Expert Statement on the Environmental Impact Study (EIS) of the PAKS II NPP

Commissioned on Greenpeace Germany

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Content

1 Introduction

In the district of Tolna, close to the city of Paks, approximately 100 km south of Budapest, the only Hungarian Nuclear Power Plant (Paks NPP) is located on the right bank of the Danube river. On the site of Paks NPP with four reactors of type WWER-440/V213, two additional reactor units are planned to be built, which would generate 1,200 MWe each. An operation time of 60 years is envisaged. The commercial operation of the new units is scheduled for 2025 and 2030, respectively.

The competent Hungarian authority is the Authority for the Protection of the Environment, Nature and Water Management of South Danubia (Dél-dunántúli Környezetvédelmi, Természetvédelmi és Vízügyi Felügyelőség).

In April 2015, Hungary submitted the Environmental Impact Study (EIS), which was prepared in order to identify and evaluate the impact of the planned nuclear power plant technology on the environment.[1] The study was prepared by MVM ERBE ENERGETIKA Engineering Company Limited by Shares and its subcontractors, for the project company MVM Paks II. Zrt.

The independent nuclear expert Oda Becker was commissioned by Greenpeace Germany to prepare an Expert Statement on this EIS. The objective of the assessment was to investigate whether the information presented in the EIS is reliable and sufficient to determine the potential risks for other countries, in particular Germany.

This expert statement is to be submitted in the framework of the transboundary Paks II EIA procedure. It mainly focuses on technical issues. Although the evaluation does not claim to be exhaustive, it comes to the following conclusion:

The content of the EIS was found to not to be in line with the EIA Directive general requirements and IAEA specific recommendations[2]. Much additional information is necessary to assess the possible consequences of the Paks II for Germany. However, the information at hand indicates that a severe accident with a major release and consequences for Germany cannot excluded.

1 http://www.stmuv.bayern.de/umwelt/reaktorsicherheit/paks/

2 IAEA (2014): Managing Environmental Impact Assessment for Construction and Operation in New Nuclear Power Programs; http://www-pub.iaea.org/books/IAEABooks/10391/Managing-Environmental-Impact-Assessment-for-Construction-and-Operation-in-New-Nuclear-Power-Programmes

2 Reactor type

According to chapter 6 of the EIS, the VVER-1200 is an improved version of the VVER-1000 unit with a longer designed operating time (60 years), with a higher built-in capacity and with higher thermal efficiency. The chapter provides some information on the Russian Federation plans for constructing such units in Russia, Europe (Finland, Czech Republic, etc.) and other parts of the world (Turkey, Jordan, etc.). The background of the VVER-1200 design development is summarized, with its two versions MIR 1200 (St Petersburg design) and AES-2006 (Moscow design).

While not specifically mentioned in EIS, according to the MMV the selected technology is the AES 2006 – a WWER 1200/V-491. There are several NPPs under construction with such a reactor (V-491) and with the same designed safety features. The WWER-1200 (V-491) design was developed by "Atomenergoproekt" St. Petersburg (SPbAEP).[3]

Development of the reactor design

Design of the AES-2006 was started after the year 2000 and completed in 2006. Besides the increased size of 1,200 MW power, the AES-2006 plant has additional safety features as compared with the advanced VVER-1000 plants.

The following chart (figure 1) shows the evolution of the WWER-1000 with different reactors types[4] :

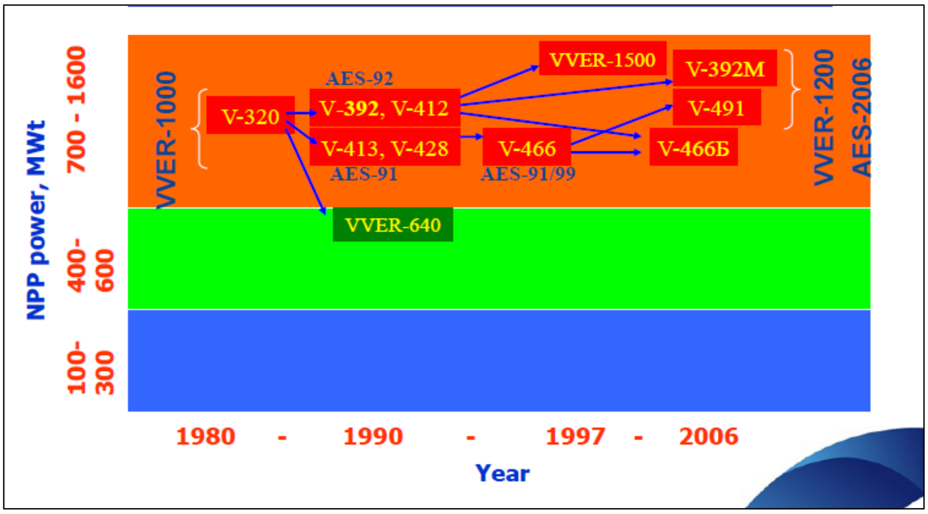

Figure 1: VVER Technology Evolution

There were two early post-Chernobyl designs, AES-91/V-428 (Saint Petersburg, t) exported to China and AES-92/V-412 (Moscow) exported to India. The reactors were essentially the same as its predecessor, VVER-1000/V-320), but with added safety systems including a core-catcher and some passive safety systems.

There are two variants of the VVER-1200: the V392M tends to depend more on passive safety systems, V491 more on active safety systems.

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3 http://www.mvmpaks2.hu/en/PaksII/TheFuture/NewUnits/Lapok/default.aspx

4 2010 GIDROPRESS (2010): Review of VVER-1000 and AES-2006; I.F. Akbashev, I.F.; Piminov, V.A.; et al.; Presentation; IAEA Technical Meeting on Irradiation embrittlement and life management of reactor pressure vessels; Znojmo (Czech Republic); 18–22 October 2010; www.iaea.org/NuclearPower/Downloads/Engineering/meetings/2010-10-TM-Czech/54.pdf

Two units of the AES-2006 / V-392M are under construction at Novovoronezh, and unit of the type V-491 are under construction at the Leningrad, Belarus, and the Baltic sites, although work at the Baltic site was suspended in 2013 and appears unlikely to restart. It is not clear which versions would be exported to the numerous export orders Rosatom claims but on which construction has not started, such as Turkey and Vietnam. There are differences between the two variants in terms of their passive safety systems.

Work on a successor design, VVER-TOI/V-520 developed by Moscow Atomenergoproekt and based on V-392M, quickly started and by 2010, it was said the new design would be available from 2012, although by 2015 no orders had been placed. VVER-TOI was expected to be 20 percent cheaper and could be built in 40 months.

Stage of design development of VVER-1200

Today, no nuclear power plant with the AES-2006 design is in operation. Currently, four units are under construction in Russia. They have been subject to construction delays.

Actually, the AES-2006 is "an as-yet untested design and the known incidents and deficiencies in the operation and construction of Russian-built NPPs provide evidence that Rosatom and its structures have serious problems of systemic nature and cannot guarantee the quality of their sites.

