Article	Ref. in National Report	Question / Comment	Answer
General	P12	Whereas some challenges identified in the previous CNS review meeting are discussed in Summary, not all of them have been included. For example, how have Challenge 2 and 3 have been addressed? Has challenge 1 been fully addressed	CHALLENGE 1 - Completion of actions related to the “weld case” All corrective measures related to the “weld case” were implemented. Special processes were included in the activities carried out by the licensee. The supervision on technical safety was separated from the performance. A new process “Special Process and Technical Quality Management” has been introduced. Verification and evaluation of external supplies and services quality have been tightened and extended. Supervision for quality delivery has been strengthened. Design basis and design authority competencies have been established. CHALLENGE 2 - Establishment of a new set of regulatory guides in accordance with the new legislation. In connection with the Atomic Act adopted in 2016 and its implementing legislation, the complete review and revision of all existing safety guides and recommendations have been underway since 2017. Up to the present time, most regulatory guides have been published, and a planned series will be completed by the end of 2020. In fact, the process of issuing regulatory guides and reviewing them never ends. The set of regulatory safety guides is focused particularly on the following areas: organization management, quality assurance, education and training of nuclear power plant employees, nuclear power plant design requirements, safety classification of structures, components, and systems of nuclear facilities, safe operation limits and conditions, PSA, PSR, operation experience and feedback, maintenance, revisions and tests of equipment, component ageing management, fire prevention, abnormal and severe accident conditions management , external hazards, safety culture, and more. Regulatory guides are published on the SÚJB website https://www.sujb.cz/dokumenty-a-publikace/publikace-sujb/ CHALLENGE 3 - Recruitment of new staff and training of the staff based on human resource strategy, for both regulator and operator. REGULATOR: The “Training” part of this challenge is being addressed in accordance with SÚJB Integrated Management System (IMS) procedures. Each employee has their own specific “Plan of Personal Development” (IPOR) where all refreshment training and/or training for new assignments represent the most important component. The implementation of IPOR is reviewed and (if needed)

			updated annually in dialogue between the employee and his/her direct supervisor. For managers at different levels, one of the sources for IPOR development/review should be the results of "competence mapping" performed periodically by each of the main SÚJB departments. Securing appropriately qualified personnel is a part of existing SÚJB policy and strategy documents at different levels. The "Recruitment" part of this challenge is a continuous "issue" for SÚJB, as probably for most of the nuclear regulators around the world. For SÚJB the main problem was that 10 to 15 posts of technical (inspector) staff were empty for a number of years. The reason was simple – the Czech Republic has had, in recent years, the lowest unemployment rate in the European Union (around 3 percent) and in the area of highly qualified technical personnel the rate is practically zero. Nevertheless, SÚJB was given the opportunity for financial years 2020, 2021 to try to find personnel for 8 inspector posts. Among others, SÚJB tries to attract young engineers by campaigning at different departments of technical universities and providing internships. Another measure to enhance professional capacity was the establishment of a new Nuclear Safety Branch in SÚRO (one of SÚJB´s TSOs). The plan is to hire around 20 highly qualified personnel for this branch in the 2016-2020 period. OPERATOR The licensee has employed and trained dozens of new professionals over the past 3 years in connection with new process and department Managing of the special processes. The strategy has been fulfilled. CHALLENGE 4 - Completion before 2022 of research and analytical activities related to prevention and mitigation of potential Temelín NPP core melt accidents. The license holder for NPP Temelín 1 and 2 performed an extensive set of analyses. Analyses confirmed the possibility of stabilization of the partially melted core before its major relocation to the pressure vessel bottom part in the event of sufficient water refilling. Analyses didn't confirm the reasonable applicability of external vessel cooling (because of the need of early start of water supplying, need of stable and sufficient water inlet and steam outlet which is extremely complicated without application of flow deflector surrounding the pressure vessel). Analyses also didn't confirm the reasonable applicability of refractory linings (core catcher) installation in the reactor cavity and adjacent area GA302. But the analyses of the rate of the containment basement melting through shows that in the event of corium cooling from the top in the reactor cavity and adjacent area GA302, the stabilization of the corium and prevention of the melting

			through is possible. Based on the results of the analyses, the license holder performed the modification of existing pressure relief valves to enable its remote controlling during a severe accident and is now also preparing the installation of another alternative direct molten fuel cooling system. This is the completely independent new diesel driven pump system dedicated for the corium cooling both in the in-vessel phase of a severe accident and during the ex-vessel phase. The diesel driven pump system is presently under the project preparation phase. Stabilisation of the corium outside the reactor pressure vessel is a challenge for all of the types of operated units and therefore the extensive research activities are still running in the topic (e.g. the research project ROSAU) to confirm the existing strategies and solutions and to bring new recommendations and inputs for safety enhancements (mostly in the MCCI topic). CHALLENGE 5 – Completion of new Integrated Management System for the regulatory body. This challenge continues – existing IMS is being continuously improved through the activities of the special task force (Quality Team) created by the SÚJB Chairperson in July 2019. Members of the Quality Team represent both management and inspectors. The Quality Team is headed by the Manager of quality, who regularly reports on its activities at the management meetings. The Quality Team meets at least once a month, its documents and minutes are accessible to all employees of the Office on the internal website. The activities of the Quality Team are monitored by all employees, and it is considered to be a welcome platform for maintaining and improving IMS. The main challenge of the Quality Team for the next years is to foster the implementation of criteria of improvement, described by the Ministry of Interior as part of the project “Support for the professionalization and quality of civil service and public administration”. These criteria include e.g. system of internal regulation, system of communication, strategy of development, policy of human resources, and change management.

General	summary	In his report, the President of the 7th review meeting had recommended that Contracting Parties consider the implementation of the good practices that where identified during the meeting. Could your country provide information on the actions carried out with regards to the implementation of those good practices in your country ?	There are 4 good practices identified in the 7th CNS RM President report.The Czech Republic actively participated in the 1st ENSREG topical peer review on ageing management as was suggested.SÚJB actively participated in INSC projects focused on nuclear safety level improvements in non-EU countries, e.g. in Armenia, Islamic Republic of Iran, etc.The licensee cooperates with non-EU countries in the field of nuclear safety through WANO Peer Review missions (Armenia, USA, India, China, Japan, Argentina, Canada, Russia, Ukraine, etc.). The Czech Republic is also highly active in communication on nuclear safety topics with neighbouring countries. SÚJB employees participate in various activities focused on informing the public on status of nuclear installations.However, our legislation does not allow SÚJB to pay the public for support in SÚJB activities as it is in Canada.The Czech Republic and SÚJB, similarly to Hungary, organised public hearings in neighbouring countries within the framework of EIA for new nuclear units in Dukovany NPP.

General	Page 13, paragraph 4,subbullet 4	The report states that one of the priorities for 2019 is: "Ensure the transfer of knowledgeand experience of specialists leaving the SÚJB to new colleagues so as not to compromise the proper exercise of SÚJB competence, even in a situation where the number of service posts in the Nuclear Safety Section was reduced and the service posts of retired employees were cancelled." Could you share what tools or strategies you plan to use to address this?	The main tool we use to ensure the transfer of knowledge and experience are Competence maps, within which each head of department must specify how to maintain and develop required skills. Knowledge and experience are transferred on an everyday work basis, also during informal teambuilding activities or by systematic or on-the-job training. In addition, TSO or professional knowledge network can be used. For an office based on knowledge, the transfer of skills is the essential process. Management encourages inspectors to share knowledge and motivates them appropriately in order to improve the exchange of experience. Good working relationships and developed internal communication support this effort. Therefore, we are finalizing a new Concept of internal communication which encourages open and effective communication.
General	Page 12	It is written that the PSR has been ongoing for the Temelin Nuclear Power Plant since 2017 after 20 years of operation. On page 8 it is stated that both units of this power plant were put into operation in 2004? It inditaces that PSR started after 13 years of opearation, not 20, could You explain these differences?	SÚJB uses the possibility to state conditions when issuing the Decision on permit to stipulated nuclear activities. As PSR was not explicitly required in the previous legislation, the first PSR of Temelín NPP was required by SÚJB Decision on permit for operation of Temelín Units (in 2004) given the fixing dates April 2010 as for Unit 1, November 2011 respectively for Unit 2. The new Decisions on the permit for operation of Unit 1, respectively 2, unified the date for next (in fact second) PSR to April 2020. Since this year, the PSR will be repeated every 10 years based on the Atomic Act requirement.

General	Summary, page 13	One of SUJB's priorities for 2019 is as follows: in connection with the implementation of the new Atomic Act and implementing decrees, crucial attention will be paid to the release of all relevant safety guides and recommendations. What first-priority safety guidelines and recommendations are planned for release? Does SUJB have a plan to develop or improve safety requirements and regulations?	In connection with the Atomic Act adopted in 2016 and its implementing legislation, the complete review and revision of all existing safety guides and recommendations have been underway since 2017. Up to the present time, the most regulatory guides have been published; a planned series will be completed by the end of 2020.In fact, the process of issuing regulatory guides and reviewing them never ends.The set of regulatory safety guides is focused particularly on the following areas: organization management, quality assurance, education and training of nuclear power plant employees, nuclear power plant design requirements, safety classification of structures, components, and systems of nuclear facilities, safe operation limits and conditions, PSA, PSR, operation experience and feedback, maintenance, revisions and tests of equipment, component ageing management, fire prevention, abnormal and severe accident conditions management , external hazards, safety culture, and more.Regulatory guides are produced by teams consisting of SÚJB employees, often with the technical assistance of external specialists in given areas, and in cooperation the legal department of the SÚJB. The process of development of a document is subject to the SÚJB Legislation Plan as approved and amended through SÚJB management meetings. In support of the development of guides, the guideline of VDS 045 “Rules for Controlled Documents” was issued. In the development of regulations and guides for NPPs, the SÚJB takes into consideration comments from interested parties and feedback based on experience. The SÚJB releases guides on its website https://www.sujb.cz/dokumenty-a-publikace/publikace-sujb/ and some of them are published in paper form.

General	Annex 4	The report stated " In the years 2016 - 2018, for example, the following projects were implemented under the Plans for Safety Enhancement", one of effort is measures to protect the elements of critical information infrastructure (cyber security). What is the scope to protect the elements of critical information? How the Czech Regulatory body assess prevention of cyber attact to digital control and protection systems in NPPs?	“Critical information infrastructure” and “cyber security” are not mentioned in the Atomic Act and therefore are not within our competence. Both terms came from the Cyber Security Act which is within the competence of the National Cyber and Information Security Authority (NÚKIB).Czech Atomic Act No. 263/2016 Coll. (valid since 2017) contains Section 163 related to Computer Security:Obligations of license holders in the area of security of nuclear installations and nuclear material (1) Holders of a license under § 9(1)(b) to (h) and (5) shall a) secure the computer systems necessary for the management of nuclear safety, nuclear material accountancy, physical protection and radiation extraordinary event management against unauthorized useRegarding assessing “prevention of cyber-attack” by our office, Decree No. 361/2016 Coll. (related to the Atomic Act; valid since 2017) in Section 19 states:Security of computer systems(1) The computer system needed to control the nuclear safety and account for nuclear material, physical protection and radiological emergency management shall be secured against unauthorized use by defense in depth, considering any possible consequences in case of the design basis threat coming true.(2) A professionally competent person shall be designated in a nuclear installation with the inner or vital area delineated to ensure the security of computer systems of a nuclear installation.(3) The licensee shall take administrative and technical measures to prevent intentional misuse of computer systems, in which case any single failure to implement the administrative and technical measures shall not result in the jeopardy included in the design basis threat.(4) The licensee shall regularly assess the level of security for computer systems including periodic testing.And in Section 28:CONTENT OF DOCUMENTATION FOR LICENSED PRACTICES IN THE AREA OF SECURITY(3) The plan of physical protection assurance shall include e) The plan of organizational measures, which shall include 3. The plans computer security in the field of nuclear safety management, accounting for nuclear material, physical protection and radiological emergency management against intentional misuse, which include a description of the organization and definition of the obligation to ensure security of information systems in a nuclear installation, the method of assets management, risk assessment and vulnerability, a description of the way and control of changes in configuration and the method of security of information systems, and a description of personnel measuresIn the period 2017-2019, the main changes in the information and cyber security management system were concerned to:• organizational changes to strengthen the competency and capacity in Cyber security for both NPPs;• registration and protection of connected equipment to OT

			(IT), records of assets, block diagrams, risk register, management system documentation (organizational measures, technical measures), training of all levels, SOC activity started;• risk analysis, conformity checks of measures (implemented to eliminate identified risks).

General	Annex 4	Under the heading PLANS FOR SAFETY ENHANCEMENT (ANNEX 4), with respect to information on projects implemented in the years 2016 - 2018, Can Czech Republic elaborate on the regulatory requirements and guidelines for aging management of buried pipes in its NPPs?	Regulatory requirements for ageing management of buried piping for Czech NPPs are the same as for other SSC, depending on its safety significance. This specifies the following laws and regulations:- Atomic Act No. 263/ 2016 Coll. (with requirements on ageing management and the list of documentation for the operational permit);- Decree No. 21/2017 Coll., on Assuring Nuclear Safety of Nuclear Installations (with requirements on ageing management programme and ageing management process); - Decree No. 162/2017 Coll., on Requirements for Safety Assessment pursuant to the Atomic Act. – with requirements for special safety assessment in case of intended operation beyond the design lifetime (Section 23[3]).We do not have any specific guidelines for buried piping.More information on ageing management of buried piping can be found in the Report for the Purposes of Topical Peer-Review “Ageing Management” under the Nuclear Safety Directive 2014/87/EURATOM , which you can find on our web page https://www.sujb.cz/en/reports/, (together with some other reports).
General	Page 13	The Report states that the rate of stability was 93.7 %. Could you explain what is the essence of this indicator?	This indicator is also known as the “employee stability index”. It comprises the number of employees who have stayed in the organisation over 12 months divided by the total number of employees in the last calendar year (multiplied by 100 to obtain percentage).

Article 6	Page 15, paragraph 3	In May 2017, WANO did a peer review mission at the Corporate level with the licence holder and highlighted the following as an area of improvement: "strengthening the corporate supervision; the reporting system in the company is not set to support the improvement process or enable early intervention in case of recognizing negative trends." Can you share what measures have been taken to address this area?	The measures taken are: Define and implement a corporate system of NPPs performance oversight, monitoring, and results communication, supporting the ČEZ Nuclear Division management strategic decision-making, and application of the control role. Implemented, for example:The information for the Board of Directors (quarterly monitoring of nuclear activities) was complemented with an independent evaluation of the technical condition of the NPP and with certain charts and comparisons. The PAS (Board of Directors) is annually informed of the results of safety culture and given tasks to eliminate identified deficiencies. An additional independent evaluation of the most prominent risks for ČEZ’s nuclear activities was established. The Safety Inspectorate of the ČEZ Group Department, Internal Audit Department, and Safety Department take part in the evaluation. The document is submitted to the meeting of the Board of Directors in the first quarter of the year.Key indicators have been set to monitor the performance of both power plants at the level of the Division’s management. The newly established divisional reporting gives the Division’s management a consolidated view of the performance of both power plants through the development of major KPIs, allowing their mutual comparisons where this is practicable. In some cases, they are also compared e.g. with the median of the results of other NPPs. Reporting on the evaluation of efficiency of the correction and prevention system is being prepared as an instrument for continuous improvement. The divisional reporting also includes an evaluation of the current condition of performance of goals and prediction of expected results (where this is possible). The historical (usually 5-year) and current performance of both power plants and development trends are also compared. Reporting is part of the programme of Division Days with the presence of the Chief Nuclear Energy Officer, the management of both NPPs, central sections up to the level of D-2, and invited managers of sections of other divisions. The risk management system concerning risks threatening the performance of divisional and site goals was set, including the goals of individual sections on NPP sites. The risk management system in the nuclear division is set and consistently linked from the level of areas for improvement to the divisional level. Inputs into risk management from the working level as well as from external resources are provided. The extension of the risk management system made in a “top down” hierarchy of nuclear division, means natural data collection from the bottom up, with risk escalation to higher levels in the hierarchy of nuclear division based on the rules set by the applicable managing document.

Article 6	Annex 4	The Report states that in the period 2016-2018 the critical information infrastructure protection measures were carried out. Could you explain what do they consist of?	"Critical information infrastructure" is not mentioned in the Atomic Act and is therefore not within our competence. This term came from the Cyber security Act which is within the competence of the National Cyber and Information Security Authority (NÚKIB).In the Czech Atomic Act No. 263/2016 Coll. (valid since 2017) Section 163 relates to Computer security:Obligations of license holders in the area of security of nuclear installations and nuclear material (1) Holders of a license under § 9(1)(b) to (h) and (5) shall a) secure the computer systems necessary for the management of nuclear safety, nuclear material accountancy, physical protection and radiation extraordinary event management against unauthorized useMore details can be found in Decree No. 361/2016 Coll. in Section 19:Security of computer systems(1) The computer system needed to control the nuclear safety and account for nuclear material, physical protection and radiological emergency management shall be secured against unauthorized use by defense in depth, considering any possible consequences in case of the design basis threat coming true.(2) A professionally competent person shall be designated in a nuclear installation with the inner or vital area delineated to ensure the security of computer systems of a nuclear installation.(3) The licensee shall take administrative and technical measures to prevent intentional misuse of computer systems, in which case any single failure to implement the administrative and technical measures shall not result in the jeopardy included in the design basis threat.(4) The licensee shall regularly assess the level of security for computer systems including periodic testing.And in Section 28:CONTENT OF DOCUMENTATION FOR LICENSED PRACTICES IN THE AREA OF SECURITY(3) The plan of physical protection assurance shall include e) The plan of organizational measures, which shall include 3. The plans computer security in the field of nuclear safety management, accounting for nuclear material, physical protection and radiological emergency management against intentional misuse, which include a description of the organization and definition of the obligation to ensure security of information systems in a nuclear installation, the method of assets management, risk assessment and vulnerability, a description of the way and control of changes in configuration and the method of security of information systems, and a description of personnel measuresIn the period 2017–2019, the main changes in the information and cyber security management system were concerned to:• organizational changes to strengthen the competency and capacity in Cyber security for both NPPs;• registration and protection of connected equipment to OT (IT), records of assets, block diagrams, risk register, management system documentation (organizational measures, technical measures), training of

			all levels, SOC activity started;• risk analysis, conformity checks of measures (implemented to eliminate identified risks).
Article 7	P22	Is there any experience with legal and regulatory enforcement actions that could be shared with the country group?	Yes, for example, a fine of CZK 10,000,000 may be imposed on the licensee (ČEZ, a.s., Dukovany Nuclear Power Plant) for the documentation of welded joints of selected equipment (hereinafter referred to as “VZ”) of the Dukovany Nuclear Power Plan. The performance of non-destructive inspections of these welds did not reflect the actual performance of these welds and the results of the non-destructive inspections, and thus did not document the actual state of the VZ. Details can be found in IRS Report No 8637 of 2017-03-11.

