CONSTRUCTION OF A NPP IN BELARUS

Assessment of Answers to the Questions posed in the Austrian Expert Statement on the Preliminary EIA Report

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INTRODUCTION

In 2009, the government of Belarus decided to construct a nuclear power plant (NPP) with a capacity of 2,300–2,400 MWe. Austria takes part in the transboundary Environmental Impact Assessment (EIA) for the construction of the NPP in Belarus. The Environmental Agency Austria, "Umweltbundesamt" has assigned the Austrian Institute of Ecology, in cooperation with Dr. Helmut Hirsch, scientific consultant, to elaborate an expert statement on the Preliminary EIA Report presented by Belarus.

As a result of comprehensive analysis of industrial reactor units, the Russian project NPP-2006 of the Generation III VVER was chosen for the Belarusian NPP. The government of Belarus is convinced that this project conforms to modern international nuclear safety and radiation protection requirements. The Austrian review of the Preliminary EIA Report was focused mainly on the safety and risk analysis, with the goal to assess if the EIA allows making reliable conclusions about the potential impact of transboundary emissions. For that safety features, equipment and procedures for severe accident management should be explained in detail. In total 20 open questions were formulated by the Austrian experts. In March 2010 Austria received the answers on these questions. In the present document the result of the evaluation of these answers is presented according to the chapters of the Austrian expert statement.

In the following chapters we refer to the documents as follows:

- "Substantiation of investments in construction of the nuclear power plant in the republic of Belarus – Preliminary Report on EIA of Belarusian NPP" as (REPORT, 2009);
- Construction of a NPP in Belarus - Expert Statement on the Preliminary EIA Report, A.Wenisch, H.Hirsch, A.Wallner; Umweltbundesamt Report 0250, Vienna 2009, (UBA, 2009);
- Replies to expert opinion on preliminary report on EIA of the Belarusian NPP carried out on request of the Federal Ministry of Agriculture, Forestry, Ecology and Water Management, A.N. Rykov, A.I. Strelkov. as (REPLIES, 2010).

In the Replies of the Belarusian Experts some of our questions are formulated in a different manner. These questions are included in this present document in both wording, in order to show where a misunderstanding might have evolved.

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SELECTION OF THE NPP TYPE

Summary and assessment of the Preliminary EIA Report

As a result of comprehensive analysis of industrial reactor units, the Russian project NPP-2006, a Generation III Russian Pressurized Water Reactor (PWR) with 1,200 MWe (VVER.1200/V491 further V-1200) was chosen for the Belarusian NPP. In the EIA some other types were compared to this reactor as alternative options.

Experience with equipment and safety systems in prototype units was, according to the Preliminary EIA Report, the main reason to select VVER-1200/V491. However, no operating experience has been gained so far in proper VVER-1200 prototypes. There is operating experience from an earlier model, the VVER-1000/V428 (an advanced version of VVER-1000). Thus, compared to other PWR types, experience relevant for VVER-1200 is not significantly more comprehensive.

It was reported in a technical magazine that there are two variants of VVER-1200, V-392M and V-491. Passive safety systems prevail in the former, whereas the latter focuses more on active systems (NEI 2009). There is no discussion in the Preliminary EIA Report why V-491 was chosen and not V-392M.

Question 1

Can the reasons for the choice of the reactor type (VVER-1200) be explained in more detail regarding experience with components and systems, and possible other reasons? (UBA 2009)

SUMMARY OF THE ANSWER

It is not elaborated why the PWR was selected as basic type (this point is very briefly discussed in the Preliminary EIA Report).

In the answer, the PWR types already listed in the Preliminary EIA Report are discussed. As the sole criterion, the number of projects for the types under consideration is considered, as well experiences with construction times. It is pointed out that no AP-600 or AP-1000 has been constructed so far, and there are serious delays of the construction of the two EPR which are currently built. On the other hand, according to the answer, there is very good experience with forerunner types of the NPP-2006 (VVER-1200)[1].

ASSESSMENT

The reasoning for selection PWR is too restricted in the Preliminary EIA Report and would have required some elaboration. For example, it is stated that the PWR has a higher power density in the core than BWR or CANDU-reactors, implying a minimum size per power unit.

This, however, only concerns the size of the reactor core and hence, the reactor pressure vessel. It does not necessarily imply that the whole plant would be of smaller size than plants

1 The identification of the different types of VVER reactors with electrical output of 1000 to 1200 MW is sometimes confusing. It seems that VVER 1000 and VVER 1200 both are used for the NPP 2006.

of other types.