There are two units of reactor type VVER 1200/V-491 under construction at Leningrad II. Construction of unit 1 started in October 2008, and it was to be commissioned in October 2013. However, a section of outer containment collapsed in 2011 and, among others, set back the schedule. Commissioning is now expected to happen during 2018. The construction of the second unit started in April 2010, commissioning start is envisaged for 2020.[5]

It was only in 2014 that first reports of delays emerged and by 2015, all four reactors were 3–4 years late. However, a January 2015 report from Russia's Audit Chamber seemed to put the blame squarely on shortage of funds. Whether there are other construction issues is difficult to tell. Two reactors using an older design at the Rostov site were ordered at about the same time as the AES-2006s; one of these was completed on time and the other appears close to schedule. It may be that this indicates more deep-seated issues at Novovoronezh and Leningrad than just shortage of capital.

Delays during construction can be an indication that the detail design of the plant was not completed before start of work, and partly proceeded in parallel with construction. If the detail design process then does not advance as smoothly as planned, delays can result. This was illustrated by the EPR project Olkiluoto-3, for which startup has so far been delayed from 2009 to 2016 or even later, mainly due to problems connected to the design of the I&C system. It is not clear to which extent this applies to Leningrad-II.

In any case, when plants of the same type are to be built at sites in other countries, it is necessary to adapt the design to the characteristics of the site as well as to the regulations of the country in question. For example, Leningrad-II is not designed against the crash of a large commercial airplane.

Other reactor types of Generation III have undergone or are still undergoing extensive processes of design review in other countries, most notably the UK Generic Design Assessment (e.g. EPR, AP1000). As part of this process, comprehensive technical documents are made public, increasing transparency and improving the opportunities for independent review of reactor types. The VVER-1200/V491 has not yet undergone a procedure of this kind.

It is noteworthy to explain that in the envisaged project Khmelnitsky 3 &4, Rosatom emphasized to build the reactor-type V-392 type. However, Annex B of the provided documents describes that the type V-392B has been selected for the project. The differences between reactor types V-392 and V-

5 http://www.world-nuclear.org/info/Country-Profiles/Countries-O-S/Russia--Nuclear-Power

392B are not pointed out. But it is not clear that this will be included all specific safety features of the V-392.[6]

The EIS also not explained to which extent the planned units at Paks II will be identical with the design of type V-491.

Regulations

Chapter 3 of the EIS includes a section presenting the Russian regulations which have to be applied for designing Russian reactors, but there is no statement on their compliance with the EUR requirements.

Nuclear safety principles and regulatory requirements are presented in EIS, but their implementation is rather only generally described.

Certification

Considering that only the predecessor of one of the versions, the AES-92 was certified by EUR, it cannot be confirmed at this point that the design to be built at Paks II will be by the time it is being built, also certified by EUR. According to another current EIA-Report concerning Bohunice 3, this reactor type has no EU certification yet.[7]

During the hearing, it is necessary to provide information about

- *The reactor type which will be chosen for Paks II*
- *The status of EU certification of the chosen reactor type*
- *The reference plant and its problems during construction*

Also, the following answer has to be provided

- *Which of the WENRA recommendations are not included in the Hungarian regulations yet?*
- *How are the nuclear safety requirements going to be implemented during the design, construction and operation of Paks II?*
- *In which areas is the design of units Paks II identical or similar to the design of Leningrad II*
- *Are there any differences? If so, in which areas?*

6 http://www.umweltbundesamt.at/fileadmin/site/publikationen/REP0441.pdf

7 http://www.umweltbundesamt.at/fileadmin/site/umweltthemen/umweltpolitische/ESPOOverfahren/UVP-EBO3/uve/JESS_UVP_Bericht_NJZ_An02.pdf

3Fulfillment of the WENRA Safety Objectives

The Western European Nuclear Regulators' Association (**WENRA**) defined and expressed a common position on the safety objectives for new nuclear power plants in November 2010.[8] The **safety objectives** were based on a report by the Reactor Harmonization Working Group of WENRA[9], also considering comments received from stakeholders. The WENRA safety objectives should ensure that the NPP which will be licensed in future will fulfill higher safety standards across Europe compared to the existing plants especially through improvement of the design. The safety objectives reflect the current state of the art in nuclear safety and can be implemented in the design using the latest available technology.

Based on these safety objectives, WENRA-RHWG developed positions on selected key issues of particular relevance considering the expectations for new reactors compared to existing ones. These positions are more detailed than the safety objectives and are intended to clarify their meaning. Together with these positions, considerations concerning the major lessons learned from the Fukushima Dai-ichi accident were published in a report.[10]

Among other issues, the positions concern the defense-in-depth approach for new nuclear power plants. This approach was developed further, with a refined structure including introducing two sub-levels in DiD level 3: level 3a for single initiating events, level 3b for multiple failures. Also, expectations on the independence between different levels of DiD were formulated. Other positions concern provisions to mitigate core melt and the practical elimination of severe accidents with large or early releases.

The **EIS does not discuss the fulfillment of the WENRA safety objectives (WSO) for new nuclear power plants**. The WENRA is only briefly referred to, as one of the international organization being covered by the current Finnish regulations. There is no discussion of the fulfillment of the individual objectives concerning the Paks II project.

*The following a*nswer is provided in section 3.5.3 of the International Chapter of the EIS: "Design of the Russian units was made in accordance with the official Russian legislation, taking into account at the same time the recommendations of the EUR, WENRA, and IAEA, as well as the requirements of the nuclear authority. Additionally, the units to be delivered at Paks must meet the Hungarian expectations and legal requirements alike, which in turn include the most up to date WENRA recommendations and the lessons learnt from Fukushima."

In the Austrian Expert Statement on Hanhikivi (2014), the WENRA safety objectives of 2010 were applied to the VVER-1200/V491.[11] The following points were assessed, with a focus on design aspects:

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8 Western European Nuclear Regulator's Association (2010): Statement on Safety Objectives for New Nuclear Power Plants. November 2010; www.wenra.org

9 WENRA-RHWG–Reactor Harmonization Working Group (2009): Safety Objectives for New Power Reactors. Western European Nuclear Regulator's Association. December 2009 (Published in the final wording in November 2010; www.wenra.org

10 Western European Nuclear Regulator's Association (2013): Safety of New NPP Designs. A report by RHWG – Reactor Harmonization Working Group. March 2013; www.wenra.org

11 http://www.umweltbundesamt.at/fileadmin/site/umweltthemen/umweltpolitische/ESPOOverfahren/uvp_fennovoima2014/REP_0479_Hanhikivi_EIA.pdf

- What can be asserted regarding fulfilment of the safety objectives, on the basis of available information?
- Which issues remain unclear regarding fulfilment?
- Are there potential challenges which could make fulfilment of the WENRA safety objectives difficult or impossible?

Important issues of this evaluation are presented in the following section.

<u>WSO 1 – Normal operation, abnormal events (DiD levels 1, 2)</u>

Objectives:

- *Reducing the frequencies of abnormal events by enhancing plant capability to stay within normal operation*
- *Reducing the potential for escalation to accident situations by enhancing plant capability to control abnormal events*

Among the basic principles and approaches of the design, the following items are mentioned by the developer[12]:

- Improving system and equipment characteristics by abandoning excessive conservatism and optimizing design margins
- Reducing capital and operating expenditures

It seems plausible that considerable efforts have been undertaken to improve the design of the VVER-1200, as compared to the forerunner types. However, there appears to be a challenge due to the potentially conflicting goals of improving safety on the one hand, and improving economics on the other.