Article 7	7.2.1, p. 26	Challenge No. 2 from CNS 2017 addresses the establishment of a new set of regulatory guides in accordance with the new legislation. Could the Czech Republic please comment on the status and scope of this work and give an overview of the process to develop these guides?	In connection with the Atomic Act adopted in 2016 and its implementing legislation, the complete review and revision of all existing safety guides and recommendations have been underway since 2017. Up to the present time, most regulatory guides have been published, and a planned series will be completed by the end of 2020.In fact, the process of issuing regulatory guides and reviewing them never ends.The set of regulatory safety guides is focused particularly on the following areas: organization management, quality assurance, education and training of nuclear power plant employees, nuclear power plant design requirements, safety classification of the structures, components, and systems of nuclear facilities, safe operation limits and conditions, PSA, PSR, operation experience and feedback, maintenance, revisions and tests of equipment, component ageing management, fire prevention, abnormal and severe accident conditions management , external hazards, safety culture, and more.Regulatory guides are produced by teams consisting of SÚJB employees, often with the technical assistance of external specialists in given areas, and in cooperation the legal department of the SÚJB. The process of document development is subject to the SÚJB Legislation Plan as approved and amended through the SÚJB management meetings. In support of development of guides, the guideline of VDS 045 “Rules for Controlled Documents” was issued. In the development of regulations and guides for NPPs, SÚJB takes into consideration comments from interested parties and feedback based on experience. The SÚJB releases guides on its website https://www.sujb.cz/dokumenty-a-publikace/publikace-sujb/ and some of them are published in paper form.
Article 7.1	Section 7.1.4, page 25	This section identifies countries with which the Government of the Czech Republic has concluded bilateral agreements on cooperation in the field of nuclear energy use. Are there plans to conclude bilateral agreements on cooperation in peaceful use of nuclear energy and with which countries?	At the moment there are no concrete plans for concluding a bilateral agreement on cooperation in the field of nuclear energy use.

Article 7.2.3	Page 29	How many unscheduled inspections have been performed in the Czech Nuclear Power Plants since 2016 and what topics did they cover?	In 2019, 11 unscheduled inspections were carried out at the Dukovany NPP aimed at ensuring physical protection; ensuring radiation protection when handling ionizing radiation sources; transport of radioactive waste; verification of the introduction of the core of the EDU Unit 3 and control of nuclear materials.In 2019, 9 unscheduled inspections were carried out at the Temelín NPP aimed at ensuring physical protection; verification of the activation of the core of the Temelín NPP units; sealing of CASTOR packaging and inspection of nuclear materials.In 2018, 7 unscheduled inspections were carried out at the Dukovany NPP aimed at ensuring physical protection; implementation, verification and evaluation of investment project of seismic improvement of EDU units; loading of CASTOR packaging sets and nuclear material control.In 2018, 6 unscheduled inspections were carried out at the Temelín NPP aimed at verifying the introduction of the core of the Temelín NPP units; loading of CASTOR packaging sets and nuclear material control.In 2017, 11 unscheduled inspections were carried out at Dukovany NPP, aimed at ensuring physical protection; the effectiveness of corrective actions in spent nuclear fuel storage and nuclear material control.In 2017, 2 unplanned inspections were carried out at the Temelín NPP, focused on the inspection of nuclear materials.In 2016, 2 unscheduled inspections were carried out at the Dukovany NPP, aimed at evaluating the setting and calibration of AZ measurement channels and nuclear materials.In 2016, 3 unscheduled inspections were carried out at the Temelín NPP, focused on the inspection of nuclear materials.
Article 8	P39	8.1.10 Inspection plans are published on the SUJB website. However, no mention is made if the inspection reports are published. Consider clarification.	Reports of inspections performed by SÚJB are not fully published on the SÚJB website. The results of the SÚJB inspection activities for the given calendar year are presented in the Report on the Results of Activities of the State Office for Nuclear Safety and Monitoring of the Radiation Situation in the Czech Republic, in the chapter “Inspection Activities”. Furthermore, SÚJB publishes, on its website, the total number of inspections for individual entities and Inspection Efforts for individual audited entities for a given calendar year.
Article 8	§ 8.1.5 p.36	Does the Czech Republic foresee a further decrease in the State Office for Nuclear Safety SUJB workforce in the coming years and if so, how does it plan to deal with it?	No reduction of workforce is expected. On the contrary – as per Government resolution No. 485/2019 – SÚJB was given back 8 posts of technical (inspector) staff that were previously taken away since they were empty for a number of years. In addition, the Czech Republic is planning to increase the number of staff of SÚJB for the next year. A further increase will be carried out in connection with the expected construction of new reactor blocks in the Czech Republic.

Article 8	§ 8.1.6 p.37	Could the Czech Republic describe more precisely the process of identifying the skills needed to carry out the tasks as well as the process put in place for effective and efficient transfer of skills?	The process of identifying the required skills is based on the "Strategy of development of human resources" and related methodical instruction "Elaboration of competence maps". There is an obligation for each head of department to specify a set of required skills (at the basic, medium, or high level) for each position. Each skill is associated with a corresponding provision of the law. Required skills can be acquired by internal or external means. For each position, a three-year competency development plan is established – internally, skills can be improved by recruitment, training or reorganisation; skills can be achieved externally by outsourcing, TSO or using knowledge network. Competence maps are approved by senior management. For institutions whose functioning is based on knowledge, such as the State Office for Nuclear Safety, the transfer of skills is an essential process. Management encourages inspectors to share knowledge and motivates them appropriately in order to improve the exchange of experience. Good working relationships and developed internal communication support this effort. The transfer of skills is conducted on an everyday work basis as well as during informal teambuilding activities, and by systematic or on-the-job training.

Article 8	§ 8.1.10 p.40	Could the Czech Republic explain how it involves the stakeholders in the regulatory body's decision-making process?	Generally, according to Act No. 500/2004 Coll., Code of Administrative Procedure, any state administrative body is obliged to be helpful to any persons whose rights could be affected, to provide them with necessary information and explanations, and to allow them to apply their rights and to preserve their interests. Additionally, according to this act, whoever demonstrates a serious interest may see the official file (documents) of the particular case, and any person may submit proposals and applications to the office and can participate in the administrative proceedings if their rights might be affected. Possible participants of the administrative proceedings (formal decision-making process resulting in administrative act – "decision", e.g. license) are defined by this act as applicants and persons whose rights or obligations might be affected. Even though the scope of possible participants in the administrative proceedings is set more specifically by Act No. 263/2016 Coll., Atomic Act, for licensing procedures, other stakeholders can participate under special circumstances (based on Act No. 100/2001 Coll., on environmental impact assessment). In other decision-making procedures of SÚJB, the aforementioned general rule is applied.Informal decision-making, as a continual process used during all administrative and inspection activities of the SÚJB, is not limited in any manner and enables wide involvement of relevant stakeholders through various communication channels. SÚJB receives different information and suggestions from stakeholders on an everyday basis and reacts to them – reflects them in the ongoing administrative proceedings (as evidence) or inspections, starts the administrative proceedings or inspections on their basis, or applies them in the assessment, enforcement, planning, or regulation-making activities. SÚJB even pro-actively searches for advice and information from stakeholders (esp. experts) to use it in its decision-making, if needed.However, informally SÚJB is opened to any proposals and inputs from the general public and other stakeholders, even regarding the decision-making process. SÚJB is obliged to evaluate each obtained information and use it within the decision-making process if relevant, regardless of the source of such information. All stakeholders' inputs are properly filed and archived to allow for their use even in the future.

Article 8	8.1.5, p. 36	Challenge No. 3 from CNS 2017 addresses recruitment and training aspects. Could the Czech Republic elaborate on the present situation, as the regulatory body is still facing a decrease of staff members. Could the Czech Republic describe the measures taken to tackle this issue.	In real terms, the number of staff has not been reduced in recent years. The problem was that 10 to 15 posts of technical (inspector) staff were empty for a number of years. The reason was simple – the Czech Republic has had, in recent years, the lowest unemployment rate in the European Union (around 3 percent), while in the area of highly qualified technical personnel the rate is practically zero. Nevertheless, SÚJB was given the opportunity for financial years 2020, 2021 to try to find personnel for 8 inspector posts. Among others, SÚJB tries to attract young engineers by campaigning at different departments of technical universities and providing internships. Another measure to enhance professional capacity was the establishment of new Nuclear Safety branch in SÚRO (one of SÚJB´s TSOs). The plan is to hire around 20 highly qualified personnel for this branch in the 2016-2020 period. In addition to the above, the Government of the Czech Republic plans to increase the number of employees of SÚJB in connection with the expected construction of additional nuclear units in NPP Temelín and NPP Dukovany.
Article 8	Page 39, section8.1.10	Secion 8.1.10 describes how the regulator uses the the SÚJB website to share information with the public. In addition to sharing information on a lot of topics through the website, do you use other forums to communicate with the public and provide opprtunity for them to ask questions, like hosting public meetings to discuss licensing activities?	SÚJB uses numerous tools to communicate with the general public. There are two laws regulating free access of the general public to official information in the Czech Republic, i.e. Act No. 123/1998 Coll., on the right to information on environment, and Act No. 106/1999 Coll., on the free access to information. Both these laws require state administrative bodies to make publicly available listed information on their official activities (e.g. the most important documents, strategies, conceptions, reports, etc.) and, in general, to create and support opportunities for the general public to communicate with these bodies and to obtain requested information. According to Act No. 500/2004 Coll., Code of Administrative Procedure, any state administrative body is obliged to be helpful to any persons whose rights could be affected, to provide them with necessary information and explanations, and to allow them to apply their rights and to preserve their interests. Additionally, according to this law, whoever demonstrates a serious interest may see the official file (documents) of the particular case, and any person may submit proposals and applications to the office and can participate in the administrative proceedings if their rights might be affected. These general principles must be applied in all official activities of SÚJB. Moreover, specific information channels are provided through the Atomic Act (Act No. 263/2016 Coll., which lists information made obligatorily public via SÚJB website (§ 28; e.g. licences, permissions, registrations, notifications). The aforementioned legal requirements form the necessary minimal base for communication with the public, supplemented with informal ways of

			communication. The general public has the right to submit any information, input, or even complaint through the official e-mail address of SÚJB (podatelna@sujb.cz), or physically through the official post address or via the official desk at the headquarters of SÚJB. Less formally, and even anonymously, a request can be sent to the SÚJB through its on-line forum on the website. Special attention is paid to the protection from radon since it impacts a great portion of the population, therefore, SÚJB operates a special website on this topic (https://www.radonovyprogram.cz/uvodni-strana/), and a special e-mail address for asking questions and providing information to SÚJB was introduced as well. The general public may communicate with the SÚJB even via Facebook Messenger.The general public can meet SÚJB's representatives in person as part of administrative proceedings (if a person participates in the proceedings or is involved in other way) and during official public meetings, workshops, and conferences organized by SÚJB to address various topics. SÚJB informs the public about its activities, regulatory requirements, particular problems to be solved, and ways to deal with them (incl. licensing process) within these meetings. Although the public meetings are not an obligatory part of the administrative proceedings from the legal point of view, it is not excluded to hold them to inform the general public and to receive valuable inputs therefrom. However, privacy issues and the general nature of the proceedings, which is non-public, must be respected.
Article 8	8.1.9, p.38	GSR Part 1 (Rev.1) Req. 15 states that the regulator should make arrangements to identify lessons learned from operating and regulatory experiences. Does the IMS of SÚJB provide for this? Especially, what type of arrangements exist for managing regulatory experience?	SÚJB has established a special Commission dealing with both operating and regulatory experiences HKI (Commission Assessing Inspections). HKI members (all relevant managers – heads of sections, units, etc.) meet regularly once a month and assess and evaluate results of and experience gained during SÚJB inspections, including efficiency of inspections and procedures of the regulatory body. Based on the HKI findings, other controls/inspections can be targeted at problematic areas with the aim to further increase efficiency of inspections and assessment activities.

Article 8	8.1.6, p.37	Could you please elaborate on how much of the training is oriented towards the changing nature of the workload that is due to the expected newbuilds?	In accordance with the SÚJB Integrated Management System (IMS) procedures, each employee has their own specific “Plan of Personal Development” (IPOR) in which all refreshment training and/or training for new assignments represent the most important component. The implementation of IPOR is reviewed and (if needed) updated annually in dialogue between employee and his/her direct supervisor. For managers at different levels, one of the sources for IPOR development/review should be results of “competence mapping” performed periodically by each of the main SÚJB departments. Securing the appropriate number of qualified personnel is, of course, a part of existing SÚJB policy and strategy documents at different levels.
Article 8	8.1.5, p.36	How does the approval of the number of staff of SÚJB by Government affect the independence of SÚJB? E.g. with the upcoming possibility of new units, the Chairperson of SÚJB could be of the opinion that more staff is needed to assess safety-aspects in the phases of siting, licensing, and/or construction. Would the Chairperson then be dependent on Government to be able to hire extra staff?	The independence of SÚJB is not affected by the mechanism of the systemization. The mechanism is regulated by Act No. 234/2014 Coll., on Civil Service. SÚJB submits a proposal of number of service posts based on its need for adequate human resources to ensure independent supervision and to be able to fulfil its assigned responsibilities. The proposal is assessed by the Ministry of Finance of the Czech Republic and the Ministry of the Interior of the Czech Republic, but only in terms of its impact on the economy, state property, state budget, etc. The draft systemization is prepared by the Ministry of the Interior. The systemization is adopted by the Government of the Czech Republic for the upcoming calendar year. Everyone involved in the assessment of the proposal and the draft is aware that the independence of the regulatory body is derived directly from the CNS and from the European directives as a legal and political obligation, and that it cannot be interfered with. Moreover, the system of fees payed on the activities of SÚJB by some licence holders provides partially (about 55 %) necessary funds to perform assessments of the safety aspects and to related staffing. SÚJB is therefore less dependent on the state budget, which is taken into consideration by the Ministry of Finance and the Government of the Czech Republic when adopting the draft systemization. The upcoming projects of strategic significance shall probably have a positive effect on the number of service posts in SÚJB.

Article 8	8.1.12, p.41	Does SÚJB also have an advisory committee that would consist only of people from outside of the State Office, who would be able to advise on matters of a more general nature and with an outsider view?	SÚJB uses external technical and scientific support from two TSOs – the National Radiation Protection Institute (SÚRO), providing support primarily in the field of nuclear safety and radiation protection, and the National Institute for Nuclear, Chemical, and Biological Protection (SÚJCHBO), providing support primarily in the field of chemical and radiation protection. SÚJB regularly uses expertise from various external experts on contractual basis, including commercial expert organizations in particular areas of interest, universities, and research and development organizations. If needed, an ad hoc advisory body composed of external experts may also be established. All these subjects provide advice not only on specific issues related to nuclear safety or radiation protection, but also on more generic topics, such as integrated management system of SÚJB, communication strategies, privacy protection, or concepts and strategies of the state supervision.
Article 8.1	8.1.11, p.41	The SÚJB also cooperates with many other organizations such as research institutes (e.g. Research Centre Řež Ltd.), departmental organizations of the ministries (e.g. Ministry of the Environment – Czech Geological Survey), technical and science universities, Academy of Sciences of the Czech Republic, relevant national and international organizations, companies and private experts in the field in question (in the field of the natural characteristics of the sites, external hazards, civil engineering industry, and assessment of	In the Czech Republic, it is not always easy to find a reputable and quality nuclear power expert who is also independent of the monopoly electricity producer from nuclear sources at ČEZ. For this reason, SÚJB cannot explicitly monitor human resources and human factor in its potential suppliers. However, in the field of nuclear safety assessment, it has been working with recognized experts in the field for a long time. As an example, assistant professor Miloš Ferjenčík, Ph.D., who works as the head of the Institute of Energy Materials at the Faculty of Chemical Technology at the University of Pardubice, significantly supports SÚJB in assessing operational events from nuclear facilities. Miloš Ferjenčík regularly lectures at a training courses aimed at investigating the root causes of events in Petten, Netherland. The quality of human performance of SÚJB suppliers is indirectly monitored by the consistent evaluation of the quality parameters of each specific contract. If these high requirements are not met, cooperation with the supplier is terminated. In order to ensure the stability of independent support in a wide range of SÚJB activities, SÚJB has been building an independent TSO support in its research organization SÚRO, v.v.i. since 2017. Regarding sitting aspects and continuous site characteristics evaluation, SÚJB uses the support only from experts or organizations that are independent of operators of nuclear facilities. Given that there is a very limited number of experts in seismicity or active fault assessment in the Czech Republic, it is known who works for operators and who independently for SÚJB (they are individuals from different institutions). This requirement of independency is fulfilled also by the

		events and human factor). The supporting entities are obliged to be separate from and independent of an operator of a nuclear installation. Expert support is particularly used in the assessment of Safety Analysis Reports and documentation required for a licensed activity. Q: Can you explain how other organizations monitor the human factor? In which activities the human factor is monitored?	Czech Geological Survey (established in 1919 and in charge of the state geological service pursuant to Section 12 of Act No. 62/1988 Coll., on Geological Works, as amended and Sections 12 to 16 of the Decree of the Ministry of the Environment No. 368/2004 Coll., on geological documentation, collects and processes geological information) in basic geological survey and the creation of sets of basic geological maps of the Czech Republic in various scales for nuclear power plant and research reactors. In the last 4 years, the Czech Geological Survey elaborated special tectonic maps of site vicinity of Temelín NPP and Dukovany NPP, which SÚJB uses for independent assessment of the issue.

