. At the end of this period it is necessary to take measures to replenish the water reserves from sources located outside of the containment – LCU tanks. Please explain how water is supplied to the emergency heat removal tanks. Is it be possible to fill the emergency heat removal tanks with a fire engine pump?	For EHRT make-up, pump JNB50 powered from BDBA power supply channel 7 is provided in system JNB90. The EHRT can also be filled using a fire engine pump through a special tie-in connection.
T3-41	Can you please provide some information regarding the automatic equipment response as well as the actions required to be taken by the personnel (e.g. open valves from control rooms panel or manually) in order to prevent a core melt under high pressure. What instrumentation is dedicated to monitor relevant parameters and how a reliable monitoring is ensured?	Within the SAR scope: Reactor vessel destruction at high pressure in case of accidents with large leakages is excluded due to fast pressure release in the primary circuit caused by significant coolant flow through the leakage. Reactor destruction at high pressure in case of small leakages from the primary circuit is excluded by opening valves of the emergency gas removal system and the pressurizer POSV upon the signal of exceeding a temperature of 400ºC (to be specified) above the reactor core. OKB Gidropress will perform the substantiating calculation. The detailed information on actions to reduce pressure and their efficiency will be presented in the report to be developed by June 2018. As for monitoring - the report will also be developed by June 2018.
T3-42	How the signal “Threat of a severe accident” is formed, based on which signals? Can you please provide more details about what conditions could result to the signal “Threat of a severe accident” leading the operator to disable the SG PHRS?	The condition for generation of the signal "Threat of a severe accident" is absence of flow in the water supply lines from the HP ECCS and LP ECCS upon reaching of the conditions in the primary circuit requiring their activation at the RP power operation. The information will be provided in SAMG in a symptom-oriented format.
T3-43	It is noted that hydrogen removal is completely passive with the location and number of recombiners being determined based on design analyses. Please explain what these analyses consisted of and what margins were included in terms of the number of recombiners, including some more information about the two subsystems for hydrogen removal from the containment in Table 2.3.3.1. Is potential H2 accumulations in other buildings than containment taken into account in development of the plant’s severe accident management program? How the process to control hydrogen by inerting the containment (sprinkler, POSV and disabling SG PHRS) is analysed and assessed? Are negative effects of such actions (e.g. cause or speed up the core melting or additional hydrogen production if cooling is decreased) considered? At which stage of the licensing procedure the measures and guidelines for controlling hydrogen and the relevant arrangements, including training of the personnel, will be assessed and validated?	In the calculations, the recombiners capacity reduction by 10% due to poisoning was simulated. Hydrogen accumulation in other buildings due to hydrogen from the primary circuit at severe accidents was not considered because such process does not occur. The negative effect of the containment inertizing with steam is acceleration of accident transfer to the severe stage. At present this algorithm is modified to mitigate the effect of acceleration of the accident heavy stage onset. The modification is scheduled to be completed in autumn 2018 (issue of FSAR).
T3-44	Can more detailed justification be provided on the prohibition for water supply to the core when the onset of a severe accident is diagnosed?	Water supply to the primary circuit is an efficient measure to stop development of a severe accident. This measure is provided in SAMG for VVER. However, late and insufficient water injection to the primary circuit can cause negative effects capable of aggravating the situation. Decision on application of such measure shall be made at the NPP based on realistic assessment of accident development.
T3-45	To prevent ex-vessel steam explosions it is noted that no water is allowed inside the core catcher and that this is ensured by the design of the safety membrane on the core catcher. Please explain how this membrane works.	The metal membrane prevents water from entering the catcher vessel until molten core material release from reactor vessel.
T3-46	If understood correctly, the containment PHRS can operate for 24 hours without replenishing of the heat removal tanks. Is there analysis available for estimating the time before the containment reaches its design loads limits, if PHRS is not available? Can you please provide whether there are alternative means considered for cooling the containment, in case of loss of containment PHRS operability?	No alternatives. It is guaranteed that transfer to the controlled state will be performed within 24 hours. Transfer to a safe state is a process specified in SAMG. The containment PHRS has 4 independent channels, operates based on natural circulation (i.e. in a passive way), the valve is always in open position. Probability of the containment PHRS failure is next to none.
T3-47	Can you please clarify whether the molten core can be sufficiently cooled in case inversion does not take place?	No effect on cooling. Has effect on water supply to the molten core material. If there is no inversion, much hydrogen is released, and steam explosions are possible. But inversion is a natural physical process. It will occur. Such a phenomenon is unlikely. Inversion of the molten core material can be impeded only by formation of a solid oxide crust between the oxide and metal components of the molten core material. As this crust will be heated from the oxide component side, it must melt. If it remains for some reasons, it represents an extra thermal resistance during heat transfer from the lower layer to the upper one, which will impede increase of heat flow in lateral direction in the metal layer (“knife effect”). It should be noted that neither calculations nor experiments proved a possibility of crust formation to impede the inversion.
T3-48	Are there alternative means (e.g. mobile) considered (or it is planned to be considered) in the severe accident management program to supply water to the core catcher in case the designed systems are unavailable?	Approach to severe accident management implies use of any available means.
T3-49	In order to understand the arrangements for preventing containment destruction due to melting of the foundation it would be helpful if a clear diagram of the core catcher could be provided. Please provide such a diagram.	The diagram of the molten core catcher is provided in chapter 12 of SAR. If necessary, this information can be submitted to PRT experts for review within the period from 12.03.2018 to 16.03.2018.
T3-50	Have BDBA I&C and the electrical equipment of the BDBA power supply located at elevation -7.20 been assessed and qualified against extreme hazards (e.g flooding, extreme weather) and severe accident environment?	In case of flooding, the EPSS DGs and a part of the safety systems located above elevation 0.00 and ensuring heat removal from the reactor plant and the spent fuel pool remain in operation. The information on them is given in section 4.2.1. The conclusion on the NPP resistance to extreme weather impacts is given in section 5.1.3 of the National Report.
T3-51	It is noted that measurements from instrumentation characterising containment integrity is displayed on panel CWL01 in the MCR but that there is no equivalent panel in the ECR. Please explain why it is not necessary to have such information in the ECR.	See the answer to T2-91.

T3-52	Can you please provide some more details regarding the assessment of the degree of core damage from the gamma radiation dose in the containment? Are there relevant computational aids developed? What are the number, type and location of radiation monitors used for assessment of containment integrity?	Results of the deterministic analyses of DBA and BDBA provided in of SAR for the Belarusian NPP (including calculations of activity of the radionuclides entering the containment atmosphere in case of fuel damage caused by accidents) allow to establish correlation between a degree of fuel damage in the reactor core and a dose rate level in the containment. In i.7.3.7 it is said that four ARMS sensors are provided inside the containment, which are designed for emergency conditions and allow for measuring a dose rate in the containment at all accidents, including severe ones. Two ARMS sensors are located in the central hall of the reactor building, and two sensors - in the containment lower part. To assess the degree of the reactor core damage two measuring channels (MC) of the ARMS are provided. The measuring channels are detecting devices UDMG-206 installed in room UJA00R120 (el. 0.0, in close proximity to the reactor cavity) connected to the ARMS top level and provided with reliable power supply (channel of the BDBA reliable power supply with a stand-by diesel generator providing power supply of the BDBA equipment within 72 hours). Measuring results of both measuring channels are displayed at the MCP (BDBA panel) and used to control the system of emergency water use from the reactor internals inspection shaft (JNB90) as follows: 1) When temperature of steam-gas mixture at the reactor core outlet reaches 400 °C, indicating beginning of the core melting, an operator opens valves on the pipelines connecting the room of molten core catcher with sump tanks JNK10,40B 2) When exposure dose rate (EDR) in the containment reaches 100 Gy / h (under melting of about 5% fuel) an operator actuates pump JNB91AP001 and opens valves on the line supplying 42% caustic soda (NaOH) solution to the sump tank 3) When temperature in the space under the reactor reaches 1000 °C the alarm on molten core material release is activated on the BDBA panel. In 30 min. an operator opens valves on the water supply line from the internals inspection cavity 4) At the stage of further molten core cooldown it is possible to fill the internals inspection cavity by the pump of the sprinkler system. Characteristics of the UDMG-206 detecting device: measuring range of photon- and gamma absorbed dose rate: 0.001 – 100000 Gy/h; gamma-quantum energy range: 0.06 – 7.00 MeV; energy dependence of sensitivity within the specified energy range: From -25 to +50%; maximum permissible error within the entire measuring range: max. ±40% at confidence probabili What are the number, type and arrangement of the radiation monitoring devices used for assessment of the containment integrity? Under normal operation vacuum pressure is maintained inside the containment compared to the annulus, therefore, in case of loss of the containment integrity the air flow into the annulus does not occur. Under pressure increase inside the containment up to 0.129 MPa and above two of four channels of system KLC11/21/31/41 are automatically put in operation. The system provides air exhaust from the annulus, air purification from radioacti System KLC11/21/31/41 performs its functions under design-basis accidents connected with pressure increase inside the containment and under beyond design-basis accidents and during post-accident measures. Air purification efficiency of system KLC11/21/31/41: aerosols: min. 99.99% (coarse purification of aerosols with diameter up to 2.0 μm, fine purification for the most penetrating aerosol particles with diameter from 0.3 μm); molecular iodine: min. 99,9%; organic iodine compound (methyl io System KLC11/21/31/41 is put in operation under pressure increase inside the containment up to 0.129 MPa and is disconnected by an operator from the MCP or ECP. The ARMS includes 4 measuring channels of IRG volumetric activity of, respectively, 4 channels of system KLC11/21/31/41 (i.e. one ARMS MC for one channel of system KLC11/21/31/41). Since system KLC11/21/31/41 intakes air from ge The ARMS MC are the UDMG-206 detecting devices receiving air after purification plants KLC11/21/31/41 due to operation of 4 RM compressors powered, respectively, from 4 channels of the emergency power supply system. The containment integrity can be assessed following the readings of the mentioned 4 ARMS MC. At present the General Designer is developing the list of ARMS preventive and emergency thresholds. When the ARMS MC readings reach the Characteristics of the UDMG-203 detecting device: measuring range of volumetric activity of beta-emitting IGR: 85Kr – 3,7•106 – 3,7•1016 Бк/м3, 133Xe – 3,7•105 – 3,7•1016 Bq/m3; beta-emission energy range: 0.08 – 2.00 MeB; maximum permissible error within the entire measuring range: max. ±20% at confidence probability 0.95.
T3-53	Can you please provide more details regarding how potential releases to the environment can be assessed during the course of an accident and also describe the availability of the necessary means in case of station blackout? Are there alternative/backup means for the ARMS sensors at the NPP site in case of loss of power supply to normal operation systems?	Emissions under severe accidents are determined on the basis of the ASRK data. Alternative measurements are not required since according to i.7.3.7 of the National report the ASRK is designed to operated under severe accidents is powered from the BDBA power supply system.
T3-54	Is the impact of the radiation, in case of spent fuel uncover, on the actions of personnel in the course of a severe accident taken account in the development of the severe accident management program?	Certainly it will be considered. Accidents management strategy of the project is based on performance of all required actions to transfer the power unit to the controlled state remotely from the MCP/ ECP. Possibility of the personnel long-term attendance of the MCR / ECR is justified taking into account operation of the MCR / ECR life supporting systems.
T3-55	Is capability of hydrogen mitigation system sufficient for coincidence of a severe accident taking place in the reactor and at the same time in the spent fuel pool (including availability of oxygen for recombination)?	hydrogen emissions from the reactor plant and spent fuel pool occur at different times.
T3-56	What are specific quantitative results of analysis of the accident progression and i termination for the severe accident taking place in the spent fuel pool (referred to the NPP stress test report)?	As it is specified in section 6.1.2 the operation personnel has a large margin of time after the moment of blackout to the moment of FA uncovering in the spent fuel pool. This situation will be a analysed in PSA-2. Taking into account this large margin of time the probability of personnel error to fail the spent fuel pool making-up is minimum.
T3-57	According to the report, the reactor developer has recommended measures in order to improve the seismic resistance of the plant. Is the time for the implementation of these measures defined? Is it required from the regulator to implement these improvements before a specific licensing stage? What is the impact of these improvements on the safety margins of the plant?	Results of the targeted safety reassessment of the Belarussian NPP have defined sufficiency of the existing design measures to ensure the NPP safety taking into account the Fukushima accident. NPP safety deficiency is not detected. The RP equipment and pipelines safety under seismic impacts with intensity exceeding the design values is justified. Therefore, there is no need to enhance additionally the design level of safety. Furthermore, for each of the considered extreme impacts the safety margins were defined, thus demonstrating a safe protection of the Belarusian NPP from factors typical for the Fukushima accident. In line with the ENSREG specification , when implementing stress-tests for the Belarusian NPP it was proposed to introduce potential safety enhancement measures for the considered impacts. Report 491-Pr-1975 specifies the RP components which have safety seismic margin less than 100% (do not withstand 8 points intensity as per MSK-64 scale) in comparison with the design level and general recommendations to enforce their design. These recommendations can be implemented as the need to increase the DBE level up to 8 points as per MSK-64 (PGA=0,24g) is required. As insufficiency of safety of the Belarusian NPP was not identified , it is not necessary to implement the proposed measures (following stress-tests results) before the start of commercial operation. At the same time, as set forth in i 8.2 of the