Furthermore, a high power density can be seen as disadvantageous, since it will lead to a faster heating-up of the core in case cooling fails.

The experience with the various reactor types is not presented in a comprehensive manner in the reply. No construction times (including a comparison of original schedules and actual outcome) for the VVER projects which are mentioned are provided; however, this would be required for assessing to which extent delays have also occurred for this reactor type.

More importantly, the features of the respective designs relevant for safety are not mentioned as a criterion. A comparison of such design features would be of high relevance for the type selection (for example regarding emergency core cooling system, emergency feedwater systems, features of the containment, electrical and I&C systems).

Questions for follow-up

- Could the relative merits and shortcomings of the PWR, as compared with BWR and CANDU, be elaborated in more detail?
- Could the experience with recent VVER-projects be elaborated in more detail, in particular regarding the construction schedules?
- Has there been no comparison of the safety significant design features of the PWR types under consideration? If no – could it be justified why this has not been taken into account? If yes – could the results be provided?

Question 2

What were the reasons for selecting the variant V491 and not V392M, and hence relying more on active, than on passive safety systems? (UBA 2009)

Summary of the Answer

It is stated that the choice has been carried out using a complex of indicators – mostly concerning safety and reliability. It is confirmed that V392M contains more passive systems.

The indicators and criteria which were used and which led to the selection of the V491 are listed in a very general manner. No details of the selection process are provided.

Assessment

It would have been appropriate to provide some in-depth information about this selection process (indicators and criteria used, methodology applied). In particular, it would have been of interest to learn which role the differences regarding active and passive safety systems have played in this process, and how the advantages and disadvantages of passive safety systems are generally seen by the Belarus side.

Questions for follow-up

- What were the indicators and criteria applied in the comparison – could some more detailed information be provided? Which methodology was applied to combine the indicators and criteria and arrive at an overall judgment?
- Which importance was given in the comparison process to the basic character of the safety systems – active or passive? How are the advantages and disadvantages of

passive safety systems seen by the Belarus side?

DESCRIPTION OF PROJECT

Summary and assessment of the Preliminary EIA Report

While a number of basic data concerning the reactor design and operational parameters are provided in the Preliminary EIA Report, there is no detailed description of the safety systems which are mentioned. It is not possible to gain a comprehensive picture of the functioning and reliability of those systems.
The capacity factor provided in the report (96 %) appears to be very optimistic. An expected capacity factor of up to 90 % has been recently reported in a technical article (NEI 2009).

Several technical features are presented as new in the Preliminary EIA Report which already are implemented in many currently operating Generation II plants.

The resistance of the VVER-1200 against external impacts (which depends to a considerable extent, but not exclusively, on the wall thickness of the containment building) as specified in the Preliminary EIA Report is, in some cases, inferior to that of modern Generation II PWRs. In the Preliminary EIA Report, the airplane crash the building has to withstand is not specified.

There are also some new features compared to Generation II plants.
Most notable is the corium localization device (system for core melt trapping and cooling, usually referred to as "core catcher" in the English technical literature). This device, however, is only mentioned in the Preliminary EIA Report; no description is provided.

The passive system for heat removal from the steam generators appears to be a genuinely new system, compared to Generation II PWRs. However, this system is not described in any detail and hence cannot be assessed further here.
The claimed capability of long-time residual heat removal by passive systems is only mentioned in the Preliminary EIA Report, and not explained and discussed further. This point also cannot be further assessed here.

Question 3

The CAPACITY FACTOR given in the Preliminary EIA Report (about 96 %) is very high. What is the basis for this assumption? (UBA 2009)

The EFFICIENCY factor specified in the Report (more than 96 %) is very high. What was the basis for the given assumption ? (REPLIES 2010)

SUMMARY OF THE ANSWER

The answer does not deal with the capacity factor, but with the thermal efficiency of the plant. There appears to have been a misunderstanding.

ASSESSMENT

The misunderstanding should be clarified and the question as forwarded by the Austrian side answered.

Questions for follow-up

See original question.

Question 4

Can a description of the passive high-pressure boron injection system (design, operating parameters) be provided? (UBA 2009)

Summary of the Answer

Basic data of the system (boric acid concentration, operating temperature and pressure) are provided. The system consists of four channels (4x50 % redundancy) and is located inside the containment.

Basically, this is a system with hydraulic accumulators as they are already widely used in PWRs operation today (Generation II plants).

It is pointed out that further details will become available in the course of the project.

Assessment

At the present stage of the project, this answer appears appropriate. It provides a basic idea concerning the essential features of the system in question.