Article 9	8.1.6, p 37	Are full-scale simulators used for training of personnel in the event of severe accidents? If not, is it planned to upgrade the simulators to ensure training for severe accidents? / It is important, that severe accidents are simulated, to take the appropriate measures when required. Having only an emergency manual is not enough.	Large complex upgrade of the models used in the Temelín NPP simulator was completed in 2017. The upgrade was coupled with the extension of the simulation, which allows training activities of both the control room staff and the technical support centre staff during the transition from EOPs (Emergency Operating Procedures) to SAMG (Severe Accident Management Guidelines). The scope of the simulation is limited by the core outlet temperature of about 900 °C. The simulator used in the Dukovany NPP for the training of the control room staff and members of the technical support centre allows training for all events according to procedures EOPs (Emergency Operating Procedures), including training the area transition from EOPs to SAMG (Severe Accident Management Guidelines) – procedures SACRG. The scope of the simulation is limited by the core outlet temperature of about 1200 °C.Training in these areas in Temelín NPP and Dukovany NPP currently meets the requirements of the new legislation of the Czech Republic (Decree 21/2017):- full-scope simulator is used for staff training in the EOPs area,- full-scope simulator is used for staff training in the transition from EOPs to SAMG, - the simulation tool is used for staff training in the SAMG area.The first two requirements are fulfilled by a full-scope simulator, and the third requirement by a special tool VINSAP – Visualization of NPP Severe Accident Progress. VINSAP is a visualizer for displaying the parameters of severe accident scenarios calculated by the MELCOR calculation code. This specialized software for staff training (especially for TSC – "Technical Support Centre Group") was completed in 2017.

Article 10	P56	10.1.7 The section seems to discuss management system of the regulator instead of programmes to be used by licence holder to prioritise safety in activities for design, construction and operation of nuclear installations. Please clarify.	Thank you for your inquiry. Yes, this section discusses the Regulatory body management system; the safety priority of the Licensee is discussed in other parts of Article 10. Below is a summary of the Licensee's programs to ensure the safety priority:“Measures taken by licence holders to implement arrangements for the priority of safety, such as those mentioned above (safety policies, safety culture programmes and development, arrangements for safety management, arrangements for safety monitoring and self-assessment, independent safety assessments, discussion on measures to improve safety culture, a process oriented [quality] management system) and any other voluntary activities, examples of Good Practices and safety culture achievements” are:LICENCE HOLDER SAFETY POLICIES: Safety and Environmental Protection Policy – valid in the entire ČEZ Group and the related Policy of Safety in Nuclear Activities of ČEZ, a. s.;SAFETY CULTURE PROGRAMMES AND DEVELOPMENT: The licence holder shall introduce the safety culture in the management system by categories, characteristics, and attributes according to the WANO (Traits of a Healthy Nuclear Safety Culture – PRINCIPLES PL 2013-1). The assessment of safety culture takes place in an annual period (January to February following the year for which the assessment is made – the previous year), and the results and the measures taken are documented in the form of a summary document “Analysis of the Safety Culture in ČEZ, a. s. for the Previous Year”. The basis for developing a safety culture within the company ČEZ, a. s. are the Plans for Safety Culture Development that determine systemic measures in response to the outcomes of the assessment of safety culture for the previous period.Ensuring the clarity of characteristics and attributes of a healthy safety culture for employees and contractors takes place in various forms of training in the area of safety culture. At the meetings, the “Safety Notes” are used to develop a safety culture and point to a specific problem or exemplary practice linked to a particular attribute of safety culture.Leaders at all levels of management consistently provide feedback on positive behaviour from the perspective of safety culture under the Observation Program. At the same time, they increase employee and contractor motivation by compliments or using other incentive-based instruments. Single and multiple information and visualization campaigns are carried out through communication.ARRANGEMENTS FOR SAFETY MANAGEMENT: The safety strategy adopted in ČEZ, a. s. focuses on the continuous fulfilment of basic safety goals and nuclear safety principles (included in the internal control documents of the company in accordance with the international standards, experience and recommendations, and in accordance with the valid legislation of the Czech Republic) with maximum use of safety culture principles

			and quality assurance requirements. ARRANGEMENTS FOR SAFETY MONITORING AND SELF-ASSESSMENT: One of the tools for systematic assessment of the level of nuclear safety is a set of indicators which characterize trends of the nuclear safety level and the radiation protection level in nuclear power plants during the past week, month, and year. Through the monthly reports, the company's managers thus obtain the feedback for the assessment of the safety requirement implementation success-rate. Target fulfilment of the obligation of superior position of the requirements for safety and environment protection to the requirements of production as well as fulfilment of the obligation concerning continuous improvement of safety culture also includes yearly updated strategic tasks of the Chief Executive Officer and Director of Nuclear Energy Division of ČEZ, a. s. as well as the tasks of separate Programs and Plans for Safety Enhancement for both power plants.INDEPENDENT SAFETY ASSESSMENTS are done by specialists (as part of licence holder) which are independent of the performed/assessed activities.DISCUSSIONS ON MEASURES TO IMPROVE SAFETY CULTURE are done during the assessment of safety culture.PROCESS ORIENTED (QUALITY) MANAGEMENT SYSTEM – The basic framework of the powers and responsibilities, as well as the method of assurance of the activities performed for fulfilment of all safety obligations within the company, are defined by the Rules “Organizational Structure, Mission and Tasks of the Units, Manual of Integrated Management System” and “Safety Management in ČEZ Group” along with related Directive “Safety Management of ČEZ, a. s.”. The above control documents describe, in terms of organization and process, control mechanism of activities in the fields with performance of activities important to nuclear safety.ANY OTHER VOLUNTARY ACTIVITIES – none
Article 10	P56	10.1.5 Are any other type of independent safety assessments undertaken other than ones related to events?	Yes, there are various types of independent assessments besides these related to events. The Licensee performs independent assessments of all submissions to the regulator, and many other independent assessments, e.g., independent assessment of nuclear safety assurance, independent assessments of management system effectivity, etc. SÚJB performs an independent assessment of all documents delivered by the operator within the framework of the state supervision, as described in other sections of the national report, e.g. in the section 8, and section 10.3.

Article 10	P55	10.1.4 Can more information regarding licence holder’s arrangements for safety monitoring and self-assessment be provided? Please clarify if this section describes the regulator self-assessment or arrangements for self-assessment that licence holders can use.	This part of the report describes the Authority's self-assessment; VDSs are the Authority's guidelines and internal regulations that govern the Authority's internal processes and which also contain the working procedures of the Authority.Below are the rules that the Licensee follows when monitoring safety and self-assessment:Licensee has procedurally set up a system of defining safety requirements (ČEZ_PP_0428) and supervise compliance (ČEZ_ME_1139). Each safety process (Nuclear safety, Radiation protection, Emergency preparedness, etc.) has an inspection system for monitoring and inspection of compliance with the defined requirements and correctness of their settings – by walk down, measurement, exercise, data collection and analysis, etc.Licensee has set up a system of self-assessment (ČEZ_ME_0848):• OVERVIEW SELF-ASSESSMENT is carried out to improve the overall performance of a division / site / company. Overview self-assessment is implemented across elements of the management system, e.g. management areas / processes / activities / structure, and thus provides a comprehensive view of division / location / company performance.• MANAGEMENT / PROCESS SELF-ASSESSMENT identifies the degree of compliance of management / process documentation and implementation with management / process requirements and good practice.• SELF-ASSESSMENT OF THE ACTIVITY is carried out for a purpose to support the necessary changes of activities, resources (human, material, etc.), and indicators.

Article 10	§ 10.4 p.58	Could the Czech Republic clarify how the State Office for Nuclear Safety ensures the confidence of the public and other stakeholders in the actions carried out?	SÚJB uses different ways to ensure the confidence of the public and other stakeholders in the actions carried out. SÚJB obligatorily publishes relevant information about its activities, including issued licenses and registered and notified activities, via its website in accordance with Act No. 123/1998 Coll., on right for information on environment, Act No. 106/1999 Coll., on free access to information, and § 28 of Act No. 263/2016 Coll., Atomic Act. Every year, the annual report of the SÚJB activities and state of the nuclear programme of the Czech Republic is published, upon requirement of the Atomic Act. Moreover, SÚJB is obliged to inform the general public through its official notice board.According to Act No. 123/1998 Coll., on the right to information on the environment, and Act No. 106/1999 Coll., on free access to information, any person has the right to request information from SÚJB regarding its official activities (except for classified and some other types of sensitive information), and SÚJB is obliged to provide such information in comprehensive manner and prescribed period of time. Subsequently, each provided information is obligatorily published on the SÚJB website.Less formal ways to establish the confidence of the public and other stakeholders in actions carried out include the special Facebook profile of SÚJB and special website on the radon topic. The office holds informational campaigns regarding issues of specific interest, such as new legislation adoption and nuclear accidents (e.g. Fukushima). Besides these activities, SÚJB organizes particular workshops and meetings dealing with specific topics, available to pertinent experts, regulated persons or even the general public. SÚJB and its officials are opened to any requests from the general public and provide ad hoc consultations and explanations as needed.

Article 10	§ 10.1.6 p.56	Does the Czech Republic observe improvements following incorporating the safety culture by the State Office for Nuclear Safety SUJB in the way they assess the safety of the NPPs according to the licensee and the results of these assessments? Could the Czech Republic provide information and examples on the influence of these improvements on safety in Czech Republic Nuclear Power Plants? Could the Czech Republic precise which kind of enforcement SUJB expects to use if the regulatory oversight of the safety culture shows bad results and needs for improvement?	The Licensee is obliged to report on safety culture assessment annually by law (SÚJB Decree No. 408/2016 Coll.). As regards the assessment of possible improvement, the frame “10 Traits” was put into practice at the end of 2018, so we still need some time for evaluation. Quarterly, SÚJB sends its findings to the licence holder as one of the inputs of the independent assessment of its safety culture. This document contains statistical evaluation and a list of concrete findings that were identified by SÚJB inspectors. As per the agreement between ČEZ, a.s. and SÚJB, the Licensee informs SÚJB on the steps they implemented/proposed, or corrective measures they have taken. In the second half of 2019, SÚJB inspectors performed the first inspection on evaluation of safety culture process within ČEZ, a.s. This inspection has not been finished yet. The safety culture in itself cannot be enforced. However, SÚJB is focused on the correct settings of the process, reaction/feedback of the Licensee, and the most reliable results.

Article 10	10.1.1, p. 54	It is stated in the National Report that the IAEA Safety Requirements No. GS-R-3 has been superseded by the document GSR Part 2 “Leadership and Management for Safety”, which was issued in 2016. Could the Czech Republic please clarify, whether it is planned to adjust the Integrated Management System in accordance with the new requirements.	At the earliest revision of the IMS Manual (it is expected in 2020), we plan to adjust the Integrated management system to the new requirements established by the GSR Part 2 “Leadership and Management for Safety”.

Article 10	10.1.2, p.55	The requirement for the implementation of the Safety Culture Programme and its development isbased on the Atomic Act, where it is one of the requirements for the Management System. Pursuant to Section 30(7) of the Atomic Act, the licence holder shall develop a healthy safety culture. Coherentand systematic activities for the development of the safety culture are called “Safety CultureDevelopment Program”.Q: Could you briefly explain the Safety Culture Programme? Could you explain more in detail how the basic points of the program can be meet?	ČEZ, a.s. regularly (once per year) evaluates the safety culture (SC). This evaluation consists of various areas of partial evaluations (evaluation of events of category 1-3 from the view of SC, periodical SC survey of ČEZ employees and employees of main contractors, independent evaluations of the National regulator SÚJB and evaluations of [internal] independent nuclear oversight, evaluations of international missions WANO and supplementary evaluations, such as Focus groups, observations, evaluations of low-level events and nonconformities [event of category 4], corrective action plan [CAP] fulfilment from previous period). Based on identified gaps, the management of Nuclear division sets Top “X” priorities (usually “X” stands for 5 top priorities, but there may be more or less) that are used to formulate the SC CAP (forming Safety Culture Program). The SC CAP also includes fostering safety conscious (required) behaviour. Further SC improvement is provided through Leadership, human performance meetings, where managers/leaders evaluate the SC improvement of their own departments. For SC development we also use various visualization tools (communication campaigns, intranet articles, TV screens, posters, Nuclear division management newsletters, booklets, etc.). These tools are used to inform the contractors’ employees on all matters of Safety Culture. SC training for all employees is provided regularly as part of entrance training and periodical trainings. Summary information on safety culture program is regularly submitted to National regulator (SÚJB) as a yearly Safety Culture Report.

Article 10	10.1.1, p.54,55	Have you been able to (self-)assess the current state of Safety Culture at SÚJB'? If so, how do you plan to improve on that in line with continuous improvement?	The management is fully conscious that a healthy safety culture within the regulatory body is a fundamental characteristic of an effective regulator. Management considers safety culture not only to be a matter of oversight, but also as a matter of self-reflection.The basic principle of safety culture is defined in IMS Policy, IMS Manual, and the principle is developed in the Strategy of State Office for Nuclear Safety. The newly established Quality Team (since June 2019) is also a strong tool for the enhancement of culture in the organization, including the safety culture aspects. One of the first tasks for the Quality Team was the improvement of internal communication, which is closely related to the safety culture. A survey on internal communication was organized with the participation of almost all employees. Answers and suggestions arising from the survey were discussed and assessed within the Quality Team and within the Top Managers meeting. As a result, we are finalizing new internal documentation “Conception on internal communication” which contributes to the improvement of safety culture by supporting open and transparent communication.The possible use of new IAEA guidelines for safety culture self-assessment for the regulatory body, which were published in September 2019, are now under consideration by the responsible SÚJB unit.
Article 10	p. 58-59	Could you, please, describe how the senior management of the regulatory body develops individual values, institutional values and behavioural expectations to support the implementation of the management system?	As mentioned, behavioural expectations, and individual and institutional values are described in the Code of Ethics, IMS Policy, IMS Manual and Strategy of SÚJB, which are regularly updated. All these documents are approved by senior management. Senior management develops the aforementioned values and behavioural expectations by permanent communication (using a top-down or bottom-up approach) including feedback assessment. In other words, leadership is provided through managers by communicating the key values and ethical behaviour. Senior management contributes to promoting and developing values within the newly established Quality Team (since June 2019). The Quality Team represents all staff including both senior managers and inspectors. Quality Team members meet regularly once a month. Topics concerning core values and behavioural expectations are often discussed and, if necessary, the Quality Team suggests appropriate measures. The Quality Team’s recommendations are discussed at the top managers meetings, which take place usually once a week.

Article 11	p 13	According to page 13 ”Ensure the transfer of knowledge and experience of specialists leaving the SÚJB to new colleagues so as not to compromise the proper exercise of SÚJB competence, even in asituation where the number of service posts in the Nuclear Safety Section was reduced and the service posts of retired employees were cancelled.” Does this mean that the number of personnel in the authority has been reduced in the last years? Please provide some detailed information on the personnel of the SUJB. / The reduction of staff in the given context would be worrying.	In real terms, the number of staff has not been reduced in recent years. The problem is that 8 posts of technical (inspector) staff were cancelled for 2019 since they were empty for many years. Nevertheless, SÚJB was given these 8 posts back for financial years 2020, 2021 to try to find appropriate personnel. In addition,the Government of the Czech Republic plans to increase the number of employees of SÚJB in connection with the expected construction of additional nuclear units in NPP Temelín and NPP Dukovany.
Article 11	Section 11	Could you clarify how much money was spent to finance the following works in 2016-2018: - Improvement of nuclear, radiation, environmental, technical and fire safety of NPPs; - Upgrading of existing NPPs; - Training and maintaining	We are very sorry, but this information is a trade secret, so we cannot share it with you.

		qualifications of the personnel?	
Article 11.2	11.2., p.63	The above mentioned legal regulations have been complemented with the Safety Guide BN JB-1.3 issued in December 2010 by the SÚJB, covering professional education and training of staff for the performance of work activities (positions) at Czech nuclear installations. The Guide specifies criteria and provides methodical guidelines for management and execution of training of employees of nuclear installation operators and employees of legal and physical entities whose activities (positions) at nuclear installations are important to nuclear safety, with the objective to minimize risks caused by	The aim of this Safety Guide is to elaborate the requirements for the system of professional education and training of workers performing activities at the nuclear installation. It includes namely the following areas: • recruitment and selection – contains its plan, criteria/requirements etc.;• education concept and system – describes forms and policy of education and training of personnel; • special professional qualification and competence – includes their definitions, qualification requirements/conditions as per position, and procedures to obtain the required qualification; • professional education and training of personnel – specifies groups of personnel and forms and parts of education (basic, periodic, other) including the areas it covers; • certificates, authorization, and licence – includes conditions and criteria for authorized personnel;• educational and training facilities – specifies required equipment and human resources;• training programs – describes individual plans, assessment, modifications, and records. The SÚJB issues the criteria and methodical instructions for conducting professional education and training of personnel to perform work activities (functions) at nuclear installations as a recommended procedure for the management and implementation of professional education and training of licensee personnel and workers of other legal and natural persons to perform work activities (functions) at the nuclear installation.Each NPP employee has a job profile that defines its activities, responsibilities, and qualifications. A qualification profile is set up by his manager in the SAP application. The relevant qualification is governed by the training program. The training program sets out the training conditions: target, target group,

		human failure.Q: Could you explain in more details what is the content of Safety Guide BN JB-1.3.. How often training is provided for the personnel in nuclear power plant?	method, content, time range, training period. The training period is set mostly in the range of 1-3 years. Mandatory trainings take place annually.