T3-58	How do you explain that differently from other field in the stress tests there are no improvements identified for the field of accident management ?	The Report on the Belarussian NPP safety reassessment (stress tests) was performed in 2016 on the basis of the design materials available at the time stress tests performing. The emergency management procedure regulating the personnel activity during accident mitigation and management was not available during stress-tests performing and could not be considered and analysed. The NPP emergency protection has been analysed on the basis of the design materials and the current revision of the safety analysis report. It has not allowed detection and formulation of the recommendations for improving the accident management activity. This work will be performed at the stage of development of the Safety Enhancement Program for the Belarusian NPP.
T3-59	The Emergency preparedness organizational (EPO) structure of Belarusian NPP shall be presented and descripted. This description shall include clarifications of how Emergency preparedness at the NPP is declared, including during weekdays and during holidays, who are responsible for that, how staff of EPO structure of Belarusian NPP are trained etc.	1. Emergency alert at the Belarusian NPP is performed according to the appendix to the "Plan of measures protecting the Belarusian NPP personnel in case of radiation accident (internal emergency plan)". The NPP shift supervisor is an official person of the Belarusian NPP authorised for information disclosure. Having received an accident or radiation-hazardous situation report the NPP shift supervisor identifies the current situation according to the specified categories of the NPP malfunctions and immediately reports the matter according to the announcing scheme given in the appendix to the "Plan of measures protecting the Belarusian NPP personnel in case of radiation accident (internal emergency plan)". Director (Deputy Director) of the Belarusian NPP receives the accident information from the NPP shift supervisor or gets acquainted with the situation in-situ (as required). If the criteria of "Emergency preparedness" / "Emergency situation" are reached the Director (Deputy Director) of the Belarusian NPP makes the decision to announce the "Emergency preparedness" / "Emergency situation" at the Belarusian NPP and gives instructions to the NPP shift supervisor to implement the "Internal emergency plan". If there is no possibility to inform the Director (Deputy Director) about the current situation the NPP shift supervisor makes the decision to announce the "Emergency prep Training of the personnel for the Belarusian NPP preparedness under emergency situations is provided according to the Resolution of Council of Ministers of the Republic of Belarus of August 23, 2001 No. 1281 "About order of training the h To provide and check the personnel preparedness to perform its duties under non-routine events of different types and levels (anticipated operational occurrences; emergency situations, accidents including BDBA) emergency response trainin a) NPP emergency response groups and crews;b) CES NPP;c) bodies and divisions of the Emergency Response Ministry (under approval);d) local executive and administrative organs (under approval);e) state administration bodies (under ap The last joint plant-wise training at the Belarusian NPP was held within the republican table-top exercises in October, 2017. In addition, the MCP operation personnel and the NPP shift supervisor pass training and proficiency maintaining at the full-scale simulator in the simulation training centre. Training of other personnel of the Belarusian NPP is performed during 2. The Belarusian NPP complex is considered as a nuclear facility. Therefore, emergency planning and response is arranged taking into account all components of the nuclear facility (the entire site).
	If there are other nuclear facilities on the site besides reactor buildings the information shall be added about that and the description of Emergency preparedness on those facilities shall be presented as well.	
T3-60	It is stated that "*Mitigation of the consequences of a severe accident (the power unit after such accidents does not return to operation) includes brining the emergency power unit to a safe state, processing a large amount of resulting liquid radioactive waste (i.e. the water of the emergency tanks of the containment for cooling the fuel), and development of a long-term project for mothballing the suffered power unit.*"	See response to comment T3-6
	There is no sufficient information on how a large amount of radioactive water will be handled and treated, which will be formed by the interaction of the make-up water with the melted fuel during management of accident (see see clause 3.3.11 of "Compilation of recommendations and suggestions Peer review of stress tests performed on European nuclear power plants").	
T3-61	It is stated that *"Design basis accidents (considered in EOP) include accidents that are initiating events for activation of the reactor protection system and/or resulting in activation of the safety systems or creating conditions for their activation."*	See response to comment G-25
	Definition of DBA is not in line with IAEA terminology. IAEA safety glossary terminology used in nuclear safety and radiation protection, 2016 revision:	
	design basis accident - a postulated accident leading to accident conditions for which a facility is designed in accordance with established design criteria and conservative methodology, and for which releases of radioactive material are kept within acceptable limits.	