Another challenge is the embrittlement behavior of the reactor pressure vessel material, given a planned service life of 60 years. In spite of extensive experiences with material behavior in the forerunner types, it appears that this is still a problem which needs observation.

<u>WSO 2 – Accidents without core melt (DiD levels 3a, 3b)</u>

Objectives:

Ensuring that accidents without core melt induce no off-site radiological impact or only minor radiological impact (in particular, no necessity of iodine prophylaxis, sheltering or evacuation).

- *Reducing, as far as reasonably achievable,*
  - *the core damage frequency taking into account all types of credible hazards and failures and credible combinations of events;*
  - *the releases of radioactive material from all sources.*
- *Providing due consideration to siting and design to reduce the impact of external hazards and malevolent acts.*

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12 St. Petersburg Research and Design Institute ATOMENERGOPROEKT (2011): Design AES-2006, concept Solutions by the example of Leningrad NPP -2. Saint Petersburg, 201

Internal hazards

Controlling internal hazards could be a challenge as far as the safety building is concerned: The safety building's structural elements containing the four parallel, redundant subsystems are physically separated, but placed side by side, connected by service corridors and channels for AC systems. Connections are separated by doors and dampers, calling into question the adequate realization of physical separation.

Furthermore, in the safety building each sub-system's low- and high-head pressure injection pumps and related equipment and pipelines have been placed in the same room without physical separation.

According to the Finnish nuclear authority (STUK), Finnish safety requirements concerning protection from internal hazards, such as floods and fires, have not yet been demonstrated.[13]

The break preclusion principle is applied to primary circuit piping. Nevertheless, break of pipe with largest diameter is taken into account in the ECCS design. Clarifications were still needed regarding dynamic effects of pipe break and their effects on the reactor's inner components.

From the documents available it is not clear to which extent a systematic consideration of all possible initiating events (including hazards) has been performed.

Multiple failures

No systematic discussion and consideration of multiple failures (level of DiD 3b, according to WENRA) could be found in the available documents. From the information at hand it does not become clear that all safety systems indeed have an active and a passive part each of which alone is sufficient to guarantee the safety function.

Electrical system

AC emergency power is provided by diesel generators (4 x 100%) and a gas turbine (1 x 100%). Thus, there is diversity, but no redundancy in the second case.

The separation principle for electrical systems has not been clearly described in the documents available for STUK assessment.

Requirements and results for CDF

Calculated CDF (mean values):

- Full power operation 1.36E-7/year, 2.24E-7/year or 3.82E-7/year
- Low power and shutdown 3.7E-7/year or 4.58E-7 /year
- CDF for all states: 5.94E-7/year or 7.52E-7 year, respectively

The published results of PSA studies appear to confirm that the limit of 1E-6/year for the core damage frequency is not exceeded. However, the results are quite close to the limit. It has to be noted that no information is available to which extent internal and external hazards are included in calculated CDF values. Furthermore, there is no information whether the numbers given refer to median or mean value; and no information on the uncertainty of the results. The published values suggest that at least the 95%-fractile of the CDF could be considerably higher than the limit, even if only the factors which can be included in a PSA are taken into account.

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13 Finnish Radiation and Nuclear Safety Authority (2009): Preliminary Safety Assessment of the Loviisa 3 Nuclear Power Plant Project; Assessment Report; 2009

Regarding core damage frequency, mean values of 5.94E-7/year and 7.52E-7/year are reported for the VVER-1200/V491 (for full-power and shutdown states; the extent of consideration of internal and external hazards is not clear), without any indication as to the uncertainty of these values.

It is commendable if fractile values are specified beside the mean or median values, to provide some indication of the uncertainty involved in the probabilistic analysis. However, it should be noted that not all uncertainties of a PSA can be quantified, and furthermore, that there are factors (for example safety culture, malicious human acts, and ageing phenomena) which cannot be taken into account in a PSA, or can be taken into account only in insufficient manner. Therefore, PSAs provide interesting indicators for plant hazards, but the numerical results cannot be taken at face value and should not be interpreted as reliable absolute measures for the frequency of severe accidents and large releases. The value of PSA results when discussing different plant types is thus limited.

WSO 3 – Accidents with core melt (DiD level 4)

Objectives:

Reducing potential radioactive releases to the environment from accidents with core melt, also in the long term, by following the qualitative criteria below:

- *Accidents with core melt which would lead to early or large releases have to be practically eliminated.*
- *For accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures.*

Containment integrity

According to the developer, physical phenomena related to severe accidents that might endanger the containment integrity are avoided as per the NPP design, namely[14]:

- Steam explosion in the reactor pressure vessel
- Hydrogen detonation
- Re-criticality of the core or the core melt
- Steam explosion beyond the reactor pressure vessel
- Direct heating of the containment
- Missiles
- Interaction between the melt and the under-reactor compartment floor and walls

It can be assumed that the formulation “avoided as per the NPP design” means that these phenomena do not have to be considered further; i.e., that they are practically eliminated by design measures.

It appears that a number of physical phenomena which could lead to large and/or early releases in case of a severe accident are regarded as practically eliminated by the designers of VVER-1200/V491. However, the concept of practical elimination is not explicitly addressed in the documents at hand.

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14 St. Petersburg Research and Design Institute ATOMENERGOPROEKT (2011): Design AES-2006, concept Solutions by the example of Leningrad NPP -2. Saint Petersburg, 2011

The concept of practical elimination has been introduced by IAEA. An accident sequence can be considered to have been practically eliminated if it is physically impossible for the sequence to occur, or if the sequence can be considered with a high degree of confidence to be extremely unlikely to occur.[15]

In the above mentioned report on safety expectations for the design of new NPPs, the Reactor Harmonization Working Group of WENRA has elaborated this concept, discussing among other issues means for practical elimination, and the demonstration of practical elimination.

In this report, it is stated that in order to increase the robustness of a plant's safety case, demonstration should preferably rely on physical impossibility. In any case, practical elimination cannot be claimed solely based on compliance with a probabilistic cut-off value. Analyses need to be supported by adequate experimental results. Uncertainties have to be taken into account, and sensitivity studies performed. All codes and calculations must be validated against the specific phenomena in question, and verified. Also, it must be ensured that the relevant provisions remain in place and valid throughout the lifetime of the plant.

Core catcher

An important feature of the AES-2006 is the core melt localization device (or core catcher). If functioning as planned, this new feature would have the potential to reduce the probability of large releases in case of a severe accident. However, the functioning of a core catcher is beset with a number of problems which have not been sufficiently clarified (for example: interaction between the molten core and concrete, considerable uncertainties regarding heat transfer between the materials involved; occurrence of cracks in the concrete of the device; hydrogen formation).

The core catcher of the VVER-1000/V466 which can be assumed to be similar to that of the VVER-1200 is placed in a concrete shaft below the reactor pressure vessel. It is filled with sacrificial material. The molten reactor core falls into this device after it has penetrated the pressure vessel bottom, and is cooled from above with water. The water from a building sump and the fuel pool is destined for this task.