Article 12	P70	Can information on how human factors are consdiered in design and modifications be provided?	(1) Aspects of human factor in the project of the control centres of NPPs:The basic set of documentation on the given topic consists of conceptual projects and especially the document TEM-DOC-011 - Control room design topical report.A comprehensive overview of human factor considerations in the NPP project is provided in the Final safety analysis report. This document states that the principles of engineering psychology and ergonomics were taken into account to the greatest possible extent in all phases of the project and construction. The project team responsible for the design of the man-machine interface included technical experts from engineering, psychology, ergonomics, nuclear power plant operation, systems security, legislation, and licensing terms. It should be noted that many of these teams are still directly available for the preparation and provision of relevant modifications and are involved in the assessment or verification of the implementation of modifications. These activities also transfer the experience and applied man-machine interface policies to other team members, who deal with modifications of technological equipment, including relevant control centres.A complex of standards is applied for the design of control centres, especially:• ČSN IEC 960 - Functional design criteria for the communication of safety parameters for nuclear power plants;• ČSN IEC 964 - Design of control rooms for nuclear power plants;• IEC 965 - Auxiliary control points enabling reactor shutdown;• ČSN IEC 1227 - Nuclear power plants – Control rooms – Operator control means;• ČSN EN 457 - Acoustic warning signals.In addition to these standards, other guidelines are also used (e.g. EPRI NP-3659), established concepts are followed, and proven management stereotypes are implemented.Changes in the design of control centres verify the requirements of NUREG 0700 – Human-System Interface Design Review Guidelines.(2) Management modifications from the perspective of their influence on the human factor:Human factor requirements (ergonomics, etc.) are an integral part of the forms and methodological documents for preparing design changes (e.g. part of the Project Plan form, which summarizes technical and other modification requirements).During the preparation of modifications, especially in the processing of verification programs, mutual multi-professional cooperation is ensured especially from specialists who prepare changes to the operating regulations and also to the operators of control centres.Before the modification is carried out, the training of operations personnel is carried out as a standard, with an emphasis on the impact on their activities in order to avoid new additional risks of human error. Verifying and evaluating proposed modifications to the human-machine interface using a full-fledged simulator is not systematically performed prior to implementation.

			The full-scope simulator is updated on the basis of the modifications made; however, it is used for the verification of modifications during their preparation only in the event of high complexity of modifications and impacts on the man-machine interface. Such cases were, for example, the modernization of the flow-through parts of the low-pressure turbine parts, the high-pressure oil control of turbine, and condenser steam dump.Several dozen modifications have been made to the continuous improvement of working comfort, optimization of the man-machine interface and measures to prevent human errors (e.g. providing a dark alarm panel, optimizing signalling, and other ergonomic aspects).The evaluation of the impact of modifications on the human factor is also part of the requirements of the valid legislation of the Czech Republic (Act No. 263/2016 Coll. and related decrees) and is part of the so-called special evaluation of safety change pursuant to Decree No. 162/2017 Coll
Article 12	12.2, p.73	The SÚJB systematically monitors the impact of human and organization factors on the operational safety. Q: How do you include human factors methods and criteria in all phases of the modification process and all modification related activities?	Section 10 (2) of SÚJB Decree No. 21/2017 Coll., on Ensuring Nuclear Safety of a Nuclear Installation, states that “Notification of modification in the use of nuclear energy pursuant to paragraph 1 (a) and (b) shall include: …. e) assessment of the impact of the modification on human factor.” A human factor expert of SÚJB then evaluates whether the assessment has been made to the best of their knowledge and belief.

Article 13	§ 13 p.74 to 80	Could the Czech Republic precise procedures and guidance to manage detection of non-conforming, counterfeit, suspect or fraudulent items received from suppliers before they are installed in the plant? Could the Czech Republic precise the inspection program focusing on preventing and detecting the incorporation of non-conforming, counterfeit, suspicious and fraudulent items?	This issue is resolved within the requirements of Section 30, subsec. 2 of the Atomic Act (Act No. 263/2016 Coll.), where it is stipulated that the Supplier of a product or service may only be a person with an established management system in accordance with this Act (the implementing regulation is the Decree No. 408/2016). The requirements of Section 6 of the Decree No. 358/2016 must be met in the procurement process.Furthermore, this issue is addressed in the IAEA guidance on CFSIs “Managing Counterfeit and Fraudulent Items in the Nuclear Industry” (IAEA Nuclear Energy Series NP-T-3.26).The licensee has the basic principles purchasing of items described in managing documentation (SKČ_SM_0023) and Protection against fraudulent and counterfeit items in managing documentation ČEZ_ME_0626:In the event that an Item which is known to have counterfeits, or if there is a risk that a Fraudulent or Counterfeit Item may be delivered, the specification of that Item shall define in particular:- identification of critical characteristics to be verified during material or service takeover;- where relevant, customer inspection at specified stages of production or testing. This requirement must be consulted with the quality control department;- determination of specific tests;- requiring complete certification related to the product;- requiring an entrance checking to focus on the risk of counterfeit delivery;- defining other risk minimization measures.

Article 13	page 78	It is undeerstood that in Czech Republic the use of non-nuclear industry strandard equipment for safety related SSC is currently not allowed (or extremelly difficult to licence).Our information is that the utilities have launched own national projects to implement processes to allow wide use of non-nuclear industry equipment for safety related SSC. Please provide information on the status and outcomes of the projetcs. What is the position of the regulator on this activity and what are the preconditions of the regulator (SUJB) that the use of non-nuclear industry equipment could be used (safety related and non-safety related)?	• It is true; the use of non-nuclear equipment, in the ordinary commercial design, is not permitted in Czech NPPs for safety SSCs and safety-related SSCs at present.• The NPP operator is preparing an internal regulation allowing, in compliance with specific conditions, the use of equipment (or parts and components of equipment) in commercial quality also for safety SSC and safety-related SSC.• Three pilot projects have been implemented recently to verify the possibility of using of commercial equipment in the position of selected equipment for safety SSC, respectively for safety-related SSC. The NPP operator is currently completing the process of using of commercial items in positions for safety SSCs and safety-related SSCs, i.e. internal working documentation / methodology is being prepared (ČEZ_ME_1156). The procedure for preparing the possibility of using commercial items in positions for safety SSC and safety-related SSC has been consulted with the regulator (SÚJB) from the very beginning, i.e. approximately from 2017. The whole process will be presented to the regulator after the completion of the fourth pilot project and the finalization of the methodology, and the operator expects the process will be approved by the regulator.

Article 13	Page 77	What do safety-relevant items mean? Are these items assigned to specific safety classes? How often are audits of vendors and suppliers performed?	Safety-relevant items are selected equipment. The selected equipment, as defined in the Atomic Act, is a system, structure, or other component of a nuclear installation that affects nuclear safety and the performance of safety functions.Safety function means the operation of a system, design, or other component of a nuclear installation that is relevant to ensuring the nuclear safety of the nuclear installation.For the purposes of a graded approach to quality assurance, selected equipment is classified into safety classes 1, 2, and 3 according to the safety functions it contributes to.Audits of vendors and suppliers are carried out by the licensee at least once every 3 years, more frequently if they are negative, i.e. once a year.Inspections of suppliers are carried out by the SÚJB in accordance with the internal guidelines of SÚJB VDS 037 and VDS 008 with the frequency specified in accordance with the SÚJB inspection plan on nuclear installations. The Inspection plan is approved for the respective calendar year, or according to the current justified need for such inspection.
Article 14	14.1.2, p.85	Have Level 3 PSA been carried out for the plants in Czech Republic? If yes, what are relevant results and conclusions? If no, is it intended for the near future? / Level 3 PSA is recommended internationally. It gives a good insight into the impact of a nuclear accident in the neighborhood.	Not at all. PSA Level 3 has not been and currently is not considered in the scope of PSA analyses, nor is it legally required by the regulator. In the Czech Republic, in accordance with applicable national legislation, it is required that the licensee process PSA probability models in the PSA Level 1 and 2 range for power states and shutdowns, for both internal and external initiation events / hazards. PSA Level 3, following on from PSA Level 2, is outside this range of the required probability analyses. It should be noted that only a minimum of countries operating nuclear installations currently have legislative requirements to process PSA Level 3. One of the reasons for this is the high uncertainty in PSA Level 3 results compared to PSA Level 1 and then Level 2.

Article 14	14.2.3, p.98	How are the structures and the structural elements assessed which are inaccessible for visual inspections (indirect/indicative methods)? / The examination of partially inaccessible components such as pipelines, concrete structures and cables is not explained in the report.	1) Concrete structures and buildingsThe evaluation of concrete structures in places inaccessible for visual inspections is performed according to the Maintenance Program in the form of operational diagnostic measurements and evaluation of ageing parameters within the Ageing Management Program. The Maintenance Program and the Ageing Management Program are part of the reliability management system.Tests and diagnostic measurements include:(a) Leak and integrity testing of the containment. The results are compared with the permitted project bases.(b) Tightness testing of reactor cavity by monitoring leak detection of non-hermetic linings and floor by local overpressure leak test.(c) Leak testing of semi-service areas and hermetic seals.(d) As part of the operational inspection program, corrosion loss mapping of hermetic linings is screened. Diagnostic ultrasonic testing of the thickness of hermetic linings is performed by the point method and Phased Array method.(e) Measurement of leaks from double-lined pools by means of diagnostic measurements using water level, valve tightness checks, and drainage lines throughout.(f) Geodetic measurements of vertical displacements of the so-called settlement of building structures: Reactor building, Ventilation stack, Auxiliary service building for primary systems, Turbine building, Cooling towers, Ultimate heat sinks.(g) Measurement of very accurate levelling and internal continuous measurements in containment integrity tests.(h) Prestressing system of the Temelín NPP containment – the prestressing of the cables of the containment prestressing system, the armature moisture and the cable manhole moisture are measured.i) Measurement of concrete solidity by non-destructive method - Schmidt hardness tester.(j) Measurement of concrete carbonation depth.(k) Non-destructive measurement of steel cladding thickness by ultrasonic method.(i) Measurement of deviation of steel cladding from concrete structure by acoustic tracing and compilation of cavity map.(j) Moisture monitoring of concrete internals, Analysis of chemical composition of concrete in internals, Tests of the effect of boric acid solution on concrete internals.The ageing management program is based on the evaluation of the effect of degradation mechanisms on the building structures and concrete and steel structures of power plants. The selection of buildings for the program is based on Scoping for ageing management.Parameters evaluated in the ageing management program include tests and diagnostic measurements according to the maintenance program, chemical analysis, measurement of solution temperatures, evaluation of the number of cycles of filling and draining of pools, measurement of prestressing of cable of the containment prestressing system.2) CablesCable evaluation

			at areas inaccessible for visual inspections is performed according to the Maintenance Program in the form of operational diagnostic measurements and ageing parameters assessment within the Ageing Management Program. The Maintenance Program and the Ageing Management Program are part of the reliability management system.Diagnostic measurements include planned periodically repeated measurements of operated cable lines (6 kV, 0.4 kV) in the form of electrical parameter measurements and reflectometric measurements. The implementation uses the ECAD measuring system. Reflectometric measurements are used to locate and determine the type of cable faults. Measurement and evaluation of electrical parameters (capacitance, inductance, dielectric absorption [DAR], polarization index [PI], phase angle, insulation resistance) and reflectometry provide complete information about the electrical properties of the tested cable line. Criteria are set for the evaluation with an assignment of the deadline for corrective action.The ageing management program is based on the evaluation of the effect of degradation mechanisms on witness cables stored in cable deposits in the power plant area and on laboratory tests of witness cables. The selection of cables for the program is based on Scoping for ageing management. Test cables are tested in the laboratory for electrical and mechanical characteristic, including tests simulating environmental conditions in the power plant. The result is ageing curves with prediction of residual life. These ageing curves are periodically reassessed on the basis of tests on parts of witness cables taken from cable deposits. The ductility of the cable insulation material is always measured in mechanical tests, and the results are compared with the criteria recommended by the IAEA. In addition, within the ageing management program, diagnostic measurements on cables that have been in operation for a long time and, after disconnection, become witness cables stored in their original location is performed. The results of all the above parameters, including ECAD measurements, are periodically evaluated once a year.3) Monitoring, testing, sampling, and checking the concealed pipelines of the Dukovany and Temelín nuclear power plants is carried out through the following activities:1. Ultrasonic measurement of residual pipe wall thickness at available locations;2. Aerial thermography for the identification of places with leakage of supply routes outside the power plant area;3. Measurement of direct gradient of potential of supply routes enabling identification of damaged asphalt insulation (DCVG check of supply line insulation status);4. Tightness test of the supply routes on the Dukovany NPP;5. Loops with corrosion coupons are installed, which are periodically evaluated at Dukovany NPP and Temelín

			NPP.On some routes at Dukovany NPP, which were recently repaired, EDMET measurement is installed.

Article 14	§ 14.2.3 p.95 to 98	Could the Czech Republic give more information on the ageing of the reactor's internal components and of the ageing phenomena of corrosion of stainless steel and base nickel alloys and of high cycles fatigue?	At present, the implementation of two specific ageing management programs (AMP) for the internal parts of the reactor has been started at Czech NPPs. Meanwhile, the degradation mechanisms that influence the ageing of the internal parts of the reactor have been identified. The development of these degradation mechanisms leads to critical events that may affect the required function of the internal parts of the reactor. Unstable crack development, loss of component bearing capacity, and unacceptable change of geometry are considered critical events in the internal parts of the reactor. By analysing the ageing issues of the internal parts of the reactor, the content of these specific AMP was defined, which were designed in accordance with the IAEA requirements defined in SSG-48 "Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants". These are the following two specific AMP: 1. Ageing Management Program of radiation and fatigue damage of internal parts of reactor.Corrosion cracking and irradiation assisted stress corrosion cracking (SCC and IASCC) are evaluated based on periodic operational inspections performed for the presence of a crack.The nucleation of IASCC crack according to the standard NTD ASI, section IV (VERLIFE, Annex C) is calculated. The assessment is performed for irradiated material in contact with the primary circuit; this assessment is not performed in LEA areas (where the crack is postulated according to methodology).2. Ageing Management Program of vibration damage of internal parts of reactor.The development of high cycle fatigue is evaluated according to the standard NTD ASI, section IV. Based on the calculation, the fatigue limit for critical spots, which are:• Upper shaft flange (VVER 440 and VVER1000)• Bottom of the shaft (VVER 440)• Bottom fixation of the shaft – welding of the bracket to the reactor pressure vessel (VVER 440 and VVER 1000)In case of exceeding, a calculation for high cycle fatigue is performed. It may also be supplemented by non-destructive inspection.

Article 14	Pag.96	For the case of Specific equipment of group A. Please describe TLAA (Time Limited Ageing Analysis) Please develop in some detail what “recommend from PSA” is meant. / Graded approach is used on ageing management strategy. “graded approach is selected on the basis of the strategy defined for the care of SSC” Criteria for the selection of systems subject to the ageing management process are set out:- equipment recommended from the PSA according to Decree No. 162/2017 Coll.,	TLAA (Time Limited Ageing Analyses) documentation is defined in accordance with IAEA SSG-48, Chapter 5.64 and the internal methodology of ČEZ_ME_1031. TLAA lists for Dukovany NPP and Temelín NPP are included as free attachments in the ČEZ_PG_0001 NPP Ageing Management Program.PSA recommendations: it is about equipment included in the Scope of the ageing according to the methodology based on the results of the regularly updated "Living" PSA. The selection criteria are described in the relevant methodology. They are based on a regular reassessment of the relative importance of the equipment included in the PSA model. There are several different measures of importance from the PSA results, but the size of FV (Fussel-Vesely), RAW (Risk Achievement Worth) and CCDP (Conditional Core Damage Frequency) are important for ageing management, and these are used for selection.A graded approach in reliability management:Reliability management must be provided in a graded manner according to the importance of SCC and the impact of potential failure on safety and production. For power plant SCC, the SCC category (SCC categorization) must be determined according to the approved methodology, which is based on the determination of the impact of the equipment on the fulfilment of system functions, considering the influence on performance of safety functions, safe shutdown, and energy production. The result of the categorization is the division of SCC into equipment: critical, non-critical, and non-important. The chosen SCC maintenance strategy must consider the category and specificity of SCC, current status, operating modes, working conditions, legislative and licensing requirements, expected reliability, and required time to restore or termination of SCC operation, medium term assignment and site concept.

Article 14	Pag.96	Please, could you clarify if active components are within the scope of the AMP? / The overall Ageing Management Programme (AMP) in ČEZ, a. s., is set for both sites (Dukovany and Temelín) and includes requirements of the relevant IAEA Standards, IAEA Safety Guides (including SSG-48 “Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants”) and WENRA Safety Reference Levels. The way of quality assurance for the process of ageing management as required in Decree No. 21/2017 Coll., is defined and described in the document ČEZ_PG_0001 Operational Ageing Management Programme for NPPs . More information about the Overall Ageing Management Programme can be found in the report to the TPR [14-1].	Yes, active components are within the scope of the AMP.

Article 14	Pag.96	Please, could you clarify if active components are within the scope of the AMP? / The overall Ageing Management Programme (AMP) in ČEZ, a. s., is set for both sites (Dukovany and Temelín) and includes requirements of the relevant IAEA Standards, IAEA Safety Guides (including SSG-48 “Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants”) and WENRA Safety Reference Levels. The way of quality assurance for the process of ageing management as required in Decree No. 21/2017 Coll., is defined and described in the document ČEZ_PG_0001 Operational Ageing Management Programme for NPPs . More information about the Overall Ageing Management Programme can be found in the report to the TPR [14-1].	Yes, active components are within the scope of the AMP.

Article 14	Pag.96	For the case of Specific equipment of group A. Please describe TLAA (Time Limited Ageing Analysis)Please develop in some detail what “recommend from PSA” is meant. / Graded approach is used on ageing management strategy. “graded approach is selected on the basis of the strategy defined for the care of SSC” Criteria for the selection of systems subject to the ageing management process are set out:- equipment recommended from the PSA according to Decree No. 162/2017 Coll.,	TLAA (Time Limited Ageing Analyses) documentation is defined in accordance with IAEA SSG-48, Chapter 5.64 and the internal methodology of ČEZ_ME_1031. TLAA lists for Dukovany NPP and Temelín NPP are included as free attachments in ČEZ_PG_0001 NPP Ageing Management Program.PSA recommendations: it is about equipment included in the Scope of the ageing according to the methodology based on the results of the regularly updated "Living" PSA. The selection criteria are described in the relevant methodology. They are based on a regular reassessment of the relative importance of the equipment included in the PSA model. There are several different measures of importance from the PSA results, but the size of FV (Fussel-Vesely), RAW (Risk Achievement Worth) and CCDP (Conditional Core Damage Frequency) are important for ageing management, and these are used for selection.A graded approach in reliability management:Reliability management must be provided in a graded manner according to the importance of SCC and the impact of potential failure on safety and production. For power plant SCC, the SCC category (SCC categorization) must be determined according to the approved methodology, which is based on the determination of the impact of the equipment on the fulfilment of system functions, considering the influence on the performance of safety functions, safe shutdown, and energy production. The result of the categorization is the division of SCC into equipment: critical, non-critical, and non-important. The chosen SCC maintenance strategy must take into account the category and specificity of SCC, current status, operating modes, working conditions, legislative and licensing requirements, expected reliability, and the required time to restore or termination of SCC operation, medium term assignment, and site concept.