Questions for follow-up

None at the present stage.

Question 5

What are the wall thicknesses (cylinder and dome) of the double containment building of the VVER-1200? (UBA 2009)

Summary of the Answer

Data on the thickness of the internal and external containment hull are provided, differentiating between cylinder and dome, as requested. Furthermore, the width of the gap between the two covers is specified.

Assessment

The question has been answered in full.

Questions for follow-up

None.

Question 6

What are the parameters of the maximum aircraft crash (plane mass and speed) the containment building can withstand? (UBA 2009)

Summary of the Answer

The answer specifies the weight of the plane – 5.7 tons, and the speed - 100 m/s. (In the English version of the reply, the speed is given as 100 km/s, which clearly is an error. The Russian version of replies states 100 m/s.)

Assessment

The question has been answered in full.

However, it is noteworthy that this aircraft crash represents a considerably smaller load than those assumed for many newer Generation II plants. For example, 20 tons and 215 m/s are assumed for German pre-konvoi and konvoi plants, corresponding to the crash of a Phantom fighter-bomber. A design on the basis of such loads also offers a degree of protection against the crash of a large commercial airliner.

Questions for follow-up

- How is the assumption of 5.7 tons, 100 m/s justified; which considerations led to this assumption?
- Is it likely that the containment building does have some safety margins in addition to the assumptions stated?

Question 7

Regarding external explosions, the maximum shock wave overpressure the containment building can withstand, according to the Preliminary EIA Report, appears RATHER LOW (10 kPa). On the other hand, a higher value reported in the literature. Which value is correct, what is specified in the regulations in this respect? (UBA 2009)?

Concerning external explosions. According to the Report, the maximum shock wave which the reactor cover can sustain appears TO BE LOW ENOUGH (10 kPa). On the other hand, in the literature higher figures have been specified. Which of these figures are true? What is specified in the specifications in the given concrete case? (REPLIES 2010)

Summary of the Answer

It is clarified that the maximum shock wave which the cover can sustain has a pressure of 30 kPa, and a duration of impact of 1 second.

Assessment

The question has been answered in full.

The assumptions are similar to the German regulations. However, in Germany the pressure is assumed to reach a peak of 45 kPa during the first 100 ms, decrease to 30 kPa during the next 100 ms, and then remain constant during the following 800 ms.

Questions for follow-up

- How is the assumption of 30 kPa for 1 second justified; which considerations led to this assumption?
- Is it likely that the containment building does have some safety margins in addition to

the assumptions stated?

Question 8

How were the assumptions for the maximum design earthquake (intensity, ground acceleration) arrived at? (UBA 2009)

How have the figures been received for the maximum loading at earthquake (numerical score, ground acceleration)? (REPLIES 2010)

SUMMARY OF THE ANSWER

The answer is not completely clear. However, it appears that the design basis earthquake (SL-2 according to IAEA safety guides) has a maximum horizontal ground acceleration 0.25 g (as already mentioned in the Preliminary EIA Report, p. 41), corresponding to intensity 8 on the MSK-64 scale. The SL-1 earthquake (an earthquake which can be assumed to occur during the lifetime of the plant) is associated with a ground acceleration of 0.12 g (intensity 7 on the MSK-64 scale).

Regarding the determination of those figures, it is stated simply that this has been done "by means of calculations".

ASSESSMENT

The question has not been answered since no information was provided on how the seismic loads assumed had been arrived at.

For an earthquake of intensity 8 on the MSK-64 scale, a horizontal ground acceleration of 0.25 g appears somewhat low; however, this depends on the local characteristics of the underground.

QUESTIONS FOR FOLLOW-UP

- Could the methodology for determining the earthquake loads – in particular for the SL-2 earthquake – be explained? (Definition of seismic zones; determination of maximum earthquakes for each zone; determination of the attenuation functions etc.)

Question 9

Can a detailed description of the corium localization device be provided? How has the functioning of this device been proven (tests, computer simulations)? In particular, how can it be guaranteed that steam explosions can be avoided? (UBA 2009)

Can you present the description of the device of localization of the fusion? Whether the tests of this device took place and if yes, what sort of tests? For example, what are the guarantees of possibility to avoid steam explosion? (REPLIES 2010)

SUMMARY OF THE ANSWER

The language of the reply is somewhat unclear. It can be understood that the purpose of the corium localization device is the reduction of the radiological consequences of a large accident. The most important task in this case is the preservation of containment integrity.