The steam explosions constitute a severe problem for the core catcher design selected for the VVER-1000/V466. It is not guaranteed that the molten core will reach the core catcher all at once, as a whole. If, at first, only a part gets into the concrete shaft, it is likely that this will trigger flooding. Further molten core material then falls into water and the melt can fragment into small particles. In this way, heat transfer to the water is very fast, with abrupt vaporization as a result. For those steam explosions it is not possible today to predict the level of potential damage.

The core catcher is characterized by complex chemical reactions as well as complex physical processes. Adequate confirmation of the functioning by experiments and analysis thus constitutes significant challenges. Not least among those is the demonstration of transferability from experiment to the real component in the plant, i.e. the transferability from experiments with induction heated, small amounts of melts to a situation with a molten core.

There are open questions regarding the reliable function of the "core catcher" regarding description of accident scenarios, timing of core flooding to avoid steam explosion etc. The proof of functioning of this device (test, computer simulations), including the prevention of steam explosions, shall be presented during the hearing.

Filtered venting system

15 International Atomic Energy Agency (2012): Safety of Nuclear Power Plants: Design. IAEA Safety Standards Series No. SSR - 2/1. Vienna, 2012

A filtered containment venting system is not included in the AES-2006 design. It has to be mentioned that the Finnish requirements call for nuclear power plants to be equipped with a filtered containment venting system to mitigate the consequences of severe accidents.

Passive safety systems

According to STUK´s preliminary assessment, the passive systems to be used in transient and accident situations, the reactor circuit cooling residual heat removal system connected to the steam generators (PHRS SG), and the natural circulation based containment building’s residual heat removal system (PHRS C) are in the process of testing-based qualification. The correct functioning of the systems can be confirmed only after the test results are ready.

According to publication[16], the constraints of the capacity of the passive safety systems are also pointed out. It is emphasized that analysis is of realistic type, i.e.:

- Initial plant conditions correspond to normal operation at rated power without accounting for possible uncertainties in plant parameters.
- Core characteristics are assumed in accordance to design without accounting for the calculation of uncertainties and errors.
- Failures of equipment (other than assumed in scenarios) and operator errors are not taken into account.

The assumptions of the analysis show potential limitations of the passive safety systems, because during an accident additional equipment failures or operator errors cannot be excluded. Thus, the capability of these safety systems under real accident conditions could be limited.

Limit for large radioactive release which has to meet is 1.0E-7/year.

The calculated LRF (mean value) is 1.8E-8/year. This value, however, includes full-power operation and internal initiating events only. There is no information concerning its uncertainty.

The published results of PSA studies appear to confirm that the **limit** of 1E-7/yr **for the large release frequency** is not exceeded; they lie well below this limit (1.8E-8/yr). However, this value includes full-power operation and internal initiating events only. Low-power and shutdown states considerably contribute to CDF. The contribution of external events can also be significant, depending on the site.

There is no information concerning the uncertainty of the value given for LRF; it is also not clear whether it refers to the mean or the median value. All in all, it is not clear from the PSA results whether the limit for LRF could not in fact be exceeded, even if only the factors which can be included in a PSA are taken into account.

It could be a challenge to demonstrate practical elimination for the VVER-1200/V491 for all phenomena in question, taking into account these principles. According to the available information it is not assured yet.

<u>WSO 4 – Independence of levels of DiD (DiD levels 1 – 4)</u>

Objectives:

- *Enhancing the effectiveness of the independence between all levels of defense-in-depth, in particular through diversity provisions (in addition to the strengthening of each of these levels*

16 Bukin, N.V. et al (Gidropress): Effect of HA-2 and SPOT systems on severe accident prevention in WWER-1000/392 design; IAEA 3rd Research Coordination Meeting on Natural Circulation Phenomena; Cadarache; September 11–15, 2006

separately as addressed in the previous three objectives), to provide as far as reasonably achievable an overall reinforcement of defense-in-depth.

The independence of the **levels of DiD** is an important and constitutive element of the concept of defense-in-depth. WENRA expects that there shall be independence between different levels of DiD, to the extent reasonably practicable, so that failure of one level of DiD does not impair the defense in depth ensured by the other levels. The adequacy of the achieved independence shall be justified by deterministic and probabilistic safety analysis, and engineering judgement. Appropriate attention shall be paid to the design of I&C and other cross-cutting systems. The design of these systems shall be such as not to unduly compromise the independence of the SSCs they support.

However, in the design of the VVER-1200/V491, the concept of defense-in-depth appears to be seen, as a general underlying philosophy and not as a principle which is to be followed consistently through the whole design. The importance of independence of the levels of DiD is emphasized in a general manner, but is not consistently realized in the details of the design.

Furthermore, there are a number of features provided for severe accidents (DiD level 4) which are also used on lower levels of DiD: The two passive heat removal systems are not for exclusive use in case of a severe accident; they are also to be employed at safety level 3 (presumably for DEC A, according to the categories employed by the designers, which roughly corresponds to safety level 3b according to WENRA). Also, there is only one set of valves for primary circuit depressurization for DiD levels 3 and 4. Primary depressurization is highly important for severe accident management, to avoid core melt at high primary pressure, with high pressure melt ejection and possible containment damage. It has to be noted that Finnish safety requirements are not met, which call for independence of primary circuit depressurization in severe accidents from the systems designed for the plant's operating stages and postulated accidents.

Also, separation of **I&C systems** supporting different levels of defense-in-depth has not been made clear so far in the available documents.

The following question has to be answered during the hearing:

- Is there any conflict between safety and economics regarding the goals of larger operational margins for reducing the frequency of abnormal events and reducing capital and operating expenditures?
- How will be observed the embrittlement behavior of reactor pressure vessel over 60 years?
- What are the measures to avoid mistakes by manufacture?
- How will be ensured that the same problems as for the reactor pressure vessel of the EPR will not occur?

During the hearing, the following additional information has to be provided:

- *Systematic consideration and controlling of internal hazards*
- *Systematic consideration and controlling of multiple failures*
- *Redundancy of all systems of AC emergency power*
- *Demonstrating the fulfillment of the limit for CDF, taking into account all relevant initiating events, and uncertainties*
- *the demonstration of functioning and reliability of passive safety systems and features,*
- *the reliability of primary depressurization,*

- *the adequate confirmation of the functioning of the core catcher, by experiments and analysis,*
- *evaluation of PSA results and assessment of the uncertainty of PSA results,*
- *the demonstration of practical elimination for steam explosion in the reactor pressure vessel, hydrogen detonation and other phenomena,*
- *the applied concept of practical elimination,*
- *the explanation, whether a filtered venting system will be implemented and if not, about the reason to avoid this measure.*
- *Demonstration of independence between levels of defense-in-depth, to the extent reasonably practicable – in particular, regarding levels of DiD 3 (with sub-levels 3a and 3b) and 4*
- *Demonstration of separation of I&C-systems supporting different levels of defense-in-depth*

4Load-following operation

According to the published documents load-following operation is envisaged for Paks II.