Article 14	Page 85	The Report states that in the Czech Republic the so-called “Living PSA” is used. How the probability of equipment failure is calculated when this approach is used? What software is used in these calculations? How are repair and maintenance planned given this risk-monitoring?	The “Living PSA” concept has been adopted since the beginning of PSA projects at ČEZ NPP, since the late 1990s. Today's requirements of the regulator (SÚJB) require not only regular updating of probabilistic models, in the event of significant changes in the design and operation of the NPP, but also regular updating of used reliability data (preferably of course NPP-specific). The manner and scope of collecting this data and converting it to parameters used in parametric models of equipment unreliability to perform its function in PSA is described in detail in the relevant ČEZ managing documentation. The data are recalculated based on the raw data collected from the specific operational experience of the units at least every 5 years, in accordance with the requirements of the regulator (SÚJB). The software used for the processing, maintenance, and quantification of models is software WinNUPRA and WinNUCAP (PSA Level 2) at Temelín NPP and Risk Spectrum Pro at Dukovany NPP. The Safety Monitor software is used at both ČEZ nuclear power plants for the purpose of risk monitoring. The data collection system is designed for both ČEZ NPPs; it is described in the relevant methodology and is also followed. Special database systems developed by PSA department together with NRI Řež are used for data collection.Calculation of the risk profile before each outage or planned OLM (on-line maintenance) activity is performed using the Safety Monitor software at both power plants. During outage / OLM, the risk is monitored in terms of deviations from the approved outage / OLM activity schedule, and the actual course of risk is compared with the risk arising from the original outage / OLM schedule after the end of these activities. Based on this comparison, recommendations and measures for further outage / OLM activities are then defined, and the comparison is sent within the overall outage / OLM evaluation to SÚJB. Criteria are set for the achieved maximum immediate risk and the cumulative outage / OLM activity.The “Living PSA” concept has been adopted since the beginning of PSA projects at ČEZ NPP, since the late 1990s. Today's requirements of the regulator (SÚJB) require not only regular updating of probabilistic models, in the event of significant changes in the design and operation of the NPP, but also regular updating of used reliability data (preferably of course NPP-specific). The manner and scope of collecting this data and converting it to parameters used in parametric models of equipment unreliability to perform its function in PSA is described in detail in the relevant ČEZ managing documentation. The data are recalculated based on the raw data collected from the specific operational experience of the units at least every 5 years, in accordance with the requirements of the regulator (SÚJB). The software used for the processing, maintenance, and quantification of models is software

			WinNUPRA and WinNUCAP (PSA Level 2) at Temelín NPP and Risk Spectrum Pro at Dukovany NPP. The Safety Monitor software is used at both ČEZ nuclear power plants for the purpose of risk monitoring. The data collection system is designed for both ČEZ NPPs; it is described in the relevant methodology and is also followed. Special database systems developed by PSA department together with NRI Řež are used for data collection.Calculation of the risk profile before each outage or planned OLM (on-line maintenance) activity is performed using the Safety Monitor software at both power plants. During outage / OLM, the risk is monitored in terms of deviations from the approved outage / OLM activity schedule, and the actual course of risk is compared with the risk arising from the original outage / OLM schedule after the end of these activities. Based on this comparison, recommendations and measures for further outage / OLM activities are then defined, and the comparison is sent within the overall outage / OLM evaluation to SÚJB. Criteria are set for the achieved maximum immediate risk and the cumulative outage / OLM activity.

Article 14	p. 81-100	Could you, please, clarify, are there any documented procedure exists for guiding the graded approach using numerical values for the SSC, including OE feedback, reliability?	The methodology presented in Article 14.2 verification of Safety, generally in the article 14.2.2 programmes for the continuous verification of safety, use the graded approach based on the SSCs screening and categorisation according its importance.The procedure described is the tool of the licensee for continuous management and care of its assets, especially the SSCss with impact to safety and also the equipment important for the production of electricity. The application of graded approach, required and recommended also by nuclear legislation, is based on two goals :- optimisation of acceptable irradiation risk- effectiveness of the plant exploitation.Assessment of safety importance, recommended in the IAEA Safety Guide SSR-30 was planned for this purpose. Nevertheless, the operating organisation finally decided to develop its own methodology following the basic principles of the IAEA Guide, but without comparison of common numerical methods and numerical criteria. The main reasons for this decision is that most SSCs are dedicated for the fulfilment of several operational or safety functions in several different plant states and regimes. The acceptance criteria for successful execution of safety function differs also for different levels of defence in depth in correspondence to optimized radiation risk.Due to this fact, only engineering judgement methodology supported by results of deterministic safety analysis is used for screening and categorisation of the of SSCs. The marking the factor for the seriousness of the function failure is evaluated as low, medium, or high (made by engineering judgement). Another factor is related to the necessity of reaching a controlled (subcritical) state (A) or maintaining a safety state (B). The group A covers the systems the function which is initiated automatically after a postulated initiating event, or according the Emergency Operating Procedure. A combination of factors is used for the categorisation of SSCsNevertheless, this approach is important in general for the scope and intensity of measures, applied for reaching SSC characteristic compliance with technical specifications done by design during the whole lifetime of the Plant. The values of technical specifications of an individual SSC, carried out by technical specification, are independent of this graded approach in general.

Article 14	p. 81-100	Could you, please, clarify, how do you take into account the aging effects in PSA model of the NPPs?	Effects of equipment ageing, i.e. systems, structures, and components (SSC) on reliability, resp. initiation event rates, are not simulated in PSA as special ageing unreliability models. This is because component unreliability statistics are updated regularly within Living PSA (at least every 5 years). This statistic also automatically reflects the current failure rate, thus including the effect of ageing on the unreliability of the device as well as the eventually increased initiation events rates (due to increased failure rate) due to the ageing of the device. No special models of equipment unreliability due to its ageing are used in the PSA of Temelín NPP and Dukovany NPP.
Article 14	page 96, List of systems, structures and components subject to ageing management process, bullet 5	According to the first criterion of your methodology for selection of systems and components for ageing management, SSCs of safety class 1, 2, and 3 fall within the scope. What are the probabilistic criteria to determine the safety class 4 equipment, selected through the PSA?	The requirement for selecting systems, structures and components subject to the ageing management process is generally defined in Decree No. 21/2017 Coll. The following should be included in the selection of systems, structures, and components subject to the ageing management process:- Selected equipment; and- Systems, structures and components relevant to nuclear safety, which are not the selected equipment.In addition, according to the requirements of the Decree No. 162/2017 Coll., the results of the probabilistic safety assessment shall be used to verify the scope of SSCs subjected to the ageing management process.In the Ageing Management for NPPs, criteria for selecting equipment subject to the ageing management are set out in ČEZ_PP_0404.The identification of equipment falling within the scope of AM is based on the core group of all equipment registered in the plant's equipment register (the EAM Asset Suite system is now being used). The following is selected of that group for the purposes of AM: 1) All selected equipment under Decree No. 132/2008 Coll. (equipment with the assigned safety class 1, 2, 3)2) Equipment with the criticality level 1 and 2 assigned under ČEZ_ME_0608 and safety function of categories 1 and 2 important to nuclear safety (under ČEZ_ME_0901 3) Equipment recommended from the PSA4) Other equipment recommended on the basis of global good practice and operating experience. Safety class 4 equipment are devices selected according to the methodology based on results of the regularly updated "Living" PSA. The selection criteria are described in the relevant methodology. These criteria are based on a regular reassessment of the relative importance of the equipment included in the PSA model (in equations of minimum critical cuts – MCS [minimal cut set]). There are a number of different measures of importance from the PSA results, but the size of FV (Fussel-Vesely), RAW (Risk Achievement Worth), and CCDP (Conditional Core Damage Frequency) parameters is important for lifetime management; and these are used for selection.

Article 14	page 90, section 14.1.2	Do you plan regulatory inspections on the basis of the information obtained from the risk monitoring system during plant normal operation? How do you define the topics and scope of the inspections if you use information from the risk monitoring system?	Regulatory inspections directly related to the risk monitor are carried out as part of the inspection "Adequacy and use of PSA". As part of this inspection, for example, an inspector may carry out an independent assessment of an operational event that is interesting from the risk point of view (and which the licensee assessed using the risk monitor in the context of the permanent risk evaluation of the NPP operation). Or, an inspector can independently check some time segment from an overall, for example, annual, risk profile. The purpose of this type of inspections is to check whether the use of the risk monitor is carried out in accordance with the licensee's internal documents, i.e. in particular, whether the input data for the risk monitor calculations are correctly selected, sorted, and validated, whether the control calculation carried out with the participation of the inspector confirms the results obtained by the calculation previously carried out by the licensee, whether they respect the acceptable risk criteria, etc.The data obtained from the permanent risk profile evaluation using the risk monitor is not fundamentally used in planning controls. This tool can be used, for example, in the authorization holder's assessments of the significance of operational events for feedback controls. When planning inspections, SÚJB uses some types of main PSA results, namely FC and RIF import rates for components or systems. These results are presented in the internal document VDS 008 “Planning, Implementation and Evaluation of Control Activities at Nuclear Installations”, Annex 4. The purpose of this is to allow inspectors to focus on safety-relevant installations.

Article 14	page 95, section 14.2.3	Are there any specific regulatory requirements related to:- updating the aging management program when new type of nuclear fuel is used;- the scope of assessment of the validity of existing analyzes of equipment aging when new type of nuclear fuel is used.	The use of a new type of nuclear fuel is being considered, in accordance with the Atomic Act as a “modification affecting the nuclear safety” and, among the other aspects, its impact on nuclear safety, technical safety, and security must be assessed before its implementation. The impact on activities relevant on nuclear safety (ageing management is such an activity) shall be also assessed. Since this type of modification is approved by the regulator, the application for approval shall be accompanied by the documentation on assessment of the modification impact on the documentation for the licensed activity (in this case, the licensed activity is the operation of the nuclear installation). The list of documentation for the licensed activity is included in the Atomic Act, including the ageing management programme. All the processes and activities shall be carried out in accordance with the requirements of the Decree No. 408/2016 Coll. on management systems. All the processes and activities shall also have their interfaces determined (like inputs and outputs of the processes and activities, important information etc.) so that the nuclear and technical safety, radiation protection, etc. is permanently ensured. The interfaces between processes and activities must be continuously monitored and documented to ensure the ability to achieve the objective of these processes and activities. This decree also includes the requirements on how the changes to the management system shall be carried out; these requirements are also applied to changes to the processes and activities in the management system. Finally, in Decree No. 21/2017 Coll. on the Nuclear Safety of the Nuclear Installation, the specific requirements on ageing management process and ageing managements programme are introduced. Within the ageing management process, the impact of ageing and the effects of degradation mechanisms on the SSCs within the scope of the ageing management programme shall be monitored and their trends predicted; this prediction may be influenced by the new type of the nuclear fuel.

Article 14	pages 86, 88	With reference to the current PSA level 2 (Dukovani and Temelin) the report states that "Relatively essential change was the change in definition of LERF, which currently includes all releases of radioactivity from the containment or reactor hall during outage exceeding 1% of the initial amount of Cs-137 within 10 hours from core damage or from fuel exposure in the spent fuel storage pool".Could you provide information about the regulatory requirements and the basis for this definition of LERF?	The developer of PSA Level 2 performed a broad survey of how L2 PSA risk measures are defined in the other countries. It was revealed that wide range of definitions of LERF is used in the world. Many of them are defined as a fraction of core inventory (as also SARNET, ASAMPSA-2or SSG-4 recommendation). Most often, Caesium or Iodine are used as a representative element indicating the size of release.In our case, Cs137 was chosen as a representative due to the long half-life of Cs137 (over 30 years). It has far more serious consequences for permanent degradation of the surroundings (soil contamination) than Iodine I131, which is decisively involved in the radiation burden of the population in the first days after an accident (half-life of about 8 days). At the same time, the relative leakage of Caesium is roughly represented by the relative leakage of Iodine, which will be predominantly in the form of CsI.Considering the timing of containment failure and release of fission products into the environment, early releases are defined up to 10 hours after core damage, respectively within 10 hours after fuel uncovering in the spent fuel pool. All other releases are considered as late releases. The main advantage of this definition is the fact that the beginning of the time interval (core damage or fuel uncovering in the spent fuel pool) is also an indication for control room operators to enter SAMGs and to alert the emergency situation classified as a radiation accident when people from the NPP vicinity have to be evacuated. The length of the time window (10 hours) is based on consultations with experts on emergency planning and on the presumption that off-site countermeasures can be taken, and that most people from NPP vicinity can be evacuated (or sheltered) within 10 hours (i.e. all the above countermeasures will be implemented by this time).The large early release thus defined covers approximately the release corresponding to the lower limit of 7. INES scale (Grade 7 Major Accident, major release of radioactive material with widespread health and environmental effects requiring implementation of planned and extended countermeasures).After extensive discussions, this approach was also adopted by the SÚJB and included in the legislation in force: in Decree No. 162/2017 Coll., § 2, letter (g) a large early release is defined as a release of more than 1 % of the initial total amount of Cs137 located in a nuclear installation within 10 hours of the declaration of the radiation accident.

Article 14	General	How are multiunit sites treated in regulatory reviews in terms of satisfying the safety goals/targets in your regulation?	The topic of Multi-Unit PSA/Site PSA has been monitored in the Czech Republic for about 5 years. Representatives of the ÚJV Řež participate in the activities of the IAEA, OECD NEA WGRISK and ETSON. This issue will also be discussed at the regular Nuclear Power Council (NPC) EPRI, which will take place this year and which will also be attended by a representative of the ÚJV Řež. So far, the SÚJB has not set any safety goals/targets for Multi-Unit sites, nor does it plan to set such targets yet. Therefore, it cannot assess their fulfilment/non-fulfilment in the frame of its activities. However, SÚJB closely monitors these issues. For example, the representative of the SÚJB also participated in the Technical Meeting on Multi-Unit PSA, which was organized by IAEA (October 2019). It is therefore likely that the SÚJB will deal with this issue in more detail in the future within its competences.
Article 14	General	Is there any safety goal/target (or planning to develop) for multiunit site NPPs in your country? If there is, is it CDF, FDF or LERF/LRF?	The topic of Multi-Unit PSA/Site PSA has been monitored in the Czech Republic for about 5 years. Representatives of the ÚJV Řež participate in the activities of the IAEA, OECD NEA WGRISK and ETSON. This issue will also be discussed at the regular Nuclear Power Council (NPC) EPRI, which will take place this year and which will also be attended by a representative of the ÚJV Řež.So far, the SÚJB has not set any safety goals/targets for Multi-Unit sites, nor does it plan to set such targets yet. Therefore, it cannot assess their fulfilment/non-fulfilment in the frame of its activities. However, SÚJB closely monitors these issues. For example, the representative of the SÚJB also participated in the Technical Meeting on Multi-Unit PSA, which was organized by IAEA (October 2019). It is therefore likely that the SÚJB will deal with this issue in more detail in the future within its competences.

Article 14.1	14.1.2, p.85	The SÚJB reviews and gives its opinion on the results of the PSR and annually monitors implementation of the measures defined during the PSR. The licence holder shall notify the SÚJB of any changes to the time schedule for PSR measures and shall discuss them with the SÚJB. Q: Can you explain how SÚJB reviews and gives its opinion on the results of the PSR?	The PSR results are reviewed by team of SÚJB inspectors who are involved in the assessment process of Final Safety Report and inspections of dedicated NPP. The team is named by the chairperson together with the specification of safety factors and subareas/criteria to be assessed by the inspector. The assessment results are recorded typically in tables where the number of safety factor (1÷14) and criterion is indicated, while the text of PSR original finding and the inspector´s comment/requirement is recorded. The table is attached to the letter prepared by the team coordinator. The letter is approved by responsible directors and sent to the representative of Utility.