T3-62	It is stated that *"During the first 24 hours following transition of an accident into a severe stage, the automated controls help the operator perform a minimum amount of actions to provide integrity of the container."* Mentioned 24 hours shall be clarified. What is the basis of this statement from the safety point of view?	Provided minimizing response within the first 24 hours is the requirement of the IAEA and EUR. In this case, this requirement is defined as unconditional for implementation when developing BDBA Management Guidelines / Severe Accident Management Guidelines (for BDBA and severe accident (SA)). According to the calculation results (section 6) - during the first 24 hours, no personnel actions are required to manage the accident in the RP or SF pool (SG PHRS, containment PHRS operates, there is sufficient water inventory in the SF pool). Then, actions to make-up the tanks of EHRS of SG /containment and SFP are required.
T3-63	It is stated that *„Members of the NPP CES shall participate in activities of the prior nominated emergency response teams to identify causes of deviation of the normal operation mode, assess the situation, forecast potential radiation consequences, and work out proposals for normalizing the situation."* It is necessary to clarify who and how will carry out the necessary restoration work (physical), and whether there are the necessary resources for this, except involvement of the special forces of the Ministry of Emergency Situations.	A non-routine emergency response team is employed to restore the damaged equipment, buildings and structures and to do other urgent works during localization of a NPP accident and mitigation of its consequences in time of peace and war. The non-routine emergency response team is an advanced readiness formation created using the personnel of the NPP departments. The non-routine emergency response team is subordinate to the General Director or Chief Engineer in the General Director's absence. The head of the non-routine emergency response team is appointed by the General Director who selects for this position the most qualified of the Deputies of the Chief Engineer. 4.4. The non-routine emergency response team is a part of the NPP accident prevention and liquidation forces. 4.5. The main tasks of the non-routine emergency response team are as follows: carrying out emergency and rescue operations; ensuring constant readiness for emergency and rescue operations; preventing emergency situations on the territories and facilities within its scope of responsibility; verifying compliance of the said facilities with the industrial safety requirements; training the management and personnel of these facilities to prevent and liquidate emergency situations; promoting information on the population and territory protection against emergency situations; participating in the programs aimed at training the personnel to act in emergencies. 4.6. To implement its tasks the non-routine emergency response team carries out the following main activities: ensuring radiation safety during the works on localization and management of emergency situations; organizing and carrying out general, radiation and engineering reconnaissance of the facility where an accident happened; analysing and making general conclusions on the basis of such reconnaissance; participating in development of tactical, organizational and technical solutions creating conditions for works of the non-routine emergency response team in the accident conditions; participating in special works on localization and management of accidents at the NPP; carrying out decontamination of the equipment, buildings, structures and platforms at the place of urgent works; organizing essential and advanced training for the members of the non-routine emergency response team, including through emergency response drills and exercises; keeping in constant readiness the structural divisions of the non-routine emergency response team to perform its duties; establishing and maintaining communication with the Emergency Response Supervisor, the NPP civil defense and other forces involved in the accident localization and management; providing medical first aid to the victims of the accident; carrying out urgent works for accident localization and mitigation on the process cycle systems; organizing works to prepare the reactor, turbine and other process equipment for repair; decontaminating buildings, structures, equipment and territory; decontaminating the personnel.
T3-64	In the section 7.1.1 it is stated: *"The containment integrity may be potentially assessed by means of the site radiation monitoring. In case the radiation background is higher than the design values , it may be concluded that the containment integrity is under threat or already damaged, thus requiring immediate measures to limit the release and spread of the radiation substances."* What are immediate measures to limit the release and spread of the radiation substances?	As a rule, the Severe Accident Management Guidelines describe measures to limit the release of the radiation substances from the containment? The typical measures include: switching on of the sprinkler system, limiting the release of radiation substances from the secondary circuit (in the event of inter-circuit leakage), closing of the primary circuit valves (limiting the release from the primary circuit into the containment), etc.
T3-65	It is stated that *„In-house and external emergency plans are interlinked regarding a timely notification of a potential or actual hazard of an accident, the volume and frequency of the transmission of the current information, as well as in coordination of actions and mutual assistance in the implementation of the activities."* The personnel of the Belarusian NPP responsible for timely notification, including during weekdays and holidays should be indicated. Does the Belarusian NPP have regulations on notification of WANO and IAEA about the events at the NPP?	Emergency alert at the Belarusian NPP is performed according to the appendix to the "Plan of measures protecting the Belarusian NPP personnel in case of radiation accident (internal emergency plan)". The NPP shift supervisor is an official person of the Belarusian NPP authorised for information disclosure. Having received an accident or radiation-hazardous situation report the NPP shift supervisor identifies the current situation according to the specified categories of the NPP malfunctions and immediately reports the matter according to the announcing scheme given in the appendix to the "Plan of measures protecting the Belarusian NPP personnel in case of radiation accident (internal emergency plan)". Director (Deputy Director) of the Belarusian NPP receives the accident information from the NPP shift supervisor or gets acquainted with the situation in-situ (as required). If the criteria of "Emergency preparedness" / "Emergency situation" are reached the Director (Deputy Director) of the Belarusian NPP makes the decision to announce the "Emergency preparedness" / "Emergency situation" at the Belarusian NPP and gives instructions to the NPP shift supervisor to implement the "Internal emergency plan". If there is no possibility to inform the Director (Deputy Director) about the current situation the NPP shift supervisor makes the decision to announce the "Emergency preparedness" / "Emergency situation" at the The Belarusian NPP concluded a cooperation agreement with the regional emergency response centre (RERC) of the WANO Moscow Center dated 01.03.2016 Registration No. G14-2016 (Moscow, Ferganskaya str., 25); this agreement The International Atomic Energy Agency is notified and informed by the Republican Centre for Emergency Management and the Special Rescue Unit of the Ministry for Emergency Situations (RC EMSRU MES) performing the function of The relevant procedures of notification are given in the "Plan of measures protecting the Belarusian NPP personnel in case of radiation accident (internal emergency plan)" and in the "Plan of protective measures in case of a radiation acc
T3-66	In the section 7.1.2 *"Capability to Use the Available Equipment"* it is mentioned, that the makeup of SG PHRS tanks and the spent fuel pool is provided by a high-pressure pump of the PHRS tank makeup system. The connection of pump and water sources (tanks of the LCU system and the sump tanks of the containment) to PHRS tanks and the spent fuel pool shall be presented in more details. The justification of reliability of this system, when both SG PHRS tanks and the spent fuel pool needs to be make-up at the same time, shall be justified.	Simultaneous make-up of the reactor plant and the spent fuel pool is not required since RP and SFP heating and melting occur in different time.
T3-67	It is stated that *„The mobile DG set is controlled and monitored directly from local control panels located on this equipment."* How the control and monitoring of mobile DG will be assured during harmful radiation condition on the NPP site. Please indicate all consumers of mobile DG, as well as the correspondence of their power capacity and reserve of diesel fuel.	The mobile DG set is controlled directly from the local control panel located on the equipment. Consumers of the mobile DG are listed in response T2-52, T-3-20. The diesel fuel reserve at the site for the mobile DG is designed for 7 days.
T3-68	It is stated that *„In order to further maintain the stable and safe state of the reactor plant, maintaining also operability of SG PHRS, it is necessary to periodically makeup tanks LCU from any sources of water available at the NPP site using an off-site mobile equipment (for example, from fire water storage tanks)."* Please describe how an additional water source connection to the existing systems of NPP during emergency will be realized? Please justify feasibility of this measure (make-up LCU tanks using an off-site mobile equipment) in case of harmful radiological conditions on the NPP site.	Diagram of the JNB50 LP pipelines (make-up line of the emergency heat removal tanks and spent fuel pool) contains the JNB50BR035, JNB50BR036 lines for connection of the off-site mobile equipment for the LCU tanks make-up. (Within documentation set BLR1.D.110.1.0UJE93.JNB50.021.DC.0001). This measure is implemented to maintain the stable and safe state of the reactor plant while maintaining the SG PHRS operation ability and to provide favourable radioactive background at the NPP site.
	In the section 7.1.2 it is stated: *„Every day 30 people go on alert duty using 6 units of basic, special and auxiliary machines"*.	The emergency response team of fire-fighting and rescue unit-2 has the following vehicles: - Fire-fighting truck on the chassis MAZ 6317X9 AC 8.0-40 (main); -Air and foam extinguishing vehicle on chassis MAZ-6317x9 AV 8,0 50 (6317) (main); - Emergency and rescue vehicle in configuration of chemical and radiation reconnaissance ERV (Mercedes) (special); - Vehicle with a ladder with a lifting height of at least 50 m, DL.CS5 CS (special), - Fire-fighting truck on the chassis MAZ 530905 AC 5, 0-50 / 4 (main); - Fire-fighting truck on the chassis MAZ

T3-69	Please provide more detailed information on this machinery concerning each type of it (basic, special and auxiliary), which shall always be ready for usage.	530905 AC 5, 0-50 / 4 (main); - Pump and hose vehicle on the chassis MAZ -6317x5 AHP-133 (6317)(special); - Fire-fighting truck on the chassis MAZ 530905 AC 5,0-50/4 (main); - Powder extinguishing truck on chassis MAZ-6317x5 AP 5000 (6317) (special), - Automotive sky lifts MAZ-6516V8-555-001 (special), - Carbon dioxide extinguishing trailer (special). Trucks, cranes, etc. refer to the auxiliary equipment.
T3-70	In the section 7.1.2 it is stated: "*Supply of water for fire-fighting purposes of NPP buildings and facilities is provided by the internal and external fire-fighting water supply systems*". Please provide information that the internal and external fire-fighting water supply systems will be able to fulfil its functions in extreme environmental conditions.	The fire-fighting water supply system includes partially the underground pump station with redundancy of the main fire-fighting pumps and power supply redundancy, embanked underground fire-fighting water storage tanks and underground pipelines given in the Belarusian NPP design documentation, section 5.2, item 5.2.1.2.3 (BLR1.B.130.&.0502&&.0101&.056.CA.0001). All structures are arranged below the level of soil freezing. The indoor fire-fighting pipeline is protected from extreme weather conditions by the protective structures of the buildings where the pipeline is routed. These solutions protects from the impact of extreme weather conditions on all components of the fire-fighting system.
T3-71	In the sub-section „*Providing resources and supply management*" it is stated that „*In terms of accident management, delivery of resources (fuel for the diesel generators, water, etc.) will be carried out within the framework of the SAMG. Relevant activities will also be provided for emergency planning.*" The information provided in this sub-section is not sufficient for appropriate evaluation of it. The results of „Stress tests" shall clearly define the needs/measures related to outsourcing for emergency management and how they will be implemented.	Availability of the logistics support is provided following the ERS's decision. The material resources of the State Civil Defence Organization of the Belarusian NPP is issued from the workshop warehouses and civil defence and emergency situation warehouses by heads of the divisions and the head of Civil Defence Office. The NPP logistics support during the rescue and other emergency actionsTo arrange logistics support of the rescue and other emergency actions (spare parts, water, food, clothes, footwear, communication, personal protective equipment , radiation and dosimetry monitoring devices) the personnel of the radiation safety division and communication shop will be involved, as required. For logistics support under emergency conditions other organizations supplying the required material and technical resources will be also involved, as required. The SERCP is provided with all required material resources and radiation and dosimetry monitoring devices for the members of the WANO Moscow Center. The list of material resources subject to storage and corresponding to the "List of material resources stored in the SERCP" is approved by the director of the Belarusian NPP. Material reservesThe emergency reserve of personal protective equipment (respiratory organs and skin) for the operation personnel is stored at the personnel workplaces. The emergency reserve of dosimetry monitoring devices is stored at the workplaces of the radiation safety division personnel. The emergency reserve of personal protective equipment and dosimetry monitoring devices for the State Civil Defence Organization is stored in the civil defence warehouse. The emergency reserve of medicines is stored in the Belarusian NPP health centre. Formation of emergency kits at the Belarusian NPP according to the relevant provisions and approved assortment provides equipment with State Civil Defence Organization outfit, personal protection means, radiation and dosimetry monitoring The emergency reserve of decontamination facilities are stored in the civil defence warehouse.
T3-72	In the section 7.1.2 "*Capability to Use the Available Equipment*" is stated: "*The buildings of NPP will use automatic modular fire-fighting systems with a finely dispersed spray, automatic gas fire-fighting systems, automatic water extinguishing systems for the main buildings and facilities of the power unit*". In which compartments the automatic gas fire-fighting systems is used and what kind of gas is used?	Gas fire-fighting units are designed to extinguish fire in the rooms with electrical equipment and computer machines according to SP5.13130.2009. Halon 125HP (C2F5H) (condensed gas) is used as a gas fire extinguishing agent. To replace gas fire extinguishing agent from the module nitrogen is used.
T3-73	In the section 7.1.2 it is stated: "*On the territory of FERU-2 there is a fire reservoir with a volume of 50 m^3 and a network of fire hydrants*". Is this fire reservoir with a volume of 50 m3 for filling fire extinguishing machines and can it be used in winter time?	The fire reservoir 50 m3 located within the area of the fire-fighting and rescue unit-2 of the Belarussian NPP is used to fill the fire trucks. According to the design solutions this fire reservoir operates in the winter period. The reservoir is installed under the ground. Water level in the reservoir is below the level of soil freezing. The reservoir is embanked according to the routine decision (TP901-4-70.83-III). The floor slab is additionally insulated with rigid mineral-wool plates of PS-200.critical function of safety when forming the corresponding conditions;. These solutions prevent from water freezing in the period of below-zero outdoor temperatures.
T3-74	It is mentioned that the distance from the fire-fighting and rescue unit building (intended for protection of the Belarusian NPP facilities) to the territory of the construction and installation base is 100 meters. It is meant that fire-fighting and rescue unit building could be affected by external hazards (e.g. earthquake) and harmful radiological conditions on the NPP site. Please justify availability of fire-fighting and rescue unit building in case of external hazards (e.g. earthquake) and ability of fire-fighting and rescue unit to perform foreseen works in case of harmful radiological conditions on the NPP site (see clause 3.3.2 and 3.3.12 of "Compilation of recommendations and suggestions Peer review of stress tests performed on European nuclear power plants").	In the territory of a fire station there is a radiation-proof shelter for 60 persons protecting the staff from ionizing radiation exposure.