Operating NPPs in Europe are mainly working in base load. Their flexibility is limited to a few two percent of nominal power. For new plants (under construction and planned) load following suggested to be fully implemented. But there is not much experience from operation practice. Investigations into the possible impacts of load following operation are limited and do not allow conclusions on the impacts in future.

Plants being built today, e.g. according to European Utilities' Requirements (EUR), supposedly have load-following capacity as a design feature. Even if a high flexibility is promised for the new reactors, some more research will be necessary until load following with the necessary capability can be implemented.

Controlling the reactor core during load following is challenging and difficult also for advanced reactors, in particular for reactors with large cores. The reactor has to perform the load changes while maintaining the core limitations for local power peaking and safety margins.

Operating NPPs in load-following mode causes technical disadvantages, because plant components are exposed to numerous thermal stress cycles; this leads to faster aging and requires more sophisticated systems for reactor monitoring and control.

Also an economic disadvantage of load following operation of NPP in a larger power range occurs if the plants are operated on reduced power.

During the hearing, the following questions have to be answered:

- Which is the possible impact of the load-following operation on Paks II?
- How will the envisaged load-following operation threat the safety of Paks II?
- Which is the impact of the load-following operation on the economic efficiency of Paks II?
- Which is the expected extent of the load-following operation?

5 Protection against airplane crash and terror attacks

According to the developer of the VVER-1200/V491, the design basis aircraft crash corresponds to the following load: Crash of an airplane with a mass of 5.7 t, at a speed of 100 m/sec.[17] Furthermore, there is protection against the impact of an (unspecified) large commercial.[18]

The protective design of the reactor building (double ferro-concrete cover[19]) appears to be well in line with the general standard of Gen III plants. It is plausible that it provides good protection against the mechanical impact of the crash of a commercial airplane, and also against the effects of vibrations.

Structural protection against collision by a large commercial airplane focuses on the outer containment and on the fresh fuel storage. However, it has to be noted that the safety buildings are not designed to withstand the impact of a large airplane. The building sections of the four redundant trains of the safety systems are located side-by-side; they are separated, but directly adjacent, without any physical distance, and hence several or all of them could be impaired by mechanical impacts. The same applies to the four diesel generators[20].

According to the STUK´s assessment in 2009, not all parts of the design objectives and principles of the AES-2006 plant alternative are consistent with Finnish safety requirements.[21] Of particular concern is the structural protection against airplane crashes:

Also, there is no discussion in the documents at hand on the possible **effects of combustion and/or explosion of aircraft fuel on structures and systems** which are required to bring and maintain the plant in a safe state after the crash.

This issue is addressed in the WENRA expectations for new reactors. It is stated that buildings or the parts of buildings containing nuclear fuel and housing key safety functions should be designed to prevent airplane fuel from entering them. Fires caused by aircraft fuel shall be assessed as different combinations of fire ball and pool fire; also, consequential fires shall be addressed.

STUK concluded in their assessment that demonstrating the realization of the safety functions in case of the crash of a large airplane is difficult. The fulfillment of Finnish requirements had not yet been demonstrated.

It has to be considered that the Paks II is also vulnerable against other attacks.

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17 ASE 2015: Provision of containment integrity at Russian VVER NPPs under BDBA conditions; Atomstroyexport; IAEA Technical Meeting; Severe Accident Mitigation through Improvements in Filtered Containment Venting for Water Cooled Reactors; 31 August -3 September 2015

18 St. Petersburg Research and Design Institute ATOMENERGOPROEKT (2011): Design AES-2006, concept Solutions by the example of Leningrad NPP -2. Saint Petersburg, 2011

19 The reactor building consists of a cylindrical part and a spherical dome. The internal cover is made of pre-stressed reinforced concrete, with a thickness of 1200 mm in the cylindrical part and 1000 mm in the a spherical dome. It has a steel lining (6 mm) on the inside to improve tightness. The external cover is made of reinforced concrete with a thickness of 800 mm in the cylindrical part and of 600 mm in the spherical dome. The gap between covers has a width of 1800 mm.

20 St. Petersburg Research and Design Institute ATOMENERGOPROEKT (2011): Design AES-2006, concept Solutions by the example of Leningrad NPP -2. Saint Petersburg, 2011

21 Finnish Radiation and Nuclear Safety Authority (2009): Preliminary Safety Assessment of the Loviisa 3 Nuclear Power Plant Project; Assessment Report; 2009

During the hearing, the following information has to be provided:

- *Which are the international requirements the physical protection is based on?*

6 Accidents

Chapter 8 of the EIS summarizes the potential impact factors related with the implementation of the Paks II project, for both normal and abnormal operations during all phases of the project. These impact factors are grouped depending on the locations that may be affected, the time when they may appear, and their characteristic types. However, only design-basis conditions are considered for the evaluation of accident conditions, and it seems that this is due to the nuclear safety requirements currently in place in Hungary.

According to the IAEA recommendations, the impact due to beyond design basis accidents and severe accidents at the nuclear power plant should also be evaluated for the purpose of assessment of environmental impact of new NPP.

The findings of the analyses performed for the purpose of environmental impact assessment of the proposed Paks II project are summarized in Chapter 22 of the EIS. This chapter presents in tabular form the results of the evaluations performed for normal operation and design-basis accident conditions during the construction, operation and decommissioning of Paks II. None of the impacts summarized in this chapter and analyzed under the EIS will have a cross-border character.

However, such impacts were analyzed in the International Chapter of EIS, though the status of this chapter is not clear. It is not included in the table of content of the EIS, which might imply that it is not part of the main report.

The results of an analysis of a severe accident with a substantial emission of radioactive substances should not only be published in the chapter on transboundary impacts of the project, but most importantly also in the main report and be discussed with the Hungarian public, local authorities, emergency authorities and security authorities. In this discussion the issue of emergency preparedness and response should also be included.

Transboundary impacts

The transboundary impact assessment presented in the International Chapter is incomplete. In case of incidents or accidents occurring at Paks site, Germany as well as other European countries could be affected as a result of an airborne release of radioactive substances. Therefore, a detailed identification and evaluation of all possible incidents and accidents which may occur at the Paks NPP site is of great importance for the EIA procedure. Due to the proximity to the German state territory and to the level of the radioactive inventory, the existing as well as the planned nuclear power plants possess a potential threat. Even if the probability for Beyond Design Basis Accidents (BDBA) is very low, they should be assessed in the framework of the EIA procedure very carefully. For the assessment of a potential impact, the evaluation of possibly severe accidents including the maximum source term and the most unfavorable weather conditions, which could lead to radioactive fall–outs, are of highest interest.

For the purposes of estimating the transboundary radiological impact on the neighboring countries in case of a severe accident resulting in airborne releases, the TREX (Euler-model) code was used in the EIS. The model is briefly described, but no other information or reference is given regarding its validation.

The EIS relies on the use of one dispersion model, the TREX-Euler model. It would be advisable to use other models as well and compare the results. Early observations after the Fukushima accident showed that the predictive value of different models could lead to qualitatively different outcomes.