Article 14.1	Page 90	Could “Safety Monitor” be followed by the regulatory body on its headquarter concurrently? What is the reference document to evaluate quality of NPP’s PSA for this application, IAEA TecDoc 1804 or other?	Permanent risk evaluation with risk monitor is a mandatory application of PSA from 2017; it is a requirement of Decree No. 162/2017 of Coll. on the requirements for a safety assessment by the Atomic Act.SÚJB regularly receives, every quarter of year, results of this PSA application. The evaluation of outages is also part of these documents.This PSA application is performed by the licensee in cooperation with TSO - ÚJV Řež (transformation of PSA model into SW Safety Monitor is performed by ÚJV Řež for NPP Dukovany, or by licensee for NPP Temelín).Risk monitors are available for the regulatory review on personal computers at resident inspector's offices in both Czech NPPs. Re-evaluation of interesting events is also the object of regulatory inspection “Adequacy and use of PSA”, which is performed every year.Safety Monitor cannot yet be used in the headquarter of the SÚJB. This is because the SÚJB does not have a purchased license to use the software environment (Safety Monitor™) that is needed to create the relevant PSA model, for its modifications and calculations. SÚJB has the option of remote access to certain applications that are authorized on the network of licensee (intranet). However, Safety Monitor is not an intranet application. It is software that must be installed on a personal computer or laptop of the licensee, according to the license terms.Safety Monitor should mainly be used by SÚJB resident inspectors on NPPs, because they have a detailed overview of the specific condition of the units they are in charge of and should therefore be able to evaluate the current operational risk of their units, independently of the licensee.In relation to this issue, two questions can be asked:§ A) If you are using risk monitor, do you compare the results of the original PSA model and Risk Monitor model? If so, how the results differ (e.g. in %)?§ B) Does the regulatory authority require this comparison of both models (for example in case, when you convert a new/updated PSA model into a risk monitor)?The Czech Republic's answers to these questions are as follows:A) Yes. An integral part of the work in converting the original PSA models into a risk monitor is also to check the consistency of the results of the original PSA models (from the Riskspectrum® PSA /NPP Dukovany/ or Win NUPRA /NPP Temelín/) and results of the corresponding risk monitor models using the validation and verification calculations. Comparative calculations are thus carried out from which it can be seen that the results obtained from both PSA models, using the same value of the cutset, do not differ by more than 5-10%. You can never get the same “identical” results comparing PSA and risk monitor model results. The reason for is that you are comparing different models. While the original PSA is dedicated to obtaining a more accurate number for average annual configuration, the risk monitoring model must cover a large number of expected

			unit configurations which are not contained in the PSA model. In addition, there must also be modelled, for risk monitoring purposes, an interface to other unit(s) and to other unit supporting systems which can be cross-tied with the current unit. Therefore, the comparison of MCS is required for both PSA and risk monitor models, but they cannot be identical by nature. Usually, a tenth of specific plant configuration calculations are to be checked and compared, and if some significant MCS differences appear, then they must be checked and fully explained.B) Yes, in general, that's the way it is. The SÚJB occasionally requests information (a report) regarding “converting the original PSA models into a risk monitor” and carry out its review.The suitability of the PSA models for the risk monitor application has been assessed several times (independent assessment of the PSA for both Czech NNPs, including suitability of the PSAs for PSA applications, initiated by SÚJB was carried out by Austrian company ENCONET Consulting in 2005, International PSA review as part of the IAEA TSR-PSA mission took place in 2016 at NPP Dukovany; all aspects of PSA for this NPP, including its use in applications have been assessed – for these purposes, the SSG-3 was used in particular).IAEA TecDoc 1804 has not yet been used for these purposes; it was issued recently, in 2016. Since then, an independent evaluation of this aspect of PSA has not been carried out. However, it is expected that a comprehensive assessment of PSA suitability for all possible applications will be carried out by SÚJB in the near future, according to this IAEA document.
Article 14.2	Section 14.2.5, page 99	Inspection activities are carried out by SUJB in the form of: planned inspection or unplanned inspection (the so-called “ad hoc inspection”). What is the frequency of scheduled inspections? What regulation defines the periodicity of scheduled inspections?	The frequency of inspections carried out by SÚJB on license holders can be seen from the Inspection Activity Plans published by SÚJB on its website. The frequency and period of inspections is set out in the “SÚJB Basic Control Program on Nuclear Installations for the Stage of Operation”, which is Annex No. 2 of the SÚJB internal control document, “VDS 008 - planning, implementation and evaluation of control activities at nuclear installations”. As can be seen from the Plans of Inspection Activities, the planned inspections include routine inspections performed by SÚJB site inspectors at nuclear facilities and specialized inspections performed by system specialists. The number of planned inspections within a one-year period may vary, in particular depending on how many nuclear power plant units are scheduled for a refuelling outage in a given calendar year.

Article 14.2	Ageing management, long-term operation	Is the design lifetime 30 years of Dukovany NPP the time from the start of operation or the actual efficient operation time (excluding the outage period)?	The originally established design lifetime of 30 years was the time from the beginning of the operation (outages were included).
Article 14.2	Ageing management, long-term operation	The report said that Group A is trying to take measures the aging management when standard methods of preventive maintenance cannot be applied. What kind of equipments do you expect to be excluded from this category (Group A)?	Specific facilities (Group A) – these are facilities for which ageing management must be ensured on the basis of the defined SCC maintenance strategy, using the specific or component Ageing management program (AMP) or TLAA analyses. In accordance with SSG-48 and the upcoming revision of SRS-57, the following may be excluded from this group: Components that are subject to documented periodic replacement or to a scheduled refurbishment plan based on predefined rules, e.g. an environmental qualification or manufacturer's recommendation, planned within the term of the licence, if the risk of ageing and inability to fulfil its safety related function is eliminated. This can be generalized for all short-lived components and consumables (e.g. seals, gaskets, filter elements) which are not considered in AMR since they are replaced before ageing can occur.

Article 14.2	Ageing management, long-term operation	In the Czech Republic, does the regulatory body require of some special approvals to extend the operating period of NPPs that have reached its design lifetime?	In January 2017, the new Atomic Act No. 263 of 2016 Coll. came in force. In this Act, requirements on ageing management are defined together with the list of documentation for the operational permit. The list of documents to be submitted for this permit include • Management System Programme• In-service Inspection Programme (specific approval is required)• Safety Analysis Report • List of selected equipment (specific approval is required)• Limits and conditions of the safe operation (specific approval is required)• Demonstration of safety - that plant equipment, personnel and internal documentation are prepared for further operation of the nuclear installation• Ageing Management Programme, etc.In the event of intended operation beyond the design lifetime, special safety assessment shall be performed in accordance with Section 23(3) of Decree No. 162 of 2017. Requirements (such as SSCs ageing rate, reliability of SSCs, fulfilment of acceptance criteria, validity of TLAAs, etc.) on scope of this assessment are defined in the Section 23(3) of this Decree. According to the Section 23 (4), this special safety assessment shall be conducted 24 months prior to the end of design lifetime. Documentation of this assessment is specified in Section 25 (1) and (2) of Decree No. 162 of 2017. As stated here, among other, general information, the following information shall be included in the documentation:- list of SSCs influencing nuclear safety- ageing management process results - results from assessment of reliability of SSCs influencing nuclear safety- results from TLAA validity assessment- list of modifications from the start of operation- time schedule for the following operation.In addition, the ageing of SSCs and other aspects closely related to LTO are also assessed within the PSR framework, in accordance with Section 13 of Decree No. 162 of 2017. To resolve the issues arisen from previous PSR is the precondition for obtaining new operational license.Other relevant regulations are:- Decree No. 358/2016 Coll., on Requirements for Assurance of Quality and Technical Safety and Assessment and Verification of Conformity of Selected Equipment and- Decree No. 21/2017 Coll., on Assuring Nuclear Safety of Nuclear Installations, where requirements for ageing management program and ageing management process are specified.Finally, proof of meeting previously defined conditions for the operation shall be submitted as well. In the case of Dukovany LTO application, all the above mentioned information was submitted to the regulator as Summary Evidence of the Readiness of Dukovany NPP Unit 1 to 4 for LTO (separately for the units).To help the stakeholders to implement the legislative requirements into the practice, regulatory safety guides have been issued. They are not legally binding. For commercial nuclear installations with a power reactor,

			recommendations are given in Regulatory Safety Guide BN-JB-2.1 “Ageing Management for Nuclear Power Plants” (now under revision to include results from the IAEA guide related to ageing management - SSG-48 “Ageing Management and Development of a Programme for the Long Term Operation of Nuclear Power Plants”).
Article 15	Page 104	Could Czech Republic give information about the organization of the licensee to ensure the radiation protection in the NPP site?	There are four departments under radiation protection (see attached file 104_RP_organization): Dukovany NPP and Temelín NPP Radiation Protection Depts. – main activities: 24/7 operation, day-to-day business, workplace monitoring; Radiation Risk Management department – main activities: Health Physicists; and Radiation Protection Laboratory department – main activities: Personal Dosimetry, Effluences and Environmental Monitoring, Metrology.
Article 15	15.3.3. Pag. 106	Regarding with the discharges into the environment, are the authorised dose limits set for the site or for each unit at the site?	The authorised dose limits are set for the site.

Article 15	15.3.3. Pag. 106	Please, could you inform on the location of the representative person, the consumption rates and the exposure pathways considered in the calculation of the doses?	For aerial discharges, the location of the representative person depends mostly on the weather (wind direction) during the year. As a rule, it is in the vicinity of the NPP (up to 5 km from the NPP).For the discharges into surface waters, the location is at a village on a riverbank down the river stream usually several kilometres from the discharge location.Consumption rates are prescribed in the Czech legislation for the age group (0-5 / 6-15 / over 15 years) namely: breathing rates are 1500 / 6500 / 8500 m3/y, drinking water consumptions are 275 / 365 / 730 litres/y.Calculations are performed with the following exposure pathways:Atmospheric pathways: irradiation from cloud, irradiation from deposit, inhalation of contaminated air, ingestion of food contaminated by atmospheric fallout. Hydrological pathways: swimming and boating, activities on contaminated river shores (e.g. fishing, sunbathing), staying on watered ground (e.g. gardening), ingestion of contaminated water, ingestion of food watered by contaminated water.
Article 15	15.4. Pag. 107	According to the text, the operator submits part of the samples taken directly to SURO laboratories for analysis. Please, could you inform on the required detection limits for the main radionuclides analysed at the SURO laboratories?	The required detection limits for the main radionuclides are set as per Decree No. 360/2016 Coll. Those values were transposed from the Commission Recommendation 2004/2/Euratom. The detection limits should be met by the SURO laboratories as well as by the NPP laboratories. In practice, better detection limits are achieved for some of the radionuclides.

Article 15	15.4. Page 107	According to the text, the operator submits part of the samples taken directly to SURO laboratories for analysis. Please, could you inform on the required detection limits for the main radionuclides analysed at the SURO laboratories?	The required detection limits for the main radionuclides are set as per Decree No. 360/2016 Coll. Those values were transposed from the Commission Recommendation 2004/2/Euratom. The detection limits should be met by the SURO laboratories as well as by the NPP laboratories. In practice, better detection limits are achieved for some of the radionuclides.
Article 15	15.3.3. Page 106	Please, could you inform on the location of the representative person, the consumption rates and the exposure pathways considered in the calculation of the doses?	For aerial discharges, the location of the representative person depends mostly on the weather (wind direction) during the year. As a rule, it is in the vicinity of the NPP (up to 5 km from the NPP). For the discharges into surface waters, the location is at a village on a riverbank down the river stream usually several kilometres from the discharge location. Consumption rates are prescribed in the Czech legislation for the age group (0-5 / 6-15 / over 15 years) namely: breathing rates are 1500 / 6500 / 8500 m3/y, drinking water consumptions are 275 / 365 / 730 litres/y. Calculations are performed with the following exposure pathways: Atmospheric pathways: irradiation from cloud, irradiation from deposit, inhalation of contaminated air, ingestion of food contaminated by atmospheric fallout. Hydrological pathways: swimming and boating, activities on contaminated river shores (e.g. fishing, sunbathing), staying on watered ground (e.g. gardening), ingestion of contaminated water, ingestion of food watered by contaminated water.
Article 15	15.3.3. Page 106	Regarding with the discharges into the environment, are the authorised dose limits set for the site or for each unit at the site?	The authorised dose limits are set for the site.

Article 15	15.4	Reference section 15.4 which states that "The SÚJB ensures its own independent monitoring of discharges and vicinity of the workplace". Czech Republic may like to share the resolution process followed in case a remarkable difference is observed between licensee and SÚJB results.	SÚJB regularly compares the NPP results with the independent ones. Usually, the results are in a very good accordance. If a remarkable difference was observed, the cause would be searched for. The next step would depend on the cause.
Article 15	Section 15.3.3, page 106	From the information presented in the section it follows that the annual effective dose from NPP discharges and releases for the control group of the public is at low level. Questions: 1. What radionuclides contribute to the annual effective dose from NPP discharges and releases for the control group of the public? 2. What is the contribution of each controlled radionuclide discharged and released into the environment to the annual effective dose for the control group of the public ?	The radionuclide set that we measure in Temelín NPP and Dukovany NPPs is defined by a regulator approved monitoring program, and it is based on Commission Recommendation 2004/2/Euratom, Annex I, Table A.1 for discharges to atmosphere and Table A.2 for liquid discharges. Contribution of measured radionuclides to annual effective dose: For NPP Temelín and discharges to atmosphere: C-14 60.2%, Ar-41 22.1%, H-3 15.7%, Xe-133 0.7%, Xe-135 0.6%. For Temelín NPP and liquid discharges: H-3 97.9%, Cs-134 1.5%, Cs-137 0.6%. For Dukovany NPP and discharges to atmosphere: C-14 57.8%, Ar-41 38.9%, H-3 2%, Co-60 1%. For Dukovany NPP and liquid discharges: H-3 88%, Co-60 11.4%. Other radionuclides contribute less than 0.5%.

Article 15	Page 104, para 2	According to the Euratom Directive 59/2013 the annual equivalent eye lense dose was reduced to 20 mSv (50 mSv in a single year or 100 mSv in five years period). How is the equivalent eye lense dose monitored in your country? Do you use special Hp3 dosimeters (or Hp(0,07) or Hp(10)) for monitoring of the eye lense dose or do you use dose assessment according to the workplace monitoring results? If the eye lense dose is measured with dosimeters what principles are used to select workers for that (e.g., individual annual dose should exceed 0.3 of the annual dose limit of 20 mSv, or special workplace conditions)? Which period of monitoring is selected? Where do you store the results of monitoring of workers eye lens dose (National dose registry or..)?	The equivalent eye lens dose is estimated from the whole-body dosimeter worn at the reference point (left side of the chest) in most cases. In specific conditions, when this method can not give sufficiently precise results regarding whether the eye lens limit was or wasn't exceeded (e.g. in non-homogeneous radiation field), special eye dosimeters placed near the eye are used. The selection of workers who should wear the eye dosimeter is based on information about their typical annual personal doses and about the radiation field. The use of protective equipment (protective goggles, shields, etc.) shall also be taken into account. Workplace monitoring results are used for determining whether the radiation field is homogenous or non-homogeneous.The eye lens monitoring interval is set to one month.State Office for Nuclear Safety stores the results of the lens dose monitoring in the National Register.

Article 15	Page 104	It is mentioned that 'General limit for exposed workers is 100 mSv for the equivalent dose in the eye lens over any 5 consecutive years.'Can Czech Republic share whether eye lens dosimetry is implemented for individual workers at the NPPs? Furthermore, can Czech Republic provide information on the important findings from this programme with respect to compliance to the eye lens dose limits?	Lens of the eye dose routine monitoring is not implemented at the Czech NPPs. The most accurate method for monitoring the equivalent dose to the lens of the eye would be to measure the personal dose equivalent Hp(3) with a dosimeter worn close to the eye. However, in a homogeneous radiation fields, a whole-body dosimeter worn on the chest provides a good estimate both of the effective dose and the eye equivalent dose. According the actual plants' radiation field characterisation, the whole-body dosimeter and effective dose are considered to be adequate to limit the exposure of a worker at the plant.
Article 16	Summary p.13	Could the Czech Republic indicate how the National Radiation Emergency Plan is addressing the development of harmonized approaches for cross-border emergencies with Austria and Slovakia?	The National Radiation Emergency Plan is currently being developed, but it does not specifically address these relationships. However, the Czech Republic has concluded bilateral government agreements on cooperation and assistance in disasters, natural disasters, and other extraordinary events with Slovakia, Austria, Germany, Poland, and Hungary. The approach for dealing with cross-border emergencies would therefore be based on these bilateral agreements.

Article 16	Pag. 109	Could you explain on what occasions you do exercises that involve the activation of the following plans?• On-site Emergency Plan.• Off-site Emergency Plan.• National Radiation Emergency Plan.Do you do Exercises that involve the three plans at the same time?Who is the responsible for informing the population affected by ionizing radiations, specifically, what responsibility has?• The licence holder.• The Fire Rescue Service.• The President of the Region.• The Government of the Nation.In what way do they plan exercises in which put in practice the needed of informing the population? Could you display the outcome of some exercise in which the practice of informing the population was done?	The method and frequency of verification of the emergency plans, intervention instructions, and the emergency rules for category IV workplace are set out in Decree No. 359/2016 Coll., on details of ensuring radiation extraordinary event management. Verification must be performed in the form of an emergency exercise, including the on-site emergency plan and the intervention instructions, where:a) a radiation incident can occur, which shall practise all intervention instructions in a period of two consecutive calendar years;b) a radiation accident can occur, which shall practise all intervention instructions in a period of three consecutive calendar years. The method and frequency of verification of the off-site emergency plan is set out in Decree No.328/2001 Coll., on some details of ensuring of the integrated rescue system. According to this decree, the off-site emergency plan must be verified in the form of an emergency exercise once per three years at minimum.According to Decree No. 359/2016 Coll. on details of ensuring radiation extraordinary event management, the efficiency and consistency of the on-site emergency plan, off-site emergency plan, and the National Radiation Emergency Plan shall be checked by joint practising the scenario for a radiation accident in a nuclear installation or a category IV workplace, which has the emergency planning zone established and which is included in threat category A or B, once every four calendar years, and shall be evaluated afterwards. As for the question of informing – the president of the region (according to Act No. 263/2016, Coll., the Atomic Act, § 224) is responsible for informing the population in the emergency planning zone. The president of the region shall:a) in the event of a radiation incident involving a suspected release of radioactive substances or ionising radiation outside the nuclear installation grounds or the premises of a workplace using sources of ionising radiation, or in the event of a radiation accident within the territory of the region, immediately provide information, within the scope of his competence defined in other legislation, to the general public affected by this radiation extraordinary event about the facts of the radiation incident or radiation accident, the steps to be taken, and where necessary, measures for the protection of the general public to be adopted;b) cooperate with the Fire Rescue Service of the Czech Republic and municipal offices of municipalities with extended authorities to provide the information under point a)The licence holder (According to Act No. 263/2016, Coll., the Atomic Act, § 157) shall ensure a response to a radiation extraordinary event that has arisen in the course of the activities performed by them, in accordance with the relevant on-site emergency plan, emergency regulations or, if the on-site emergency plan is not drawn up, intervention instructions, specificallya) in the event of the occurrence or

			suspected occurrence of a radiation accident, in cooperation with the Fire Rescue Service of the Czech Republic, immediately start warning the general public in the emergency planning zone and ensure the immediate broadcast of the emergency information; the information shall include the instruction to take urgent protective action in the form of sheltering and application of iodine prophylaxis;b) in the event of a radiation accident, immediately inform the general public affected by this radiation accident about the facts and expected development of the radiation accident.The Fire Rescue Service (According to Act No. 263/2016, Coll., the Atomic Act, § 220) shall:a) in the event of a radiation incident or radiation accident, immediately provide information, within the scope of its competence defined in other legislation, to the general public affected by this radiation extraordinary event about1. the facts of the radiation incident or radiation accident,2. the steps to be taken, and3. where necessary, measures for the protection of the general public to be adopted;b) cooperate in the provision of information referred to in point a) with the president of the region and the municipal office of a municipality with extended authorities, in the event of a radiation incident where release of radioactive substances or ionising radiation outside the nuclear installation grounds or workplace using sources of ionising radiation is suspected or in the event of a radiation accident.The Government of the Czech Republic has no obligation to inform the population; however, it participates in management of the radiation extraordinary event within the involvement of the Central Crisis Staff, which is a working body of the Government designed to deal with crisis situations in Czech Republic.The Czech Republic has been conducting emergency exercises aimed at informing the public and which were organized by the Region. Furthermore, communication with the public is also practiced within the multi-level exercises which are carried out once per two years.