T3-75	It is stated that „*The actions of operational personnel to mitigate consequences of the accident are set forth in the operational instructions and emergency procedures, including:* - *"Technological Regulations";* - *"Instructions for Mitigation of Accidents"…;* The Technological Regulations could not be directly related to emergency procedures, because the main purpose of this document is to assure normal/safety operation and define the limits and conditions of safety operation. The management of accidents and mitigation of consequences of them shall be performed according to emergency procedures. Please provide additional information who (shift operating personnel or personnel of Commission for Emergency Situations of NPP or both) will be responsible for implementation of accident management measures foreseen in EOPs, BDBAMGs and SAMGs.	The Process Regulations contain rules and methods of the power unit safe operation, general procedure of the safety-related operations, limits and conditions of safe operation and operational limits of the main parameters. The Process Regulations are applied to power unit operation under normal conditions and under "operation with deviations". In the Safe Operation Regulations there is a section describing general safety requirements for power unit control under anticipated operational occurrences including accidents. This section includes the principles of management of anticipated operational occurrences and accidents and contains instructions for the personnel to operate the power unit in this mode using the Procedure on Elimination of Anticipated Operational Occurrences or Procedure on Elimination of Design Basis Accidents. Operations manager during mitigation activities of the BDBA is the NPP shift supervisor. The NPP shift supervisor must: - supervise the accident mitigation works; - supervise the personnel and make sure that they act in compliance with the accident management guidelines; - order to transition to the guidelines on restoring the critical safety function when the required conditions are formed; - order to transition to the BDBA management guidelines when the required conditions are formed; - continue act in compliance with the emergency operation procedure once the critical safety function is restored; - report on accidents and progress of mitigation of their consequences to the management of the Belarusian NPP, inspection of the supervisory authority at the Belarusian NPP and the grid operator. The Unit shift supervisor must: - identify the cause of an accident and report to the NPP shift supervisor; - manage activities of the MCR and Unit personnel; - read out the actions from the corresponding section of the emergency operation procedure and supervise their implementation; - supervise the accident mitigation works. The turbine control lead engineer and the reactor control lead engineer monitor the state of the systems and equipment and make the required switches over following the instructions of the unit shift supervisor. In the event of a radiation pre-a A more senior administrative and technical employee is empowered to temporarily suspend from his functions an operations supervisor if his actions lead to deterioration of the emergency situation or do not comply with the requirements of the The location and actions of the shift operating personnel are determined by the duty and operational instructions and orders of the operational manager. Shifts handover before restoration of a safe state of the reactor plant or without a relevan Operations manager during mitigation activities of the BDBA is the NPP shift supervisor. The NPP shift supervisor must: - make sure that the personnel act in accordance with BDBAMG; - issue orders to Unit shift supervisors in regard to application of the respective BDBA sections and, if required, necessary additional actions; - direct the personnel performing switch-overs in the common-plant systems; - monitor the accident response progress; - monitor the state of the critical safety functions; - report to the NPP managers on accident occurrence and response progress. The Unit shift supervisor is an accident response operations manager when the NPP shift supervisor is absent in the MCR. The Unit shift supervisor must: - direct the MCR and Unit personnel's actions; - issue orders to the lead engineer for reactor control, monitor the state of the critical safety functions if the NPP shift supervisor is unable to fulfil his duties; - after a fault in the critical safety functions has been identified, read actions out of the respective BDBAMG section and supervise their fulfilment. The lead engineer for reactor control must monitor the state of the critical safety functions if the NPP shift supervisor is unable to fulfil his duties, report to the Unit shift supervisor on the necessity of application of the respective BDBAMG sect The lead engineer for reactor control and the lead engineer for turbine control fulfil orders issued by the Unit shift supervisor, report to the Unit shift supervisor on actions performed. The Unit shift operating personnel must act in accordance with orders issued by the MCR operating personnel. Within 30-40 minutes from the emergency protection actuation, Deputy Chief Engineer and a specialist from the list approved by Deputy Chief Engineer must be delivered to the MCR. The specialist must: - monitor the state of the critical safety functions; - report to the operations manager on the necessity of application of the respective BDBAMG sections; - make sure that the personnel act in accordance with BDBAMG;
T3-76	It is stated that "In an event of damage or complete destruction of MCR and ECR, the accident management activities can be carried out from the power plant's shielded emergency control post." What are design characteristics of the NPP's shielded emergency control post? May this post resist 8-point earthquake?	See T-3-68.
T3-77	In the section 7.1.3.4. it is stated: „*The senior management of the Republican Unitary Enterprise "Belarussian Nuclear Power Plant" and NPP operations shops must be staffed with a qualified and experienced personnel with high and/or secondary vocational education in the respective area and related spheres of knowledge and also with work experience in the respective area".* Please provide information what is current state of staffing?	The required number of operating personnel for commissioning of Belarusian NPP Unit 1 is 1160 persons. As of 01.01.2018 the total number of the personnel of the Belarusian NPP constitutes 761 persons (66% of the required number). 2. At present the Belarusian NPP is sufficiently (for this phase of construction) staffed with the following operating personnel: - foreign managers and specialists having higher education in the respective sphere and experience in working at NPP - 62 persons (8% of the total number); - managers and specialists of the Republic of Belarus having higher education in the respective sphere and experience in working at thermal power plants and other enterprises of the power industry – 345 persons (45% of the total number); - young specialists of the Republic of Belarus (graduates of higher education institutions) having education in the respective sphere – 175 persons (23% of the total number); - young specialists of the Republic of Belarus (graduates of specialized secondary schools) having education in the respective sphere – 25 persons (3% of the total number); - other specialists and workers of the Republic of Belarus employed as independent contractors – 154 persons (21% of the total number).
T3-78	It is stated that „*NPP SERCP is arranged in a separate standing shelter (coordinates 01UYX at the NPP plan: 11A, 6B) with a capacity of 100 people. It is designed to manage the NPP divisions and all forces involved in emergency activities during localization and mitigation of the consequences of emergency."* What is a structure of NPP SERCP and T SERCP and communication ways between those structures?	The shielded emergency response control post (NPP SERCP) is located in a separately standing shelter (01UYX, coordinates at the NPP plan: 11A, 6B) with a capacity of 100 persons, number of operating personnel is 80 persons. SERCP is designed for management of the NPP structural units and forces involved in emergency response activities at the NPP during localization and mitigation of the emergency consequences, interaction with Ministry of Energy of the Republic of Belarus, with local self-government bodies, regional authorities for emergency situations. Communication between NPP SERCP and Town SERCP is provided through fibre-optic communication lines within the external communication system of the Belarusian NPP. In order to improve reliability of the network, a ring topology is provided, with looping of the ring through fibre-optic lines of Beltelecom communication operator. Also, communication between NPP SERCP and Town SERCP is provided through the operative radio communication system (CYS) as per TETRA standard. In addition, see T-3-28.