Also, the results presented in section 2.3.5 of the international chapter are incomplete. According to the EIS, inhalation doses for children and adults have been calculated for several regions that are in a distance of more than 30 km to the NPP. While it is recognized that inhalation doses might be the main contributor to the total dose, the presentation of the other doses, as well as of the total doses, is needed. In order to allow the verification of the radiological cross-border impacts it is necessary to calculate and present all the doses on all exposure pathways, as well as the total doses. Ground-shine, cloud-shine and ingestion doses are of relevance for assessing short- and long-term health consequences.

Moreover and most important, the source term used for these calculations is very low compared to other calculations. For Cs-137 a source term of 5.2 TBq has been assumed – this is several orders of magnitudes lower than source terms from Fukushima or Chernobyl.

In 2012, the Norwegian Radiation Protection Authority published a report concerning the potential consequences in Norway after a hypothetical accident at the NPP Leningrad II (Russia). [22]The calculation was based on the most severe radiological consequences that could occur after a 'credible' accident in a VVER-1200 (AES-2006/V491). The definition of the release categories and the associated source term data were based on simulations conducted as a part of Level 2 PSA for a VVER-1000/V320 plant. The radionuclide inventory of the core was based on Russian data derived for the original Soviet fuel. The source term was calculated to 2,800 TBq Cs-137.

It is necessary to include a conservative worst-case release scenario in the EIS, since its effects can be widespread and long-lasting and even countries not directly bordering Hungary, like Germany, can be affected.

Atomstroyexport claims that the frequency of a large release (LRF) is 1.8x10-8/year, well below the limit of 10-7/year. However, the published result includes full-power operation and internal initiators only. Low-power and shutdown states can contribute significantly to LRF, as can external events. Furthermore, no indication is given for the uncertainty of this result.

Most important: the numerical results of PRA studies should not be taken at face value. PRAs are beset with uncertainties, and cannot completely capture reality. The following factors cannot at all, or not adequately, be taken into account in a PRA: unexpected loads from internal occurrences, bad safety culture, ageing-related common cause failures, problems at the interface between civil engineering and systems engineering, unexpected external events acts of terror and sabotage.

It is unjustified to exclude accidents in NPPs from further consideration, solely on the basis of PRA results. Taking into account the open issues and challenges identified regarding the safety of this reactor type, and in particular the lack of demonstration that accidents with large releases are extremely unlikely, with a high degree of confidence, severe accidents with large releases cannot be excluded for the VVER-1200/V491, based on information available today. According to evaluations of the chosen reactor type VVER-1200/V491 it has to be concluded that severe accidents with large releases cannot be excluded.

Moreover, the EIS does not give any or insufficient attention to multi-unit incidents and accidents including problems caused by incidents or accidents in other units on the site; management of contaminated water after a severe accident and the risk and potential impacts of sabotage, terrorist attack and acts of war.

Given the long operation time of the proposed project (more than 60 years) and the needed cooling down period before final decommissioning, it is not possible to guarantee political stability in Hungary for this period of time. An analysis of potential environmental impacts after malevolent acts of war against the project is therefore a vital aspect of any EIA for a nuclear project.

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22 Statens strålevern: Potential consequences in Norway after a hypothetical accident at Leningrad nuclear power plant; Potential release, fallout and predicted impacts on the environment; Norwegian Radiation Protection Authority; June 2012

To estimate the possible consequences of a severe accident at Paks II the results of the research project flexRISK could be used. [23]

Additional information is necessary to evaluate the possible consequences of a severe accident.

During the hearing the following issues have to be addressed:

- *Will the revision of the Hungarian NSR imply the modification of the requirement to analyze only design base accidents for the purposes of environmental impact assessment in accident conditions?*
- *Why are the consequences of a severe accident calculated in the international chapter of the EIS not included in the main report?*

- *Detailed description of the measures for prevention of severe accidents and the mitigation of accident consequences*
- *Validation of the TREX (Euler-model) code used for modelling of the dispersion of accidental airborne releases*
- *Calculated doses in different distances, not only for the distance of 30 km*
- *Calculation and presentation of the doses on all exposure pathways, as well as of total doses*
- *Justification of the used source term*
- *Results of PSA (Level 1, 2 und 3) in particular the probabilities/frequency of core damages (CDF) and severe accidents with (early) large releases (LRF and LERF) including probability distribution (fractiles) and source terms for the most important release categories*

- *Cumulative impact of all nuclear installations existing at the site and planned to be built on the site for accident conditions, including the impact of one installation on the others, and the cumulative impact of accidents affecting more than one unit at the same time*

23 FLEXRISK(2013): The Project „flexRISK“: Flexible Tools for Assessment of Nuclear Risk in Europe; http://flexrisk.boku.ac.at/en/projekt.htm

7 Radioactive waste and spent fuel

Chapter 19 of the EIS deals with the radioactive waste (RW) and spent fuel (SF) management.

National Program

Subsection 19.2.1.1 explains the National Policy and National Program; however, there is no mention of national strategy and program for RW and SF management being actually in place in Hungary. According to Directive 2011/70/Euratom every member state has to establish a national program for the management of spent fuel and radioactive waste until August 2015.

It is not clear if a national program for the management of RW and SF does exist in Hungary, as requested by the Council Directive 2011/70/EURATOM establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste.

The spent fuel and radioactive waste that will be generated during operation and decommissioning of Paks II has to be discussed in the light of the (future) national management program for spent fuel and radioactive waste.

Interim storage facility

It is mentioned that a joint waste management for Paks NPP and Paks II NPP is considered a "non-viable" solution and, therefore, a separate waste management technology should be established for the new units. The SF management section is presenting the options taken into consideration for the new units of Paks NPP, and it includes a comparison of the wet storage technology with the dry storage technology. From all the options analyzed in the study, the dry cask storage seems to be the preferred solution for Paks II: It is explained in section 19.8.1.4 that, after being discarded from the reactor, the SF will be placed into the SF pool for cooling down until reaching a temperature which will allow its dry temporary storage.

Due to the configuration of the stored fuel assemblies, a severe accident will certainly impact more fuel assemblies in a wet storage facility and most likely more assemblies in the block storage facility (which is currently used for the storage of the spent fuel of Paks NPP) than in a dry storage facility using transport and storage containers; therefore the release potential is higher, too. For this reason it is of utmost interest to know which kind of facility will be built for the interim storage of the spent fuel of Paks II.

The operation time for the interim storage facility is not mentioned in the EIS. Section 8.1.2.1.5 of the EIS only states that the temporary storage of spent fuel on site for several decades "perhaps even beyond the plant's operation time" is envisaged. Thus, interim storage of spent fuel for more than 60 years is possible. Such a long-time storage would entail consideration of all the specific (enhanced) safety features to be addressed.

According to the IAEA, long-term storage is considered to be storage beyond approximately 50 years. The IAEA highlighted that a defined storage end point is important since it determines the basis for the design lifetime of the facility, packaging requirements and financial guarantees, and the planning basis for subsequent disposal facilities.[24]

In spite of the envisaged long storage periods, the EIS does not discuss the issue of safety-relevant aspects to guarantee long-term safety for the planned storage facility.