Article 16	Pag. 109	Could you explain on what occasions you do exercises that involve the activation of the following plans?• On-site Emergency Plan.• Off-site Emergency Plan.• National Radiation Emergency Plan.Do you do Exercises that involve the three plans at the same time?Who is the responsible for informing the population affected by ionizing radiations, specifically, what responsibility has?• The licence holder.• The Fire Rescue Service.• The President of the Region.• The Government of the Nation.In what way do they plan exercises in which put in practice the needed of informing the population? Could you display the outcome of some exercise in which the practice of informing the population was done?	The method and frequency of verification of the emergency plans, intervention instructions, and the emergency rules for category IV workplace are set out in Decree No. 359/2016 Coll., on details of ensuring radiation extraordinary event management. Verification must be performed in the form of an emergency exercise, including the on-site emergency plan and the intervention instructions, where:a) a radiation incident can occur, which shall practise all intervention instructions in a period of two consecutive calendar years;b) a radiation accident can occur, which shall practise all intervention instructions in a period of three consecutive calendar years. The method and frequency of verification of the off-site emergency plan is set out in Decree No. 328/2001 Coll., on some details of ensuring of the integrated rescue system. According to this decree, the off-site emergency plan must be verified in the form of an emergency exercise once per three years at minimum.According to Decree No. 359/2016 Coll. on details of ensuring radiation extraordinary event management the efficiency and consistency of the on-site emergency plan, off-site emergency plan and the National Radiation Emergency Plan shall be checked by joint practising the scenario for radiation accident in a nuclear installation or a category IV workplace, which has the emergency planning zone established and which is included in threat category A or B, once every four calendar years and shall be evaluated afterwards. As for the question of informing – the president of the region (according to Act No. 263/2016, Coll., the Atomic Act, § 224) is responsible for informing the population in the emergency planning zone. The president of the region shall:a) in the event of a radiation incident involving a suspected release of radioactive substances or ionising radiation outside the nuclear installation grounds or the premises of a workplace using sources of ionising radiation, or in the event of a radiation accident within the territory of the region, immediately provide information, within the scope of his competence defined in other legislation, to the general public affected by this radiation extraordinary event about the facts of the radiation incident or radiation accident, the steps to be taken, and where necessary, measures for the protection of the general public to be adopted;b) cooperate with the Fire Rescue Service of the Czech Republic and municipal offices of municipalities with extended authorities to provide the information under point a)The licence holder (According to Act No. 263/2016, Coll., the Atomic Act, § 157) shall ensure a response to a radiation extraordinary event that has arisen in the course of the activities performed by them, in accordance with the relevant on-site emergency plan, emergency regulations or, if the on-site emergency plan is not drawn up, intervention instructions, specificallya) in the event of the occurrence or

			suspected occurrence of a radiation accident, in cooperation with the Fire Rescue Service of the Czech Republic, immediately start warning the general public in the emergency planning zone and ensure the immediate broadcast of the emergency information; the information shall include the instruction to take urgent protective action in the form of sheltering and the application of iodine prophylaxis;b) in the event of a radiation accident, immediately inform the general public affected by this radiation accident about the facts and expected development of the radiation accident.The Fire Rescue Service (According to the Act No. 263/2016, Coll., the atomic act, § 220) shall:a) in the event of a radiation incident or radiation accident, immediately provide information, within the scope of its competence defined in other legislation, to the general public affected by this radiation extraordinary event about1. the facts of the radiation incident or radiation accident,2. the steps to be taken, and3. where necessary, measures for the protection of the general public to be adopted;b) cooperate in the provision of information referred to in point a) with the president of the region and the municipal office of a municipality with extended authorities, in the event of a radiation incident where release of radioactive substances or ionising radiation outside the nuclear installation grounds or workplace using sources of ionising radiation is suspected or in the event of a radiation accident.The Government of the Czech Republic has no obligation to inform the population; however, it participates in management of the radiation extraordinary event within the involvement of the Central Crisis Staff, which is a working body of the Government designed to deal with crisis situations in Czech Republic.The Czech Republic has been conducting emergency exercises aimed at informing the public and which were organized by the Region. Furthermore, communication with the public is also practiced within multi-level exercises which are carried out once per two years.

Article 16	Part 16.2.1., page 132	Is there any legislative basis for sms notification system mentioned in report? Does state participate on its implementation, or is it managed solely by licence holder?	General requirements are defined by the Atomic Act and its providing degrees, but the Licence holder is free to find their own solutions within this framework. According to Section 25, paragraph 1, letter f) and g) of the Atomic Act:“Licence holders and registered persons shall… f) monitor, measure, evaluate, verify, and record quantities and facts relevant to nuclear safety, radiation protection, technical safety, radiation situation monitoring, radiation extraordinary event management and security and retain and forward information about them to the Office, as well as participate in comparative measurements organised by the Office and take corrective action if the participation in comparative measurements is not successful;g) ensure appropriate instrumentation for the measurement of quantities referred to in (f)”According to Section 12, paragraph 3, letter d) of Decree No. 329/2017 Coll., on the requirements for nuclear installation design: “…[nuclear installation design shall] include the means and procedures for monitoring the incidence of location characteristics and provision of alerts about them…”A seismic monitoring system is installed on both NPP Temelín and Dukovany; its amount of seismic instrumentation is based on Section 7.4 of the NS-G-1.6.

Article 16	Section 16.1.1-16.1.3	Sections 16.1.1.-16.1.3 describe detailed requirements and functional diagram of radiation emergency management. Could you give more information about the boundary conditions based on which emergency plans were prepared. For instance, do emergency plans include conceptual decisions on post-accident management after stabilization of severe accidents? Such issues can include the possibility of transport and reprocessing of large volumes of highly radioactive liquid waste.	The issues concerning post-accident management after stabilization of severe accidents will be addressed in the National Radiation Emergency Plan, which is currently under preparation. In any event, dealing with these issues will require the involvement of multiple organizations, such as the Ministry of Industry and Trade (which refines the concept of the management of radioactive waste and spent nuclear fuel in the Czech Republic), Ministry of the Environment, Ministry of Agriculture, and other state as well as non-state organizations.

Article 16	Page 122	As observed from Fig.16.3 on structure of emergency organisation at EDU and ETE, the Emergency Command Centre (ECC) receives support from the Technical support centre, Off-Site Emergency Support Centre, Emergency Information centre and Off-Site Logistic Support centre.Can Czech Republic clarify whether there are regulatory requirements with respect to emergency response/ command/ control centers remaining functional even in case of extreme external events, exceeding the design bases of NPP.	Yes, there are regulatory requirements with respect to emergency response/command/control centres remaining functional even in case of extreme external events, exceeding the design bases of NPP. Decree No. 329/2017 Coll., on the requirements for nuclear installation design stipulates the requirements for the emergency response/command/control centres. The design of the nuclear installations shall ensure that the shelters or special premises, in which the emergency control centre and technical support centre are located, are permanently operable, including in the event of a complete power supply failure in the nuclear installation and under design extended conditions.
Article 16	General	What is the reference study (deterministic or Level 2 PSA) to identifying reference accident scenarios as a basis for emergency planning? If it is probabilistic analysis when do you expect to receive this study in the licensing process?	Accident scenarios and respective source terms are selected based on Level 1+ and Level 2 PSAs and cover all operating states of the unit, including reactor low power and shutdown states and refuelling. At the same time, they cover the full spectrum of initiating events, starting from events initiated by a combination of equipment failures and operator errors, to internal hazards such as fires and floods within NPP, ending to external hazards such as aircraft crashes, seismic events, or extreme weather conditions. They also consider all of the most important sources of radionuclides inside the plant – I.e. the fuel in the reactor core as well as fuel located in the spent fuel pool.

Article 17	17.1.5, p. 148-149	Could the Czech Republic please provide information about the current status of the extension of validity of the EIA-Statement for the Temelín site?	In December 2019, a request for the extension of validity of the EIA-Statement (including the documentation) was given by Elektrárna Temelín II (a subsidiary wholly owned by ČEZ, a. s.) to the Ministry of the Environment. The documentation is currently being checked. We can not estimate the term of extension or non-extension.
Article 17	17.1.5, p. 148-149	The National Report states that the EIA procedure for the Dukovany NPP is completed by issuing the relevant opinion of the Ministry of the Environment. Could the Czech Republic please provide information regarding conditions set in the binding statement of the Czech Environmental Ministry?	There are 47 conditions in the binding statement. 25 conditions will be applied to the stage of the project preparation (forest protection, protection of Nature 2000 site, rain water drainage system, construction solutions of emergency shelters, limitation of liquid effluents, water balance calculations, luminous pollution, urbanistic and architectural solutions, optically shielding, dendrological survey, road infrastructure, nuclear safety, monitoring of climatic conditions, water protection, operation synchronous with EDU1–4, acoustic study). 14 conditions will be applied to the project implementation stage (noise measurements in areas affected by construction-related transportation, using railway transport, deforestation, eliminate dustiness, construction organization principles concerning minimization of noise load impacts and ground water impacts, giving information about the project preparation, environmental [biological] supervision, floristic and faunistic surveys, non-indigenous and invasive plant species, protection of the chapel situated at the site of the extinct Lipňany village). There are 4 conditions to the new nuclear source (hereinafter “NNS”) operation stage (carrying out health assessments of population in the distant exposed area, regularly informing the public about the environmental impacts of the NNS operations, minimum residual flow rate in the Jihlava–Mohelno Downstream profile, rainfall water trapped in retention tanks will be discharged gradually). The last 4 conditions are to the topic of the NNS environmental impact analysis and monitoring (noise level measurements, implement additional anti-noise measures if necessary, discharge of the Jihlava River is monitored annually in terms of physical and chemical parameters, rainfall water drained from the NNS site is regularly monitored).

Article 17	17.1.5, p. 150	According to the National Report the updated version of the Initial Safety Analysis Report for the new Nuclear Units 3 and 4 of the Temelin NPP was submitted for review of the compliance with the new legislation to SÚJB. Could the Czech Republic please elaborate on the findings of the review? Was the review restricted to siting aspects?	The most important requirement for processing the updated Initial Safety Analysis Report (ISAR) was to harmonize the complete system and documentation under the new Atomic Ac.On the basis of this requirement, the revised documentation was submitted to the SÚJB at the end of November 2018, in the structure and with the content according to the requirements set out in Annex 1 to the Atomic Act and in accordance with the requirements of its implementing regulations. SÚJB reviewed these documents, and after regular proceedings and modifications on the licensee side, the SÚJB representatives approved the compliance of these documents compliant with Czech legislation in force in accordance with the transitional provisions of the Atomic Act. The site of Temelín NPP has been reviewed in detail during the preparation of Initial Safety Analysis Report (ISAR) before 2012 and particularly within the licensing process in 2012-2014. With regard to the ongoing evaluation of site characteristics (ongoing surveys e.g. of tectonics, seismicity, and circulation of groundwater which SÚJB constantly monitors) and the updated Operational Safety Analysis Reports which are submitted every year (since 2018 as required by the New Atomic Act and its implementing decrees), the updated version of the ISAR for the New Nuclear Units 3 and 4 of the Temelín NPP do not show any new results as for siting aspects.

Article 17	Page 148, section17.1.5, paragraph 4	The report states that construction of new nuclear units in the territory of the Czech Republic is desirable and an environmental impact assessment (EIA) was conducted for the Temelin NPP site and an EAI is being performed for Dukovany NPP site. Do you have a timeline for licensing of the plant and expected construction?	Dukovany II siteThe binding opinion on the Environmental Impact Assessment (EIA) was issued on 30 August 2019. In the I. half of 2020, ČEZ, a.s. will apply for a siting license; according to the Atomic Act, the application is submitted to SÚJB. This will be followed by the procedure for the siting of the building under the construction law (2021). The tender for the vendor (delivery model for Dukovany NPP is to be finalised in 2020) is expected to be announced at the end of 2020 or at the beginning of 2021. Construction is expected to start in 2029. Unit 1 is scheduled to be commissioned in 2036.Temelín II siteThe binding opinion on the Environmental Impact Assessment (EIA) was issued in 2013. It is extended until 2020. In December 2019, ČEZ requested the Ministry of the Environment (MoE) of the Czech Republic for an extension. The siting license according to the Atomic Act was issued by SÚJB and is valid until 2020. ČEZ is applying for its extension for an indefinite period.As for the nuclear new-build projects in the Czech Republic, Dukovany has the priority within the timeline, according to the Resolution of the Government of the Czech Republic of 8 July 2019 No. 485.
Article 17.1	17.1.2, p.143	Could you please indicate parameters (aircraft weight, speed, etc.) that you used to calculate the aircraft crash? / In the national report, there is reporting of measures to reduce the impact of aircraft crash. The underlying load assumptions, however, are not specified.	The designation of the design aircraft is based on the requirements of the Czech legislation, specifically from Decree No. 329/2017, Section 11, paragraph 4, letter b. The unified methodology was elaborated to determine the threat to the nuclear installation site in the Czech Republic by unintentional aircraft crash. Based on this methodology, design aircraft for the Dukovany NPP and Temelín NPP sites were determined for which the project's robustness was demonstrated. For the Temelín NPP, an aircraft with the following parameters was determined: weight 7 tons, speed at the moment of impact 100 meters per second, an angle of impact 45 degrees For EDU, the aircraft is of the following parameters: weight of 2 tons, speed at the moment of impact 100 meters per second, angle of impact 45 degrees.

Article 17.1	17.1.4, p.141	Is an automatic reactor scram system in place for earthquakes? If so, would it be possible to provide a description of the system and criteria including thresholds for an automatic reactor scram? What are the criteria (including thresholds) for a manual reactor scram due to earthquakes? Are there any guidelines for NPP response to earthquakes? If so, what are the specific guidelines? / NPP response to possible earthquakes is not described in the report	The Temelín NPP has a seismic monitoring system (SMS) installed as part of the monitoring and diagnostic system (TDMS). It is a seismically resistant qualified information system, the outputs of which are introduced into the diagnostic stations and to the Fixed Alarm System of the control room and emergency control room of both main production units. The output of the seismic monitoring system is not introduced into the reactor trip system. In case of a seismic event, the reactor is not automatically shut down.The seismic monitoring system of Temelín NPP is equipped with four three-axis accelerometers (type AC-23); two accelerometers are located on the open ground, one is located on the bottom base plate of the reactor unit of the main production unit No. 1, and one is located on the containment internal of the main production unit No. 1.If the absolute surface acceleration on any of the four sensors of the seismic monitoring system exceeds the set threshold sensitivity (trigger), the Fixed Alarm System "START OF SEISMIC ACCELERATION RECORD" alarm will be activated on both main production units, and the seismic event recording in the seismic monitoring system begins.Trigger levels are set for sensors as follows:• 0.01 g for all sensor axes on the open ground• 0.01 g for all sensor axes on the bottom base plate• x = 0.015 g, y = 0.015 g, z = 0.045 g for sensor in the containmentThe seismic monitoring system automatically evaluates a seismic event; when the design earthquake level (0.05g PGA for Temelín NPP) is exceeded, it activates the Fixed Alarm System "DESIGN EARTHQUAKE LEVEL EXCEEDED" alarm on both main production units.The operating personnel responds to the alarm in accordance with the operating instructions for response to alarm signalling (TC016/2) by informing the shift engineer about a seismic event and entering the operating instructions for abnormal operation (TC006/1). According to this operating instruction, the control room operators will check and stabilize the main parameters of the unit, verify the validity of the alarm signalling of the seismic monitoring system, and inform the shift engineer about the level of seismic event and the state of the unit. Furthermore, if the level of the project earthquake is exceeded, the operators of the control room will activate the Emergency Response Board and the Technical Support Centre, and they ensure the inspection of the condition of the units before shutdown within 8 hours of the start of the seismic monitoring system recording. If the level of the project earthquake is not reached, the control room operators will ensure a quick visual inspection of the seismic damage indicators within 4 hours from the start of the seismic monitoring system recording.Depending on the level of the seismic event, the outcome of the inspections performed and the current state of the unit, the shift

			engineer then decides on the further operation of the unit, or on the variant of its shutdown.