T3-79	In the section 7.1.3.6 it is written "*in case of infrastructure disruption, the Belarusian NPP site is self-contained - the residual heat is removed by BRU-A or SG PHRS for at least 72 hours*". However, considering single failure of one channel of passive system (if 3 out of the 4 emergency heat removal tanks are used, the self-sufficient operation reduces from 72 to 24 hours). Only 24 hours period of self-sufficient operation shall be assumed for the passive systems in the analysis.	As demonstrated in section 6.1.2 , a simultaneous functioning of 3 channels of PHRS from 4 emergency heat removal tanks, the PHRS ensures a safe state of the reactor plant during 72 hours without violation of the acceptance criteria. At a later stage, when the routine EHRT makeup from desalinated water tanks (LCU) starts working, SG PHRS maintains a safe cooldown state of the reactor plant during 165 hours since the start of accident process. Overlapping of additional failures is considered within PSA and is not subject to consideration within stress tests.
T3-80	In the section 7.1.3.8 it is written "*According to the design, each power unit of the Belarussian NPP is self-contained, that also means that one power unit does not influence another one*". However, for example severe accident in one power unit and increased radiation doses will significantly complicate the operation of remaining reactor. The protective measures for operators in case of severe accidents affecting multiple units and other technical and organizational measures shall be foreseen (see clause 3.3.4 of "Compilation of recommendations and suggestions Peer review of stress tests performed on European nuclear power plants").	According the design all the required actions for shutdown, cooldown and transfer of Unit to a controlled and safe state are performed remotely from the MCR/ECR. Possibility of the personnel long-term attendance of the MCR / ECR is justified taking into account operation of the MCR / ECR life supporting systems. If it is impossible for the personnel to stay in the MCR/ECR, accident management is performed from the shielded emergency centers located both on the NPP site and off-site.
T3-81	It is stated that „*As part of BDBAMG and SAMG, procedures are drafted for the power unit shutdowns (including the dismantled reactor head), and management of accidents caused by the fuel damage in SFP.*" Strategies of accident management are parts of BDBAMG and SAMG. The list of all strategies shall be presented to understand differences between BDBAMG and SAMG taking into account a critical safety function also for corresponding assessment.	Within BDBAMG the management is aimed at meeting the following safety targets: - reactivity control (ensuring subcriticality of the reactor - fast shutdown and maintaining the reactor core in a subcritical state); - heat removal from the reactor core and the primary circuit to the ultimate heat sink (RP cooldown); - integrity of the primary circuit (ensuring reliable heat removal from the reactor core during an accident as well as after stabilization of parameters in the post-accident state); - integrity of the containment (ensuring localization of accident consequences by sealing the reactor containment to minimize radiological impacts, retain radioactive products within the established boundaries and quantities); - availability of safety functions (ensuring the required inventory of operating media in the primary and secondary circuits). BDBAMG procedures are not applicable to severe accidents for the following reasons: 1. BDBAMG procedures are success-oriented. The recovery measures imply that the reactor core damage can be avoided by implementing a certain sequence of actions. Therefore, all the emergency procedures have a cycle of actions which can be terminated only if a sequence of actions turns out to be successful. 2. SAMGs are developed on the basis of thermohydraulic analysis of accidents which are not brought to a stage when the reactor core is considerably overheated. When an intensive steam-zirconium reaction begins and the reactor core geom 3. The BDBAMG procedures are specifically aimed to prevent the core damage. If the core is damaged then the control actions are aimed to prevent (mitigate) fission products emission outside the NPP. Under the conditions caused by the co The respective strategies are represented by procedures constituting BDBAMG. When a severe accident stage starts, transfer to the severe accident management strategies takes place. Such strategies are divided into three groups: strategies aimed to control release of fission products; strategies aimed to prevent damage of the containment; strategies aimed to bring the Unit to a stable controlled state. The above strategies are implemented by taking the following measures: 1. Supply water to the primary circuit; 2. Ensure subcriticality of the molten core material; 3. Reduce pressure in the primary circuit; 4. Reduce pressure in the steam generator; 5. Arrange for cooldown by heat removal through the SG PHRS; 6. Supply water to the spent fuel pool; 7. Supply water to the containment PHRS; 8. Isolate the molten core material in the molten core catcher; 9. Reduce release of fission products. Severe accident management group is the responsibility of the accident management group (AMG). After the AMG undertakes the responsibility for severe accident management, the MCR personnel follow the AMG instructions to implement severe accident management strategies and inform the AMG on operability of equipment, possibilitie
T3-82	It is stated that „*Debris removal at the evacuation pathways and the protective constructions entrances clearing are carried out by the available means and means of joint NPP rescue team, and also by the engaged forces and forces of rescue and other emergency actions.*" Please describe and present a description of the forces and rescue forces involved to assess and confirm their sufficiency.	See T-3-19, T-3-35, T-3-63, T-3-69.
T3-83	In the section 7.1.5 "*Measures to Improve Capabilities for Accident Management*" the organizational and technical measures related to improvement of accidents management are mentioned. All in the section 7.1.5 identified measures should be included in the section 8.3.4 "Possible Measures to Improve the NPP Safety in Terms of Accident Management". The guaranteed confirmation, that these measures will be implemented before the start of operation of NPP shall be presented.	Results of the targeted safety reassessment of the Belarussian NPP have defined sufficiency of the existing design measures to ensure the NPP safety taking into account the Fukushima accident. Insufficiency of safety was not identified and additional measures to enhance the design safety level are not required. Furthermore, for each of the considered extreme impacts the safety margins were defined, thus demonstrating a safe protection of the Belarusian NPP from factors typical for the Fukushima accident. In line with the ENSREG specification , when implementing stress-tests for the Belarusian NPP it was proposed to introduce potential safety enhancement measures for the considered impacts. As insufficiency of safety of the Belarusian NPP was not identified , it is not necessary to implement the proposed measures (following stress-tests results) before the start of commercial operation. At the same time, as set forth in i 8.2 of the National Report ,adhering to the principle that safety is priority and it must be constantly improved, the proposed measures plus measures following the peer-review will be incorporated into the Safety Enhancement Program of the Belarusian NPP.
	It is stated that "*In the Belarusian NPP design it is assumed that the transition of the accident to a severe phase of its development occurs when the temperature reaches + 650 ° C (according to preliminary estimates) above the core*".	The numerical value of the criterion will be substantiated at the stage of developing and substantiating symptom-oriented BDBAMG/SAMGs. This criterion definitely characterises accident transition to a severe phase (fuel melting process begins, which is characterized by exceeding the maximum limit of fuel elements damage). At present this value is used as a criterion for introducing SAMG (starting the severe accident management actions) for reference project LNPP-2. In the course of BDBA, in case of failure of all active core cooling systems, a threat of accident transition to a severe phase appears. In this case, up to a certain moment (state of the reactor core), if make-up of the primary circuit is restored, it is possible to prevent the core melting. However, when water is supplied to the overheated zone, significant aggravation of the situation is possible mainly due to generation of hydrogen caused by the zirconium-steam reaction and, as a result, possible loss of the last safety barrier - the containment - in case of subsequent detonation of the hydrogen-containing mixture after its release to the containment through a leakage.

<table>
<tr><td>T3-84</td><td>It is not clear what physical meaning of this value is. To what temperature of fuel and vessel internals it corresponds? How such limit was defined? Why estimation is preliminary? The physical meaning of limit "+ 650 ° C" shall be provided.</td><td>At FSAR stage the variants calculations will be performed. Based on their results the final temperature value will be determined, starting from which water supply to the core is prohibited. (See T3-38).

From the point of the core heating and melting development, a temperature of + 650 ° C above the core also corresponds to a fuel element cladding temperature of 1200 C, i.e. the temperature at which the zirconium-steam reaction becomes</td></tr>
<tr><td>T3-85</td><td>It is stated that „The Belarusian NPP design considers measures for managing beyond-the-design-basis accidents."

Please submit the list of the Belarusian NPP BDBA and the way to determine / select / classify them.</td><td>List of scenarios of severe accidents for the deterministic analysis
<table>
<tr><th>No.</th><th>Accident scenario</th><th>Objectives of deterministic analysis</th></tr>
<tr><td>1</td><td>Double-ended break of the reactor coolant circuit (DN 850) with a failure of the active emergency core cooling systems</td><td>Substantiation of the hydrogen removal system, the molten core catcher, the containment passive heat removal system. Analysis is carried out up to the moment when the controlled state is achieved</td></tr>
<tr><td>2</td><td>Rupture of the pressurizer surge line (DN 346) with a failure of the active emergency core cooling systems</td><td>Substantiation of the hydrogen removal system, substantiation of the radiation safety. Analysis is carried out up to the moment when the controlled and safe state is achieved. Substantiation of compliance with the radiation criteria.</td></tr>
<tr><td>3</td><td>Leakage from the reactor collecting chamber (DN 279 mm) in the event of a rupture of the supplying pipe of the emergency core cooling system hydro accumulators with a failure of the active emergency core cooling systems</td><td>Substantiation of the hydrogen removal system. Analysis is carried out up to the moment when the controlled state is achieved</td></tr>
<tr><td>4</td><td>The results of the analysis of a severe accident: Leakage from the cold leg (DN 179) of the pressurizer loop with a failure of the active emergency core cooling systems</td><td>Substantiation of the hydrogen removal system. Analysis is carried out up to the moment when the controlled state is achieved</td></tr>
<tr><td>5</td><td>Leakage from the cold leg (DN 80) with a failure of the active emergency core cooling systems</td><td>Substantiation of the hydrogen removal system. Analysis is carried out up to the moment when the controlled state is achieved</td></tr>
</table></td></tr>
<tr><td>T3-86</td><td>It is stated that „The application of passive safety systems in the design increases the NPP reliability, because only uncompensated leakages of the primary circuit as the initiating events can lead to the accidents with core damage. Failed heat removal from the secondary circuit in case of SG PHRS failure is a rather unlikely event that reduces the probability of an accident by 3 orders of magnitude."

In accordance with EU "Stress Tests" Specification, deterministic approach should be used for assessment of safety systems. The postulated initiating event leading to SG PHRS failure shall be analyzed and possible consequences of them evaluated.</td><td>This work is in progress. It is scheduled to be completed in June 2018.</td></tr>
<tr><td>T3-87</td><td>„The symptom-oriented procedures are applied after actuation of the reactor emergency protection and\or safety systems or appearance of conditions for their actuation and before failure of the critical safety functions, but only in the following cases:

- the operating personnel failed to determine which event-oriented procedure shall be applied;

- overlapping of initiating events occurred and the operating personnel failed to determine which event-oriented procedure shall be applied first of all;

- application of the event procedures does not lead to expected results".

The main task of the operation personnel is the monitoring of the critical safety functions when acting according to the symptom-oriented procedures. Additional emergency procedures (event-oriented procedures) shall be used together with symptom-oriented procedures in the case of strategies implementation to prevent an accident.

According to presented information operation personnel at the Belarusian NPP first of all must act according to event-oriented procedure and in case they fail to determine which event-oriented procedure shall be applied then act according to SOPs. For the critical safety function management SOP's shall have priority over EOP's. Please justify why this approach will be used.</td><td>For accident management the personnel shall follow the validated procedures. International standards (IAEA) recommend to apply symptom-oriented approach for BDBA and SA.
The personnel apply event-oriented emergency procedures for response to simple, easily identifiable accidents considered in the design. The personnel cease to follow event-oriented emergency procedures and start to apply symptom-oriented procedures in the following cases:
- the operating personnel failed to determine which event-oriented procedure shall be applied;
- overlapping of initiating events occurred and the operating personnel failed to determine which event-oriented procedure shall be applied first of all;
- application of event-oriented procedures does not lead to expected results.
Simultaneous application of both types of procedures is excluded.</td></tr>
<tr><td></td><td>"Figure 7.1.1.1. - Diagram of the operational subordination of the duty personnel of the shift of the Belarusian NPP"</td><td>The presented structure of the operational subordination is schematic. Below is the operational subordination of the Unit MCR personnel represented by: Unit shift supervisor, reactor shop shift supervisor, lead engineer for reactor control, turbine shop shift supervisor, lead engineer for turbine control. Reactor shop shift supervisor, lead engineer for reactor control, turbine shop shift supervisor, lead engineer for turbine control are directly subordinate to Unit shift supervisor. Unit shift supervisor, in his turn, is operationally subordinate to NPP shift supervisor.</td></tr>
</table>