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24 International Atomic Energy Agency (2012): Storage of Nuclear Fuel; IAEA Safety Standards, Specific Safety Guide, No. SSG-15, Wien 2012. http://www-pub.iaea.org/MTCD/publications/PDF/Pub1503_web.pdf

Reprocessing of spent fuel

The EIS explains that spent fuel which will be produced during the operation of the new power plant units will rest for maximum 10 years in the spent fuel pool placed near the reactors. It is further explained that Paragraph 2 of Article 7 of the Convention promulgated in Act II of 2004 ensures the possibility – following storage in the spent fuel pool – to transport the spent fuel produced during the operation of the new power plant units to Russia for technological storage, or for technological storage and recycling. In this case, the time frame of the storage can be prolonged up to 20 years. Building a reprocessing facility in Hungary is seen as very unlikely, so if reprocessing will be chosen, this will have to be done abroad.

Reprocessing and transportation of spent fuel and radioactive waste are combined with additional risks for people and the environment. Thus, it would be seen as a good decision if Hungary will cancel the option of reprocessing abroad.

Management of high-level waste (HLW)

According to EIS section 19.8.2.1.2, HLW will be managed by collecting it in shielded containers that will be stored in the auxiliary building until the dismantling of the unit or the commissioning of a HLW repository. However, it is not mentioned how the residual heat generated by this waste will be removed. According to the international standards and practices, HLW has to be stored in a similar way as SF. This aspect is not properly addressed in the EIS.[25]

Current status of planning the back-end strategy of SP and RW

For final disposal of HLW, long-lived LILW and SF a deep geological repository will have to be established in Hungary. In section 19.6 of the EIS the current stage of the activities undergone in Hungary for developing a repository (deep geological disposal facility) are described shortly. It is stated, for choosing the location, year-long studies were conducted in the area of the Western Mecsek hills, which is the most suitable for disposing radioactive waste of high activity. In 2008, the conceptual plan of the long-term program of the study was completed. The first period of study phase 1 was closed in 2010. In 2014, the studies were expected to restart in Western Mecsek, where a deep geological laboratory is planned in the clay soil to be built until 2030, then the repository will also be worked up.

It is of utmost interest to know about the progress of deep geological disposal, because the potential hazard of spent fuel and HLW stored at the interim storage is much higher than of those waste stored in the repository.

Options of the final disposal of spent fuel

Two options for the management of the spent fuel are presented and analyzed in subsection 19.4.3 of EIS. The first one is the direct disposal of spent fuel, without reprocessing, into a deep geological

25 http://www.umweltbundesamt.at/fileadmin/site/umweltthemen/umweltpolitische/ESPOOverfahren/UVP_paksII/REP0533_PAKSII.pdf

repository. The second approach is the so-called “closed” fuel cycle[26], with reprocessing the spent fuel and recycling of the recovered plutonium and uranium.

However, a third approach is mentioned: The third scenario is the approach called “wait and see” strategy, where the spent nuclear fuel is loaded into an interim storage for a longer period. The fuel elements are stored until the decision on their permanent disposal is made. According to the EIS, this strategy was replaced by the *”Do and See”* policy in the last few years. It is stated that this policy is briefly about to consider that every process of program could contain consecutive phases and between these phases there could be branching. At these branching points decisions have to be made concerning the program according to the appropriate deliberations.

It is of utmost interest to understand the difference between the approaches “wait and see” and “do and see”. As mentioned before, the potential hazards of the interim storage facility for the transboundary impacts in case of a severe accident are much higher compared to the deep geological repository.

Final strategy for disposal of low and medium radioactive wastes

The low and medium radioactive wastes of the Paks II are to be stored at the existing repository in Bátaapáti. Therefore, the EIS needs to contain information concerning the capacity of the repository at Bátaapáti and the need/possibility of enlargement. However, this information is not included in the EIS.

Amount of radioactive waste and spent fuel

According to the EIS, the SF that will be generated during the operation of Paks II after 60 years of operation will account for 3,348 t. For this estimation the data provided by the supplier were used, as well as the data known from the operating units.

Data on the amount of radioactive inventory on site as a whole, subdivided into the applied categorization of radioactive wastes, are missing, including the existing quantities of RW and SF.

Accidents concerning RW and SF

Section 19.8.5 of the EIS deals with accidents, in regard of the generation of radioactive waste. Like in the whole study, only the design-basis accidents were considered. From these, the only events kept for analysis are those generating RW. For these it is mentioned that they will be collected and stored in the auxiliary building before further treatment. Since the list given in this section includes damaged fuel, storing in the auxiliary building is not adequate. The quantities of the RW/SF to be produced during a design-basis accidents are not given; the only information mentioned is that the respective quantity will not exceed the capacity of the temporary storage facility. These conclusions should be substantiated with adequate calculations. Furthermore, an even more important generation of RW and SF following severe accidents (and not only design-basis accidents) should be estimated and presented.

Accidents affecting the RW and in particular SF management facilities to be established on site are of utmost importance. These accidents should be evaluated and their impact on the environment considered. However, chapter 19 of the EIS does not include the evaluation of an accident in the spent fuel pools or in a spent fuel storage facility. Such analyses should include design-based

26 This fuel cycle is not really closed, because there remain also long-lived radionuclides which have to be stored.

accidents as well as severe accidents including those that could lead to emissions of substantial amounts of radioactive substances into the environment.

Concerning transboundary impacts it is stated in subsection 19.8.2.3 of the EIS that in the case of compliance with the strict instructions and process descriptions during normal operations with respect to the management of radioactive wastes and spent fuel, the environmental impacts shall not reach or go beyond national borders.

Strict instructions to ensure safe operation are necessary to limit the probability of an accident. However, these measures are not at all sufficient to prevent any accident. Thus, the impacts of a possible accident, despite their low probability, have to be analyzed.

The following issues are to be addressed during the hearing:

- *Does a national strategy / program for the management of RW and SF exist in Hungary, according to the Council Directive 2011/70/EURATOM establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste? And, if yes, does this national program include the spent fuel and radioactive waste of Paks II?*
- *the total estimated quantities of RW and SF at the site (when all units will be in operation) and the existing quantities on Paks NPP site.*
- *Envisaged storage concept for the interim storage facility of the spent fuel of Paks II*
- *Planned storage time for the spent fuel in the interim storage facility*
- *Measures to ensure safety-relevant aspects of long-term storage*
- *How will residual heat be removed from the high level waste which is planned to be stored inside the Auxiliary Building?*
- *Is reprocessing abroad an option for the management of spent fuel in the (new) national program for the management of spent fuel and radioactive waste according to Directive 2011/70/Euratom, which had to be provided until August 2015? If yes, are there any intentions to cancel this option?*
- *current stage of evaluation of the Western Mecsek Mountains,*
- *the necessary capacity of the planned repository to store all additional radioactive waste and spent fuel,*
- *the time schedule for the construction/start of operation of the repository. (To assess the reliability of the time schedule it is also of interest to know whether there have been any changes in the time schedule since 2010.)*
- *Types and quantities of RW and SF following design-basis accidents and severe accidents*
- *the repository at Bátaapáti and the need/possibility of its enlargement.*
- *Radiological consequences, in particular possible transboundary impact of accidents affecting the RW and in particular SF management facilities to be established at the Paks II site*
- *the differences of the strategies/ approaches “wait and see” and “do and see” for the final disposal of spent fuel.*

8 Considering Alternatives

The alternatives to the Paks II project are not presented in the EIS, neither regarding alternative reactor designs, as contained in the Scoping Report, nor regarding non-nuclear alternatives.