Article 17.1	Page 141	It is stated that “The assessment of the tectonic conditions and the potential occurrence of a movement-capable fault in the Dukovany NPP site and in the area at the minimum distance of 25 km from the nuclear installation takes place on a continuous basis.”What are the criteria for a fault whether considered as capable fault?	Requirements and criteria for the evaluation of capable faults are determined by national regulations (Decree No. 378/2016 Coll., on siting of a nuclear installation) and are based mainly on SSG SSG-9 IAEA and specific geological conditions of Czech Republic (defined by national experts).All faults with the potential to be capable must be assessed at minimum distance of 25 km from the nuclear installation. The results of this assessment must be compared with “exclusionary criteria”. For a capable fault, our national regulation determines exclusionary criteria in Section 6 of the Decree:The characteristics of the crack of the site for a nuclear installation as a result of a fault, the achievement of which causes the siting of a nuclear installation to be prohibited, are:a) The occurrence of the motion or seismic capable fault or any other motion of the Earth’s crust which could cause any nuclear safety-reducing deformation of a nuclear installation, up to a distance of 5 km; orb) The formation of an associated fault on the site area of a nuclear installation.In any case, for fault assessment there is a list of the evaluation requirements in § 6 “Crack of the site for a nuclear installation as a result of a fault” of the Decree:(1) The assessment of the site for a nuclear installation in terms of its crack as a result of fault shalla) Evaluate the faults1. with evidence of past movement over the last 2.6 million years;2. with a documented occurrence of historical earthquakes or a group of the focuses of earthquakes directly linked to the fault; or3. In a structural relationship with a known capable fault meeting the conditions of point 1 or 2, where there is a significant probability that the displacement on the fault could cause a movement of the other or near the surface of the site for a nuclear installation;b) Make use of geological, geophysical or seismological data;c) Be carried out up to a distance of 25 km; andd) Include an assessment of1. The occurrence of slow deformations of the surface of the area, including faults that have not any geological effect but can be reactivated;2. The occurrence of linear topographic morphological features of the relief;3. The occurrence of sharp lithological boundaries;4. The occurrence of signs indicating mechanical deformation of rocks on tectonic lines, in particular crush zones, clay minerals, and saturation by water;5. The occurrence of instrument-recorded earthquakes or documented historical earthquakes; and6. Signs of the occurrence of a fault on the site area of a nuclear installation, in particular their increased permeability for the groundwater flow through the rock environment.

Article 17.3	Page 152	How the engineering-geological and geotechnical monitoring are correlated for the measurement of structural settlement?	The monitoring of geotechnical parameters of the nuclear power plants under operation is carried out by measuring the settlement of buildings affecting the nuclear safety (periodically once a year). The measurement is mainly used for confirmation of the proper foundation of the buildings. The new site investigations, such as geological drills and trenches, are mainly used for providing the additional evidences for the suitability of the selected site for the siting process of the new nuclear power plants.
Article 18	18.1.4, p. 159	The current and previous National Reports address the possibility of direct or indirect cooling of molten fuel during severe accidents for the NPP Temelín 1 and 2. Following Challenge No. 4 from CNS 2017, could the Czech Republic please provide the latest insights on this issue? What kind of analyses/experiments were done?	The license holder for NPP Temelín 1 and 2 performed an extensive set of analyses. Analyses confirmed the possibility of stabilization of the partially melted core before its major relocation to the pressure vessel bottom part in the case of sufficient water refilling. The analyses didn’t confirm the reasonable applicability of external vessel cooling (because the need of early start of water supplying, need of stable and sufficient water inlet and steam outlet which is extremely complicated without the application of flow deflector surrounding the pressure vessel). The analyses also didn’t confirm the reasonable applicability of refractory linings (core catcher) installation in the reactor cavity and adjacent area GA302. But the analyses of the rate of the containment basement melting through shows, that in case of corium cooling from the top in the reactor cavity and adjacent area GA302, the stabilization of the corium and prevention of the melting through is possible.Based on the results of the analyses, the license holder performed the modification of existing pressure relief valves to enable its remote controlling during a severe accident and is now also preparing the installation of another alternative direct molten fuel cooling system. This is the completely independent new diesel driven pump system dedicated for the corium cooling both in the in-vessel phase of a severe accident and during the ex-vessel phase. The diesel driven pump system is presently under the project preparation phase. Stabilisation of the corium outside the reactor pressure vessel is a challenge for all of the types of operated units, and therefore the extensive research activities are still running in the topic (e.g. the research project ROSAU) to confirm the existing strategies and solutions and to bring new recommendations and inputs for safety enhancements (mostly in the MCCI topic).

Article 18	18.1.4	Reference section 18.1.4 where it is stated that a transition to a new fuel with better mechanical properties took place. Czech Republic may like to share qualification process for new fuel.	ČEZ operates in Temelín NPP (VVER 1000) the TVSA-T fuel fabricated by company TVEL. The subject of the modification were changes intended to provide a closer grid span, strengthen the skeleton, and provide more effective thermo-hydraulic performance and the implementation of new fuel rods with enlarged outer diameter of fuel pellets without central hole (more uranium loaded in the fuel assembly). Other minor changes have been implemented as well. From the qualification and licensing point of view, current practice in Czech Republic is that the NPP Operator follows requirements and licensing procedure requirements of Czech Republic (which is being implementing legislation of European Union) and of country of origin, and takes into account IAEA requirements and appropriate world practice. The Operator in Czech Republic requires the condition that the upgraded nuclear fuel shall have sufficient operational experience. During the Development Program, the Fuel Vendor and Operator have performed needed tests, calculations, prepared licensing documentation, and finally obtained license from the Czech regulatory body (SÚJB). Independent calculations and reviews were performed. In addition, the out of pile fuel rod alloy program has started. Operational experience is required to be submitted to the regulatory body. The upgraded nuclear fuel was loaded to Unit 2 in 2018.

Article 18	18.1.2	Reference section 18.1.2, a set of analyses of extended design basis conditions has been included in the safety analysis report for Dukovany NPP in the last years. Czech Republic may like to share the detail of specific set of analyses related to extended design basis conditions included in the safety analysis report.	DEC AA set of DECs is derived and justified as representative, based on a combination of deterministic and probabilistic assessments as well as on engineering judgement.The selection process for DEC A starts by considering those events, and combinations of events, which cannot be considered with a high degree of confidence to be extremely unlikely to occur and which may lead to severe fuel damage in the core or in the spent fuel storage. Where applicable, all reactors and spent fuel pools on the site are considered. Events potentially affecting all units on the site are covered, as are potential interactions between units.Initiating events for DEC A:• initiating events induced by earthquake, flood or other natural hazards exceeding the design basis events;• initiating events induced by relevant human-made external hazards exceeding the design basis events;• prolonged station black out (SBO; for up to several days)- SBO (loss of off-site power and of stationary primary emergency AC power sources);- total SBO (SBO plus loss of all other stationary AC power sources), unless there are sufficiently diversified power sources which are adequately protected;• loss of primary ultimate heat sink, including prolonged loss (for up to several days);• anticipated transient without scram (ATWS);• uncontrolled boron dilution;• large reactivity insertion;• total loss of feed water;• LOCA together with the complete loss of one emergency core cooling function (e.g. HPI or LPI);• uncontrolled level drop during mid-loop operation or during refuelling;• total loss of the component cooling water system;• loss of core cooling in the residual heat removal mode;• long-term loss of active spent fuel pool cooling;• multiple steam generator tube ruptures ;• loss of required safety systems in the long term after a design basis accident.DEC BThe set of category DEC B events is postulated and justified to cover situations in which the capability of the plant to prevent severe fuel damage is exceeded or where measures provided are assumed not to function as intended, leading to severe fuel damage. For DEC B (severe accidents), an approach different from that for the selection of DEC A is taken, since there is a very large number of possible scenarios which cannot all be captured at the start of a selection process. A set of severe fuel damage scenarios is identified for analysis, covering the different situations and conditions which can occur at the outset and during the course of a severe accident. The selection process of representative scenarios uses the PSA results, the overall understanding of the physical phenomena involved, the margins in the design and the systems’ redundancy and diversity. For the practical elimination of scenarios leading to early or large releases, it is necessary to implement additional robust design and administrative measures. For practical purpose, the

			scenarios used for demonstration of practical elimination of early and large releases are grouped within the following categories:1. Events that could lead to prompt reactor core damage and consequent early containment failure:a. Failure of a large component in the reactor coolant system (RCS)b. Uncontrolled reactivity accidents2. Severe accident phenomena which could lead to early containment failure:a. Direct containment heatingb. Large steam explosionc. Hydrogen detonation3. Severe accident phenomena which could lead to late containment failure:a. Molten core concrete interaction (MCCI)b. Loss of containment heat removal4. Severe accident with containment bypass5. Significant fuel degradation in a storage pool
Article 18	18.4	Reference section 18.4, Czech Republic may like to further elaborate the problem faced due to which a need was felt for installation of a new hydrogen removal pipeline for pressurizer safety valve.	According to the IAEA Safety Issue S04 complex of pressurizer safety valve (Pressurizer relief valve, Main safety valve, Impulse control valve) is required to be qualified not only for steam flow, but also for steam water flow for the successful management of transition modes in emergency states (e.g. ATWS, Feed & Bleed) where these valves were opened, while their working medium would be saturated or slightly supercooled water. The prerequisite for this qualification (reliable and stable function of complex of pressurizer safety valve in two-phase flow) is to ensure the removal of hydrogen from the piping in front of these valves. Hydrogen is generated in the primary circuit due to radiolysis and thermo-chemical reactions and, due to its low density, it continuously accumulates at the highest points of the pipeline at the complex of pressurizer safety valve. For this reason, the new hydrogen removal pipeline was implemented.

Article 18	Major modifications implemented	According to the report, the fire extinguishing equipments had been switched from the water type to the powder type. Is the the powder fire extinguishing able to ensure the same performance as the water type?	Yes, it is. Its main extinguishing effect is anticatalic (inhibitive). The powder extinguishes quickly and safely. Its high efficiency lies in the fact that, at the time of the intervention, a large amount of small particles from 0.001 to 0.1 mm in size is produced from the powder mixture (whose exact composition is the property of the individual manufacturers). The reaction, which takes place on a large surface of such particles, takes a large amount of energy from the fire and thus destroys it.
Article 18	Major modifications implemented	According to the report, the safety valves of the pressurizer has been remodeled as a hydrogen countermeasure. Under what situation do you expect to be the countermeasure against hydrogen generated?	According to the IAEA Safety Issue S04 complex of pressurizer safety valve (Pressurizer relief valve, Main safety valve, Impulse control valve) is required to be qualified not only for steam flow, but also for steam water flow for the successful management of transition modes in emergency states (e.g. ATWS, Feed & Bleed) where these valves were opened, and their working medium would be saturated or slightly supercooled water. The prerequisite for this qualification (reliable and stable function of complex of pressurizer safety valve in two-phase flow), is to ensure the removal of hydrogen from the piping in front of these valves. Hydrogen is generated in the primary circuit due to radiolysis and thermo-chemical reactions and, due to its low density, it continuously accumulates at the highest points of the pipeline at the complex of pressurizer safety valve. For this reason, the new hydrogen removal pipeline was implemented.
Article 18	Major modifications implemented	The forced-draught towers including fans are used as the new ultimate heat sink. How is the reliability of its fans' the power supply ensured?	Two fan tower cells are installed for each of the three essentials service water divisions. Each cell is assigned to a different safety system division respectively to secured power supply 1, secured power supply 2, or secured power supply 3 in terms of control and power supply.

Article 18.1	p 70 ff	How does the Czech Republic identify Reasonably practicable or achievable safety improvements that are oriented to meet the objective of preventing accidents in the commissioning and operation and, should an accident occur, mitigating possible releases of radionuclides causing long-term off site contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions / It is noted in the CNS report 2020 that the state-of-the-art in science and technology should be considered. How is a deviation evaluated and assesed?	An identification of potential safety improvements is continuously performed by the license holder based on various inputs, mainly:- deterministic safety analysis and criteria;- probabilistic safety assessment;- periodic safety review;- benchmarks with other operators, evaluation of industry best-practice.These (and other) activities / sources provide inputs for subsequent feasibility analyses and design modifications. The modifications intended for implementation are summarized in the (periodically updated) plant Safety Improvement Programs (separately for Temelín and Dukovany). Such modifications are selected by the license holder based on the following basic criteria:- contribution from the point of view of safety objectives (accident prevention or mitigation, practical elimination of large and early releases, minimization of radiological consequences of accident scenarios);- effectiveness (how the selected means is physically effective in execution of the pertinent safety function);- simplicity and robustness, credibility of use in accident conditions (i.e. new systems intended for a specific defence-in-depth [DiD] level and their actuation and operation should not be more complicated than for the systems designed for lower DiD levels);- degree of independence on existing systems;- reasonable implementability.Evaluation of whether the proposed design (safety) improvements are reasonably practicable (achievable, implementable) is the responsibility of the license holder. The evaluation is based on the following basic criteria:- no negative impact on the systems intended to operate in previous DiD levels; this is a crucial criterion (e.g. new systems / means intended for BDBAs (DiD 3b,4) must not negatively influence the control of design base operational modes and operation of systems designed for design base operational modes [i.e. DiD 1,2,3a]);- the implementation shall not impose additional risks (e.g. radiological risks during implementation and subsequent maintenance, risk of damage of existing plant systems and constructions, …);- cost-benefit factors; even-though there is a methodology for cost-benefit evaluation, this criterion is practically not used by the license holder.The philosophy of continuous improvement of all DiD levels (in both the areas of design provisions and personnel / procedures) is adopted by the license holder.Regarding the last portion of the question, it is not clear what kind of deviation is meant. The license holder obviously has a standardized process for the assessment and resolution of deviations identified during surveillance and testing of plant safety systems, which is in compliance with the requirements of national legislation.

Article 18.2	p.160	"Incorporation of proven technologies": How does the authority verify the compliance of the licensee's new equipment related qualification requirements with the Atomic Act and the relevant industry standards? At what intervals are these documents reviewed?	An example of the Incorporation of new technologies is the spent fuel cask for the transportation, storage, or disposal of the nuclear spent fuel.The spent fuel cask is subject to type approval based on the manufacturer's application submitted to SÚJB under the Atomic Act. It includes documentation containing especially spent fuel cask description, technological procedures, test documentation, calculations, analyses and their independent verification. The documentation also includes the method of quality assurance of the spent fuel cask. SÚJB shall carry out the spent fuel cask type approval by decision. During the design, production, and assembly of the spent fuel cask, an authorized person performs an assessment by conformity with the technical requirements.SÚJB carries out an independent review of compliance with the requirements of the spent fuel cask type-approval documentation, including compliance with the requirements of the relevant legal regulations and technical standards. If there have been changes in the documentation or manufacturing variations, the manufacturer is obliged to submit the parts of the documentation that have been changed. It shall assess this documentation and decide whether to issue a new type-approval decision.
Article 19	19.5, p. 178-179	The technical support organization of SÚJB is established as part of the national Radiation Protection Institute. Following Challenge No. 1 from CNS 2017, the number of experts at the TSO has to be increased. Could the Czech Republic please provide information on the current status of the recruitment at the TSO? What are the changes since 2017?	In December 2016 , SÚJB adopted the “Strategy of building independent scientific and technical support of SÚJB for nuclear safety in the National Radiation Protection Institute”. The aim of this decision was the gradual establishment of a new Nuclear Safety Section in SÚRO v.v.i. so that SÚRO v.v.i becomes a fully operational TSO for SÚJB in 2021, covering all key areas of SÚJB's activities, in particular nuclear safety, technical safety, radiation protection, emergency preparedness, and security (physical protection). Currently, the Nuclear Safety Section in SURO v.v.i consists of three departments: - Department of nuclear safety assessment and research - Department of direct support of state supervision by SÚJB - Department for the safe waste management and decommissioning of nuclear installations At the beginning of 2017, the Nuclear Safety Section in SURO v.v.i. consisted of 3 FTE. At present it consists of 29 experts representing a total of approx. 14 FTE.

Article 19	Page 181, para 2	Who determines the level of the event on the INES scale and who needs to be informed when reaching the established level of the event on the INES scale (for example: inform the IAEA when reaching level 2 of the event on the INES scale)?	SÚJB has established a special internal INES working group to assess the events on the INES scale. In general, we inform our public about events ≥ INES 0 and IAEA (NEWS) about events preliminarily assessed as INES ≥ 2. As regards other possible events, there is no specific order, but we would act according to our international obligations. We have also a special arrangement with Austria to inform them about events (even preliminarily rated as) ≥ INES 1 that occurred at our Temelín NPP.
Article 19	Page 181-182	What is the status of the SNF storage facilities at Dukovany NPP and Temelín NPP and Dukovany RW Storage Facility (do they fall under the category of nuclear installation or are they covered by other licensing requirements)?	All AFR SF storage facilities in the Czech Republic are separate nuclear installations developed and operated on the basis of nuclear installation licenses. The same is true for RAW disposal facility at the Dukovany site. There is no separate RAW storage facility in Dukovany, and all predisposal installations are covered by NPP Dukovany licenses.
Article 19.7	19.7, p.183	In the context of the investigation of an operational event, a licensee shall evaluate the impact of the safety culture on an operational event, Q: How the licensee evaluates the impact of the safety culture on an operational event?	Each operational event must be analysed to determine: - Direct causes of the event - Obvious causes of the event - Safety culture assessment The assessment of the impact of the safety culture on the operational event is carried out according to ČEZ_ME_1096 Safety culture assessment and development. The Dukovany and the Temelín NPP feedback departments, which investigate individual events, usually assigned 5 positive or negative attributes of safety culture (from 40 attributes described in “Traits of a Healthy Nuclear Safety Culture” by INPO/WANO), especially in terms of behaviour and attitudes of persons involved in the origin, solving and subsequent investigation of events. Performance Improvement Specialists perform validation and, if needed, propose the correction of assigned attributes. At the Correction and Prevention Commission, the assignment of safety culture attributes is discussed and subsequently included in the commission report.

Article 19.8	Page 188	What is the final stage of the radioactive waste management? Disposal?	Yes, it is. All generated RAW is or will be disposed in disposal facilities which are in the operation or under development (geological repository). For further details, see national reports under JC.
Article 19.8	Page 188	How is radioactive waste immobilized after recycling?	Secondary radioactive waste resulting from solid radioactive waste treatment with the use of technologies available in facilities of external contractors outside the territory of the Czech Republic (dry solid waste incineration and melting of contaminated metal) are characterized, packed in approved radioactive waste packages, and disposed in the LLW repository without additional conditioning.
Article 19.8	Page 188	How is it planned to ensure the safety of the bitumen compound during long-term storage?	The stability of bitumen used for the fixation of liquid RAW has been monitored on a long-term basis at the Nuclear Research Institute Řež. Bituminous matrices and stabilizing additives with higher resistance to thermal oxidation ageing and incrust formation during the treatment of liquid RAW in bituminization facilities at NPP Dukovany and NPP Temelín are continuously being developed. The treated liquid RAW is continuously disposed in the low-level RAW repository. The stored samples of treated liquid RAW do not show degradation due to the presence of contained chemicals and ionizing radiation.