T3-88	According to the presented diagram the MCR personnel is not directly operationally subordinated to the Head of the shift of the NPP unit. For example, the personnel of the reactor shop or turbine shop has own tasks and this could to lead to conflict in teamwork during emergency. Please justify why this operational subordination approach will be used.	
T3-89	It is stated that „*Further actions after transfer of the reactor plant to a stable safe state are determined by a separate decision of the authorized bodies based on the results of the investigation of the accident causes and consequences.*“ The duties and the names of the authorized bodies should be explained. The meaning of "stable safe state" for SAMG's and for BDBAMG shall be explained.	An authorized body making decisions in this situation is the Commission for investigation of accident causes and consequences which is established on the basis of the Provisions on registration and investigation of deviations in NPP operation. The Commission includes representatives of ministries, departments and government institutions, whose rights and authorities are defined by the laws, including Law on Nuclear Energy Use. In order to develop measures for bringing RP to a stable state, design, engineering, research & development and other organizations are involved if required. A stable safe state for BDBAMG is defined by meeting the following safety targets: - ensuring subcriticality of the reactor - fast shutdown and maintaining the reactor core in a subcritical state; - cooldown of the RP; - ensuring reliable hear removal from the core during an accident as well as after stabilization of parameters in the post-accident state; - ensuring localization of accident consequences by sealing the reactor containment to minimize radiological impacts, retain radioactive products within the established boundaries and quantities; - ensuring the required inventory of operating media in the primary and secondary circuits. A stable safe state for SAMG, when application of SAMG may be ceased, is defined by reaching a state where all threats to the containment as a barrier on the way of fission products are eliminated, all releases of fission products are under c
T3-90	It is stated that „*To mitigate the BDBA consequences, the following organizational and technical measures are suggested:* *- assessment of the documentation on personnel actions in case of development of emergency situations at earthquakes, seismic impact exceeding the design value;* *- the documentation on personnel actions shall include sections providing for measures to diagnose the NPP state, restore the normal operation conditions, failed safety functions and prevent or limit the effects of the core damage: Process Regulations, Reactor Plant Emergency Operation Procedure, BDBA Management Guidelines, Severe Accident Management Guidelines, as well as Action Plan for Personnel Protection in Case of an Accident, which will contain sections providing for measures to solve the following tasks:* *- to monitor operation of the safety function algorithms ;…*“ Safety systems have algorithms to perform their functions in accordance with the project of NPP. Safety function could not have algorithms. If here the systems of monitoring of safety functions algorithms are meant, then additional explanation should be presented.	We suggest changing it to "the personnel monitor operation of systems and equipment according to BDBAMG/SAMG. The personnel check fulfilment of the tasks specified for systems according to BDBA/SAMG diagnostic schemes". We suggest changing "- to monitor operation of the safety function algorithms;…" to: "actions of the algorithms implemented in SS (safety system) I&C to perform safety functions"
T3-91	It is stated that „*-to achieve the final state where the fission chain reaction is discontinued, the reactor subcriticality is ensured and the core re-criticality is prevented, with account for its possible damage;* *- to prevent (mitigate) severe damage of the fuel by both automatic actions of the systems and control actions of the personnel;*“ Please specify the measures related to spent fuel pools, especially taking into account situations on maintenance works during removal of spent fuel from the core to the spent fuel pools.	Subcriticality in the spent fuel pool is ensured by using a liquid absorber (boric acid solution 16 g/dm3), by the design of the racks and spent fuel spatial arrangement, as well as by heat removal from the spent fuel pool. See the answer to T3-56
	It is stated that „*To manage severe accidents, the design provides for a set of technical and organizational measures aimed at transferring the NPP to a controlled state. The means applied are, as far as possible, independent of the means applied at levels 1-3 of the defense-in-depth .*" 1. Please elaborate what particular technical and organizational measures aimed at transferring the NPP to a controlled state are provided in the design.	B-491 design provides for sufficient quantity of safety systems and accident management means. The design complies with the entire scope of the requirements in the Russian Federal norms and regulations. The calculations show that available quantity of the means is sufficient for management of BDBA (at DiD level 3) and SA (at DiD level 4). Particular measures for accident management are specified in the answer to T3-106. In addition, in case of a severe accident, the following design technical measures are applied to bring Unit to a controlled state: the molten core catcher, the system of water supply to the molten core catcher, the system of NaOH solution supply to the emergency sump, the hydrogen removal system, including the hydrogen concentration monitoring system, the containment PHRS. As for the WENRA recommendations quoted in the question, we would like to point out that when these documents were developed the Belarusian NPP was already under construction. Meanwhile, the WENRA documents in question make recommendations to the newly designed NPPs.

T3-92	2.The widely known sad experience with managing of severe accidents at Fukushima Nuclear Power Plant has shown that the defense-in-depth level 4 events must be managed by safety features that are independent of safety systems designed to manage the defense-in-depth level 3 events. Independence of the defense-in-depth levels is emphasized in WENRA and IAEA requirements for new reactors (2014-2017). Belarus participates in the WENRA activities as an observer and certainly shall be familiar with the WENRA requirements for new reactors. Could you please clarify whether the Regulatory Authority of Belarus Republic is requiring improvements of the reactor design (to its compliance with the WENRA and IAEA requirements for new reactors) in order to minimize the consequences of severe accidents? If so, please specifically indicate which safety systems are not in compliance with the aforementioned independence principle and will be used to manage both the defense-in-depth level 3 and 4 events? 3. A description of the means applied at levels 1-3 of the defense of depth at the Belarusian NPP shall be provided.	
T3-93	It is stated that „*In the adopted concept of severe accident management the operator's actions are specified in Severe Accident Management Guidelines. Severe accidents are expected to be managed by the personnel actions. Diagnostics of the reactor plant state on the basis of which a decision will be made to proceed to the severe accident management is implemented from the MCR, the diagnostic tools are provided with reliable power supply…*" The beginning and the end of accident management in accordance with SAMG and BDBAMG shall be explained.	See the answer to T2-126.
T3-94	It is stated that „*In accordance with the adopted severe accident management strategy, the primary circuit pressure reduction to prevent the molten core material releasing beyond the reactor vessel at high pressure is performed by opening the valves of the emergency gas removal system and the pressurizer POSV by the operator. The procedure for the operator's actions to open the valves of the emergency gas removal system and the pressurizer POSV is specified in Severe Accident Management Guidelines.*" Please describe a procedure how operator performs primary circuit pressure reduction (initial conditions, safety criteria, etc.	Information on SAMG development is given in T3-1. To confirm adequacy of the facilities (emergency gas removal system, pressurizer POSV) in severe accident scenarios substantiating calculations are being performed at present. The report containing the results will be provided in 2018.
T3-95	The decrease of the primary circuit pressure is ensured in the event of severe accidents with the safety valves of the pressurizer and the emergency gas removal system. The systems designed for managing severe accidents shall be independent of the systems that are designed for the operational conditions and postulated accidents of the plant. How such independence will be ensured?	Opening of the emergency gas removal system valves and pressurizer POSV upon the signal of exceeding a temperature of 400 °C above the core (See Chapter 15 of SAR) by the operator is provided to prevent the reactor destruction at high pressure with small leakages only. Reactor vessel destruction at high pressure in case of accidents with large leakages is excluded due to fast pressure release in the primary circuit caused by significant coolant flow through the leakage. As indicated in i.7.2.1, any available serviceable technical means intended both for ensuring safety at design basis accidents and for normal operation are applied for BDBA management. The requirements of the Russian Federation and the Republic of Belarus, fulfilment of which was mandatory for the project implementation, do not impose restrictions on application of these systems for severe accident management.
T3-96	In the section 7.2.3 the design substantiation of the molten core catcher operational efficiency is performed for the case of severe beyond design basis accidents using an example of DN850 leakage. However, there is no justification that such accident will lead to the most severe consequences to the containment. Evidence that selected for the analysis accident is the most dangerous from the safety point of view shall be presented.	From the whole list of considered BDBAs, in case of an accident with DN850 pipeline rupture the operator has the shortest time to take exhaustive measures for prevention of the accident transition to a severe phase with the molten core material release from the reactor vessel. For this reason it is said in i.7.2.3 of the National Report that this accident was used to prove efficiency of the molten core catcher operation. Impact of this accident on the containment is not considered in i.7.2.3.
T3-97	It is stated that „*The operating organization submitted to Gosatomnadzor the results of the design analysis, from which it follows that the molten core catcher is able to perform its design functions, namely:* - *water is supplied to the surface of the molten material by passive methods after the inversion of molten materials.* " The measures shall be explained and described when these measures will be implemented for this operation.	This operation is performed without participation of the operator. The system is based on the passive operation principle. Under severe accident conditions with destruction of the reactor vessel when the molten core catcher is filled with water an air lock is formed in the upper part of the valve for water supply to the molten core material due to the vertical portion. The air lock allows for heating of the plug solder at a specified rate by the heat from the melt mirror side (meanwhile, water level in the shaft is higher than the valve location). When a solder temperature close to its melting point is reached, the plug falls off and water starts flowing from the catcher shaft through the valve to the melt mirror. Thus, the water supply valve performs its design function - water supply to the melt mirror. This valve is a passive water supply device. In total the system includes 8 valves. Type of solder and valve installation depth are selected so as to provide water supply to the melt mirror after inversion of the melt, i.e. upfloating of the molten core fuel-containing fractions over the metal layer.
T3-98	It is stated that „*For fixation of radioactive iodine isotopes and reduction of radioactive release from the containment, injection of alkali solution into the sump tanks of the containment is provided. Alkali supply is implemented by the operator's actions.*" What tools/actions are used for this operation?	When gamma-radiation dose rate in the containment atmosphere becomes equal to or over 100 Hy/h (this dose means melting of about 5% of the fuel in the core), the operator starts pump JNB91AP001, opens the valves (JNB91AA801, JNB91AA802), and after that 15m3 of caustic soda solution (42% $NaOH$) is supplied for 60 minutes to the sump tanks (JNK10,40BB001). Technical characteristics of system JNB91: - capacity of tank JNB91BB001 – 16 m3; - rated throughput of pump JNB91AP001 – 15 t/h; - head at rated flow rate – 86 m of water column; - power – 15 kW. In order to obtain data on fluid availability in the emergency caustic soda supply pipelines, four level alarms are provided in the design. One is installed on the pipeline in building UKC, the second one in building UKD, the third and the forth - in building UJA near sump tanks JNK10,40BB001 respectively. Process flow diagram of the emergency caustic soda supply to the sump tanks is shown in Fig.12.1.14.1.1 in SAR Chapter 12. Diagram of electrical connections for power supply to the consumers used during BDBA is shown in working documentation set BLR1.D.110.1.0UJE&&.&&&&&.031.DC.0001. In case of the Unit blackout, the emergency caustic soda supply pump (JNB91AP001) and the valves (JNB91AA101, JNB91AA801, JNB91AA802) are powered from a mobile 500 kW diesel generator. Electrically-driven components of systen