Originally, the suppliers of five reactor types were selected as candidates for being invited for the tendering process:

- AP1000 (Toshiba-Westinghouse)
- AES-2006 (Atomstrojexport)
- EPR (Areva),
- ATMEA (Areva-Mitsubishi)
- APR1400 (KEPCO–Korea Electric Power Corporation)

Out of the above-mentioned five reactor types considered in the first phase of the EIA procedure (the Scoping phase), only the VVER-1200 technology has been selected for environmental assessment. Moreover, in January 2014 the Hungarian Government and the company Rosatom signed an agreement on the cooperation between the parties in the field of development of nuclear technology with the possibility of establishing new NPP units in Hungary.

The EIS did not put forward any of their arguments which led to the choice of this reactor type. For such a decision, however, the expected environmental effects have to be considered, and the reason for the decision has to be presented.

A justification for this choice in respect of a comparison of environmental impacts of the different designs as proposed in the scoping phase is lacking. However, according to the EIA Directive of the EC and the ESPOO Convention, alternatives have to be presented in an EIA Report.

According to Article 5 par. 3 lit (a) of EU Directive 2011/92/EU the project applicant has to provide a “Comparison of power generation alternatives from environmental point of view”. This comparison has to include a description of justifiable alternatives (e.g. for the site or technology) for the project under consideration in line with Annex II Espoo Convention.

The Espoo Convention prescribes in Appendix II (b): “*A description, where appropriate, of reasonable alternatives (for example, locational or technological) to the proposed activity and also the no-action alternative*;” and adds that these alternatives have to be assessed on their environmental impacts (Appendix II (c, d)). These alternatives furthermore have to be subject to the transboundary consultations (Espoo art. 5(a)). Aarhus 6(6e) requires “*An outline of the main alternatives studied by the applicant*”, whereby all information relevant for decision making needs to be included.

Thus, it is necessary to include an objective comparison of the environmental impacts of each potential alternative reactor technology.

Alternatives for the site are also missing. However, two more reactors on the site which already host four reactors increase the risk on multi-unit accidents due to an earthquake or other natural hazards. This also increased the negative influence on the cooling water

Alternatives for energy production, especially scenarios based on renewable energies and energy efficiency, are also missing in the EIS.

During the hearing the following information concerning the alternatives of the Paks II project has to be provided:

- What was the reasons for the selection of the VVER-1200? Which environmental effects have been considered to make this decision?

- Which reactor technology has the lowest risk of a severe accident? Which technology has the least impact on the Danube water temperature?
- Which alternative site for the envisaged new NPP are considered and what are the reasons for the current choice?
- Are there reasonable alternatives that can lead to the same goals (*reduction of greenhouse gas emissions, increasing security of supply of electricity, provision of affordable electricity for economic development, provision of economic activity for job and wealth creation)* without the chance of a severe accident that could release large amounts of radioactive substances?

9Economical aspects

According to the contract Hungary signed with Russian Federation, the two reactors of the new plant will cost 12.5 billion EUR. 80% of the foreseeable capital cost is financed by an intergovernmental loan from Russia. This loan has to be paid back during 21 years, the starting date is not linked to the start of operation of the new units. But the construction of new nuclear power plants is usually delayed, so the loan burden will even become more severe. In March 2015, the Hungarian Parliament decided to keep these contracts secret for 30 years. [27]

The Hungarian NGO Energiaklub recently published an analysis on the issue of state aid[28]: Unless the wholesale power prices show a permanent real price growth, the project will not pay off. Considering international power price forecast it is very likely that Hungarian taxpayers will have to help out on a large scale. NPP Paks II will be in permanent need of additional capital, which will make state aid a fact.

Serious doubts arise whether nuclear plant Paks II can survive in the electricity market without massive state aid considering the high investment costs for new nuclear power leading to high demands for state aid for such projects and a lack of private financing instruments.

In the light of the high investment costs for new nuclear power plants there is real danger that, due to economic reasons, a constant upkeep of the required high safety level of the plants cannot be guaranteed.

Due to the high investment costs for new nuclear power plants the ability to guarantee a high nuclear safety level is of high importance. It should be explained how the project applicant can guarantee a continuous implementation of a high nuclear safety level with increasing investment needs.[29]

However, the EIS clearly states in section 1.3.2.3 that economic or financial matters related to the installation of the planned units are not addressed. In the international chapter of the EIS it is explained that the discussion and answering of any kind of economic or financial issues does not constitute a subject matter or a function of the Environmental Impact Study.

According to current knowledge none of the reactor suppliers can categorically exclude severe accidents. Therefore, the economic assessment should include also follow-up costs of severe accidents and be put into comparison with the existing nuclear liability provisions in Hungary.

The following information has to be provided during the hearing

- *Production costs of NPP Paks II over the whole project cycle – from drawing up the project to construction and operation to decommissioning and storing all the radioactive wastes in interim storages and a repository*
- *Comparison of the production costs of Paks II with alternatives.*
- *Explanation how the project applicant can guarantee a continuous implementation of a high nuclear safety level with increasing investment needs*

27 In addition to the Russian loan, a fuel supply contract with Russia for 20 years has been signed, without a tender procedure. The EU Commission considered this exclusive 20-year contract as a threat to energy security and in violation of the ESA – Euratom Supply Agency rules and signed the agreement only after the duration had been cut down to 10 years.

28 Felsmann 2015: Balász Felsmann: Can the Paks-2 NPP operate without state aid? The power plant company: a business economics approach. Energiaklub. 23.6.2015

29 http://www.umweltbundesamt.at/fileadmin/site/publikationen/REP0418.pdf

10Non-radiological environmental impacts

Former energy undersecretary Attila Holodi told the media that, on the basis of the 2008 feasibility study, a number of academics concluded that the Danube was incapable of handling the additional heat discharged by Paks II, projecting that it would raise the river's temperature to 30 degrees with catastrophic consequences for its wildlife.

A few months after the official documents for the EIA procedure have been published, Benedek Jávor, Member to the European Parliament obtained **secret documents**. On the website of Nuclear Transparency Watch he declared that the MVM documents contain fundamental mistakes and that the modelling is based on outdated data and no appropriate modelling of the system integration was provided. The authenticity of the documents has been confirmed by both MVM and the Hungarian Academy of sciences.

One of the secret documents Benedek Jávor obtained is specifically about cooling water and highlights the faulty design of the cold-water channel of Paks I. The documents stated that the new NPP will cause serious nature protection problems and that excess heat will call for (partial) capacity reduction which will in itself hinder the 95% planned use of the new NPP – thus the financial goals will not be achieved.

Furthermore, it is likely that nature protection standards will rise in the coming 60 years and might lead to new situations like the one in the Rhône where drought threatens the cooling of nuclear plants.

The following questions have to be answered during the hearing:

- *What is the current status of the evaluation of the possible effect to the Danube?*
- *Are there any measures implemented to avoid negative effects besides stop operation or decrease the thermal power?*