T3-99	It is stated that *„Formation of explosive concentration of hydrogen-steam-air mixture is prevented by operation of the JMT system. The hydrogen removal system (JMT) is completely passive, and the autocatalytic recombiners included in the system do not require electric power. Another measure to prevent formation of explosive concentration of hydrogen-steam-air mixture is inertization of the atmosphere with steam (steam concentration increase in the containment)"* What are design capabilities of JMT system, especially if additional inertization of steam to the containment is needed.	The design provides for 44 autocatalytic hydrogen recombiners of RVK-3 type - 16 pcs. And RVK-4 type - 28 pcs. Depending on location and capacity, the recombiners are combined into eight process groups (see table 12.2.3.4.1 of PSAR Chapter 12). The total design capacity of the system at a pressure of 0.15 MPa and a hydrogen volumetric concentration of 4 % is min.192 kg of hydrogen per hour. This capacity of system JMT is selected on the basis of the conditions when up to 1000 kg of hydrogen are released inside the accident localization zone within 5-7 hours during a severe accident. Design capacities of system JMT: Characteristics of RVK-3 recombiners: 1. Area of catalytic coating, m2 8.4 2. Initial concentration of hydrogen, % vol.0.45 3. Rated capacity at a pressure of 0.15 MPa and a hydrogen volumetric concentration of 4 %, kg/h 2.68 4. Time for reaching the rated capacity, s, max.180 (at steam-gas mixture temperature over 60 0C and hydrogen volumetric concentration over 2 %) max.180 (at steam-gas mixture temperature over 60 0C and hydrogen volumetric concentration over 2 %) 5. Final concentration of hydrogen, % vol.0.45 RVK-4 1. Area of catalytic coating, m2 16.8 2. Initial concentration of hydrogen, % vol.0.45 3. Rated capacity at a pressure of 0.15 MPa and a hydrogen volumetric concentration of 4 %, kg/h 5.36 4. Time for reaching the rated capacity, s, max.180 (at steam-gas mixture temperature over 60 0C and hydrogen volumetric concentration over 2 %) max.180 (at steam-gas mixture temperature over 60 0C and hydrogen volumetric concentration over 2 %) 5. Final concentration of hydrogen, % vol.0.45 Technical requirements for system JMT specify operating fluid parameters, at which operability of the system shall be maintained. One of such parameters is: Under DBA conditions with leakages from the primary and secondary circuits the e
T3-100	In the section 7.3.2 the measures used for the prevention of formation of explosive concentration of hydrogen-steam-air mixture in containment are mentioned (1) autocatalytic recombiners, (2) sprinkler system operation control, (3) opening of the emergency gas removal system and pressurizer PORV, (4) disabling the SG PHRS to increase the amount of steam in the containment. The justification of reliability and proper operation (efficiency) of these equipment / measures shall be presented.	The system of hydrogen removal from the containment with application of the catalytic hydrogen recombiners is a fully passive system not requiring the operator's involvement. It is designed with a margin of 10% in regard to the number of recombiners. After completing the answer to T3-99, give the reference to it. See the answer to T3-107. Deactivation of the sprinkler system during a severe accident is one of the measures to control the hydrogen situation in the containment aimed at additional inertizing of the containment atmosphere with steam. The main measure to limit pressure rise in the containment during a severe accident is using the containment PHRS. Provided that the sprinkler system remains in operable condition, it can be used to reach a safe controlled state after a severe accident. Particular instructions to the operator for control of the sprinkler system pumps will be specified in SAMG.
T3-101	*„Disabling the sprinkler system is one of the measures for managing the hydrogen situation by controlling the amount of steam in the containment with the sprinkler system. The steam amount in the containment is increased with the sprinkler system controlled by the operator."* Reducing the effectiveness of the sprinkler system can lead to an increase in pressure in the containment and loss of its integrity.	See the answer to T3-100.
T3-102	In the section 7.3.3 it is stated: **„** Damage to the concrete reaches 1 at a pressure of about 0.8 MPa. Damage to the internal surface reaches 1 at a pressure of about 0.98 MPa". Please explain what does it mean "reaches 1"?	This value characterizes concrete damage. 1 corresponds to 100% concrete destruction.
T3-103	In the section 7.3.3 some calculation results, which justifies the containment integrity in the case of loss of coolant accidents, are presented. It is not clear who and using what methodology performs these calculations. Please provide references to the calculation.	References to the calculations are given in /31/.
T3-104	In the section 7.3.4 the prevention of re-criticality in the case of core melting and failure of reactor vessel is presented. The model of the molten core catcher used in the Keff calculations is presented in Figure 7.3.4.3. it is visible from the figure, that inside of vessel of the molten core catcher there are three layers of different materials: (1) metal layer; (2) molten core material layer with voids filled with water; (3) water. It is not clear: • where is the mix of corium with the sacrificial concrete (before entering into the vessel of the molten core catcher the corium is contacting the top layer of the slab, which is made from special concrete which forms a liquid under layer at thermal contact with the hot molten core material)? • what is temperature of corium and how to explain the presence of the water in very bottom of core catcher (the density of corium is higher as water density)? • how the mix of corium with water is possible, how the steam explosion is prevented? The description of core catcher and justification of reliability and functionality of this equipment shall be presented.	Chapter 12 of SAR contains a detailed information regarding the core catcher.
	In the section 7.3.5 the consequences of corium entering into core catcher is briefly described. The core catcher in Belarusian NPP is very innovative and important equipment, because it allows: • prevention of the molten core material release beyond the established boundaries of accident localization area; • guaranteed cooldown of the molten core material; • ensuring subcriticality of the molten core material in the concrete shaft; • minimizing the release of radioactive substances and hydrogen into the space of the containment.	1. The special design of the bottom slab prevents escaping of the molten core material beyond the boundaries of the accident localization area when the molten core flows from the reactor vessel to the molten core catcher. The bottom slab is shaped as a funnel; it covers the bottom part of the reactor vessel and it is protected against thermal radiation by steel foil insulation. The bottom slab consists of a layer of low melting point concrete (to prevent blocking on the way of corium); beneath this layer there is a layer of heat insulating refractory concrete (to protect metal structures of the bottom slab and the cantilever truss. To prevent blocking of the way of corium at the breakaway of the reactor vessel bottom the bearing ribs extend above the thermal insulation. The central channel of the bottom slab has an easily melted removable biological protection plate made of leaden sheets. In the event of a severe accident with destruction of the reactor vessel the bottom slab receives the molten core from the reactor and directs it into the molten core catcher. Afterwards, the molten core material is kept within the established boundaries through core localization in the molten core catcher vessel. 2. The guaranteed cooling down of the molten core material within the molten core catcher is ensured by water supplied to the catcher shaft to cool down the catcher vessel from the outside. In the standby mode the shaft of the molten core ca The cooling system of the molten core catcher can supply water on the surface of the molten corium in two ways: passively through the water supplying valves and by the operator from the reactor internals inspection shafts (via the pipelines o - prevent destruction of the molten core catcher due to the radiant heat transfer from the melt mirror; - minimize release of radioactive fission products and hydrogen into the containment atmosphere; - additionally remove heat from the molten corium. 3. During interaction of corium with the sacrificial materials uranium oxide preserves the homogeneous structure. The made analysis showed that the homogeneous structure maintains deep subcriticality within the whole design range of temp At the final stage of cooling down corium in the molten core catcher the heterogeneous structure of corium and water may occur due to formation of cracks in the thickening corium doused by water. To prevent occurrence of secondary criticality at this stage strong neutron absorber (Gd2O3) is added into the sacrificial materials. 4. Water is poured on the surface of the molten corium also to minimize release of the radioactive fission products and hydrogen into the containment atmosphere.

T3-105	The description of core catcher and justification of reliability and functionality of this equipment shall be presented.	
T3-106	It is stated that *„Under these conditions, facilities and organizational measures mentioned in paragraphs 5.1.2 and 5.1.3 of the Report [31] shall be used for BDBA management*.“ Please provide facilities and organizational measures that shall be used for BDBA management.	In case of complete loss of auxiliaries of the power unit in combination with failure to start of DG, emergency power supply systems, and system of reliable power supply of normal operation, the following facilities (Technical measures) are used to control the accident: -Passive heat removal system from the steam generators (SG PHRS); -Passive heat removal system from the containment (containment PHRS); -Batteries of the BDBA management system; -Mobile DG to feed the BDBA consumers; The organizational measures to manage the said BDBA include: -On-line evaluation of the NPP power supply equipment (including the emergency one), and availability in the operational state of relevant systems and equipment; enactment of the "Plan of Measures to Protect Personnel in the Event of an Emergency at the Belarusian NPP" (if necessary). -Ensuring integrity of undisturbed physical barriers; -Limiting the radiation impact on the personnel, population and environment. -Organization of priority (urgent) activities for restoration of the power supply, including starting into operation of the mobile DG plant; -Ensuring water supply to the emergency heat removal tanks of SFP;
T3-107	In the sections 7.3.6 and 7.3.7 the equipment and instrumentation used to maintain the containment integrity is presented. The passive autocatalytic hydrogen recombiners and indicators for hydrogen concentration control are mentioned. The concentration of hydrogen is different in different places of containment during the accident. It is not clear how the places of installation of such equipment and instrumentation were selected. Are the computer simulation performed for selection of these places? The selection of the places of installation of such equipment and instrumentation shall be discussed.	System JMT is designed with a productivity to handle up to 1000 kg of oxygen which will be releasing during a severe accident inside the accident localization zone in the course of 5....7 hours. (Ch.12.2.3.4 SAR). 3-D analysis of the containment medium demonstrated that the hydrogen will be evenly distributed across the containment volume; the containment PHRS contributes at a large extent to the air circulation inside the containment. The sensors and recombines are placed evenly, taking into consideration links between the premises and main routes (directions) of movements of the medium. The SG boxes are equipped with supplementary recombiners, as the computation analysis identified a possibility of accumulation of substantial hydrogen concentration in the first seconds after the intensive hydrogen release. The design sets forth a 10% productivity margin thus ensuring that the system will reliably perform its functions. The hydrogen monitoring points are evenly distributed in the premises of the inner containment; The locations of the hydrogen monitoring points are selected basing on the results of analysis of propagation, accumulation, and burning of hydrogen. If practical, the measuring points and cable routing of various measuring channels are placed at a maximum distance from each other to exclude their simultaneous failure.
T3-108	In the section 7.3.8 the management of severe accidents in case of simultaneous core melting and nuclear fuel damage in the spent fuel pool is discussed. It is written that "simultaneous accidents in the spent fuel pool and the reactor have no impact on each other". But, in case of loss of the ultimate heat sink in combination with the NPP blackout the PHRS tanks and the spent fuel pool are make up by the same single low-power high-pressure pump JNB50AP001. The justification of reliability of the system with JNB50AP001 pump shall be justified. What actions should be taken in the case of failure of JNB50AP001 pump?	As demonstrated in section 6.1.2 , a simultaneous functioning of 3 channels of PHRS from 4 emergency heat removal tanks, the PHRS ensures a safe state of the reactor plant during 72 hours without violation of the acceptance criteria. At a later stage, when the routine EHRT makeup from desalinated water tanks (LCU) starts working, SG PHRS maintains a safe cooldown state of the reactor plant during 165 hours since the start of accident process. The total time of the spent fuel pool boiling-off to the FA heads from the beginning of the accident will be at least 89 hours. This way, the potential events of a severe accident in the reactor and SFP are occurring with a time lag (76 and 89 hours) , and may occur only in case of a complete inactivity of the NPP staff. -In the course of 72 hours the personnel takes no steps to makeup the EHRTs; -in the course of 89 hours the personnel takes no steps to makeup the EHRTs; The following measures must be taken to provide the makeup: -To makeup SFP no later than 69 hours after the start of an accident, as the makeup time lasts 3 hours, consequently in 72 hours it becomes possible to switch over to makeup of EHRT and ensures its makeup starting since 72 hours. BDBA management manual shall detail these actions of the personnel.
T3-109	*„Simultaneous accidents in the reactor core and spent fuel pool are analyzed in terms of their mutual impact in [31], Section 5.1.1.* *Severe accident management in case of simultaneous core melting and nuclear fuel damage in the spent fuel pool is analyzed in [31], Section 6.3.8.* *Simultaneous accidents in the spent fuel pool and the reactor have no impact on each other, because different safety systems are used to manage accidents in the spent fuel pool and the reactor. For example, FAK or JMN system is used for heat removal from the spent fuel pool, while JNG1,2, JND systems are used to remove heat from the reactor [31]."* It is stated only that simultaneous accidents in the reactor core and spent fuel pool have no impact on each other due to different systems to be used in case of severe accident, however, the broader results of the analysis from [31] shall be provided.	In our opinion, the National Report contains exhaustive information to comprehend that events in the reactor plant and SFP are independent, and the available systems and equipment are adequate to manage accidents in the reactor plant and SFP.
T3-110	It is stated that *„After the molten core material is released to the molten core catcher, the emergency alkali supply system (JNB91) supplies sodium alkali to the containment pits for fixation of the iodine volatile forms.“* What is influence on Keff value due possible sodium alkali supply?	Caustic soda supply to the sumps of the containment has no impact on multiplying properties of the molten core material in the molten core catcher, because the containment sump is not related in any way to the internal volume of the molten core catcher.

T3-111	It is outlined that „*Despite the fact that there are several different systems for implementing each of the accident management strategies, there are areas for further improvement in terms of the measures for management of beyond design basis (including severe) accidents. Taking additional technical measures and introduction of instructions for their use to ensure safety functions in case of loss of the design systems will improve the NPP ability for management of beyond design basis accidents at their severe stage*." Please elaborate in details what additional technical measures (and introduction of instructions for their use) to ensure safety functions in case of loss of the design systems will improve the NPP ability for management of beyond design basis accidents at their severe stage?	Development of supplementary technical measures which enhance manageability of severe accidents after failure of the design systems are subject to additional research and engineering efforts, and, consequently cannot be provided as an answer to this particular expert's comment.
T3-112	In the section 7.4.1 it is stated "*The design provides for measures to prevent loss of the containment integrity. Implementation of these measures for BDBA management ensures mitigation of Severe accident effects by: ... suppressing explosive concentrations of hydrogen by the combustion system to maintain the integrity of the containment". Chapter 7.3.7. Instrumentation Required for Maintaining the Containment Integrity (Page 138) „There are 44 recombiners in the containment rooms*". Please provide information on how the 44 numbers of recombiners were selected and whether this amount is sufficient.	Productivity of recombiners and their installation locations are selected following the analysis of propagation, accumulation, and potential hydrogen combustion modes inside the premises of the inner containment. In order to ensure maximum efficiency of the system, the recombiners are installed in places where the hydrogen concentration during the accident can reach maximum values, as well as on the ways of the steam-gas medium movement. The recombines are distributed fairly evenly in the premises of the inner containment. Exception is the SG boxes. The SG boxes contain the maximum amount of recombiners as compared to other premises. This is due to the fact that concentration of hydrogen in the SG boxes following the loss of coolant accidents reaches maximum values. The analysis of hydrogen situations under a severe accidents substantiate it. See also T3-107
T3-113	It is stated that „*The design provides for measures to prevent loss of the containment integrity.* *Implementation of these measures for BDBA management ensures mitigation of severe accident effects by:* *- using the normal operation systems and actions of operating personnel to prevent significant radioactive releases."* For BDBA management applying only of the systems of normal operation could lead to more harmful situation and consequences in case one of those systems will be not available. For BDBA management all available systems and equipment shall be applicable to prevent and mitigate accident consequences.	The measures to manage BDBA (SA) are implemented by functioning of the safety systems. The systems of normal operation can be used to prevent and mitigate severe accident consequences, however for a successful implementation of management actions their application is not necessary. In the National Report the use of the normal operation systems is considered as one of the measures which can be applied to manage BDBA (SA): "Implementation of these measures for BDBA management ensures mitigation of severe accident consequences by: --prevention of the reactor core destruction at an early stage by means of the primary circuit pressure reduction system; - suppression of explosion-prone hydrogen concentration by the incineration system to ensure integrity of the containment; - utilization of the passive heat removal system from the containment (containment PHRS); - keeping the maximum design pressure inside the containment (0.49 mPa) and ensuring the design air- tightness of the containment (permissible leak rate is 0.2 % of the volume per 24 hours); 158 -utilization of the core catcher for retaining the corium following displacement of the reactor core from the reactor vessel; -application of systems of normal operation and personnel actions to prevent severe radioactive releases"
T3-114	„*The life-support system of the MCR/ECR equipped with efficient treatment of the supply air at the aerosol and iodine filters, as well as the civil structures of the double containment and control room building UCB allow for permanent stay of the personnel at the MCR/ECR to manage the accident.*" How ventilation regime changes during an accident with radioactive products releases outside established boundaries?	In case of appearance of radioactive substances on the NPP site and increase of the radiation background, the Personnel Life Support System of the MCR and ECR is automatically (upon signal from the radiation monitoring sensors) switched over to the mode of aerosol and iodine filter purification of the intake air. In case of over-range of the design set point of the radiation control, the Personnel Life Support Systems of the MCR and ECR is automatically - by means of the radiation monitoring sensors- switched over to the mode of recirculation. MCR / ECR operator may proactively switch over the MCR / ECR ventilation to the purification or recirculation modes prior to beginning of an intensive emergency radioactive release. . In case the external atmospheric air is exposed to radioactive contamination in the concentrations when the filters cannot ensure the necessary degree of air purification , the design provides for automatic switching-off of the Personnel Life Support Systems of the MCR and ECR upon a signal from 2 out of 3 radiation monitoring sensors installed in air ducts of system SAC10/20/30/40. The air conditioning system of MCR premises SAC12/22/32/42 and air conditioning system of ECR premises SAC17/27/37/47 are switched over to an operation mode without supply of the outside air (full recirculation) , and provide the requ
T3-115	In the Report (see page 8) there are indicated a few documents - TKP 566-2015, dated 28.04.2015, and "Requirements to stress tests (objective safety reassessment) of a nuclear power plant", dated 12.04.2017, stating requirements for a format and content of the Belarus national stress test report. Following to the above-mentioned documents TKP 566-2015 (see para 10.6.4.2) and "Requirements to stress tests (objective safety reassessment) of a nuclear power plant" (see para 32.2), in the subsection „Management of Severe Accidents after Uncovering of Nuclear Fuel in the Spent Fuel Pool" of the Report, the following information must be provided: - control of hydrogen concentration; - providing adequate protection against ionizing radiation; - limiting emissions after severe damage of spent nuclear fuel in the cooling pools; - the means necessary for tracking the state of spent nuclear fuel for managing the accident; - the possibility of personnel staying in the premises of a unit control room.	See answer T3-109. The hydrogen concentration management inside the containment is provided in section 7.3.2 of the Report. Protection of the personnel and population from the external ionizing radiation caused by radiation sources inside the containment is provided by the biological protection of the construction structures of the double containment with a thickness of concrete no less than 1800 mm. Management of Severe Accidents after Uncovering of Nuclear Fuel in the Spent Fuel Pool is provided in section 7.4.2 of the Report. Description of the instrumentation to monitor the accident is provided in section 7.3.7 of the Report. The possibility of personnel staying in the premises of MCR is assured by the Personnel Life Support Systems of the MCR and ECR (see 7.4 of the Report and answer T3-114).

	Unfortunately, in the Report, in the subsection 7.4.2 „Management of Severe Accidents after Uncovering of Nuclear Fuel in the Spent Fuel Pool“, on page 144, there is no above-mentioned information. Could you please provide detail information (according to "Requirements to stress tests (objective safety reassessment) of a nuclear power plant", para 32.2) about readiness of nuclear power plant for management of severe accidents in the spent nuclear fuel pools.	
T3-116	It is stated that "the radiation effects of severe accidents will not exceed level 5 as per INES scale". Such statement shall be justified by deterministic analysis. The references to safety analysis of severe accidents in Belarusian NPP shall be presented.	As set set forth in 7.4.1, this statement is based on results of the deterministic analysis of radiation consequences of a Severe Accident, carried out in SAR of the Belarusian NPP (section15.7.5 SAR).