As far as can be understood, this device is a vessel below the bottom of the reactor pressure vessel in which the corium is to be collected and cooled by water. Radioactive releases inside the containment and hydrogen formation is to be minimized by the design. Containment failure pressure should not be exceeded.

Functioning should be completely passive for at least 72 hours. It is pointed out that tests of this system have been carried out the Tianwan NPP in China.

It is pointed out that further details will become available in the course of the project.

ASSESSMENT

It appears that it was attempted to provide a basic idea concerning the essential features of the corium localization device in this reply. However, the language is not very clear and some aspects can only be guessed at.

It is possible that the answer can be seen as appropriate for the present stage of the project, if a better translation into English could be provided.

QUESTIONS FOR FOLLOW-UP

- It is conceivable that there will be no follow-up questions at the present stage after an adequate translation of the reply has been provided.

Question 10

Can the passive system for heat removal from the steam generators be described (design, operating parameters)? Which role does this system play in the context of long-time passive residual heat removal, what other systems are there for this purpose? How has their functioning been proven? (UBA 2009)

Can you present the description and characteristics of a passive system of bleeding from steam-gas generators (design, drawing, operating characteristics)? (REPLIES 2010)

SUMMARY OF THE ANSWER

As in the case of the previous question, the language of the reply is unclear.

As far as can be understood, the system in question is completely passive and is to function even in case of station blackout. It is to provide residual heat removal in case of a complete loss of feedwater.

In case of a primary-to-secondary leakage, the system is to minimize radioactive discharges. It consists of four parallel trains (4x33.3 % redundancy). The heat is transferred to tanks located outside the reactor containment. Containment failure pressure is to be avoided with the aid of this system.

ASSESSMENT

It appears that it was attempted to provide a basic idea concerning the essential features of the passive system for heat removal from the steam generators. However, the language is not very clear and some aspects can only be guessed at.

It is possible that the answer can be seen as appropriate for the present stage of the project, if a better translation into English could be provided.

QUESTIONS FOR FOLLOW-UP

- It is conceivable that there will be no follow-up questions at the present stage after an adequate translation of the reply has been provided.

PROJECT TARGETS & DESIGN LIMITS

Summary and assessment of the Preliminary EIA Report

The quantitative probabilistic targets appear to be fulfilled by the NPP-2006. However, it is not entirely clear from the Preliminary EIA Report that the CDF and LRF values provided really include all plant states (full power, low power and shut-down) and all initiators (internal and external). Regarding general safety requirements, it is important to note that the NPP-2006 was developed from NPP-92, which is certified by European Utility Requirements (EUR). Thus, it is plausible that NPP-2006 also fulfills the EUR. But the main safety characteristics and the concept of multiple barriers are described only in a very general manner. In this form, they apply to many operating NPP of Generation II.

The PWR types considered in the Preliminary EIA Report do not display significant differences regarding core damage frequency (CDF) and large release frequency (LRF). For the VVER-1200, a large release frequency of <1 E-7/a is given in a recent article (NEI 2009), one order of magnitude higher than the value in the Preliminary EIA Report. Results of Probabilistic Safety Assessments (PSA) in any case should only be taken as rough indicators of risk.

Question 11

Do the values for core damage frequency (CDF) and LARGE RELEASE FREQUENCY (LRF) provided for the VVER-1200 in the Preliminary EIA Report include all plant states (full power, low power and shut-down) and all initiators (internal and external)? (UBA 2009)

Do the figures on probability of serious damages of the active zone and PROBABILITY OF MAXIMUM PERMISSIBLE DISCHARGE presented in the Report on water- moderated water-cooled power reactor-1200 cover all operating conditions of the nuclear power plant (full capacity loading, low power operation and shutdown), as well as all initiating factors (internal and external)? (REPLIES 2010)

SUMMARY OF THE ANSWER

It is stated that the limit for the probability of a core damage accident is 10^{-6}/yr, and for large releases which require short-term countermeasures beyond the site 10^{-7}/yr.

Further information is provided which is, however, not clearly formulated in the replies. The following represents an interpretation by the authors: For releases larger than 100 TBq of Caesium-137, the probability must be lower than 10-7/yr. In case of releases with a probability of 10^{-7}/yr or higher, evacuation of the population should not become necessary at distances of more than 800 m from the reactor. Protective measures like sheltering or iodine prevention are limited to a 3 km zone around the NPP.

Finally, it is stated clearly that the probabilistic targets cover all operating conditions as well as all initiating factors.

It is not stated explicitly that the values for CDF and LRF for the VVER-1200 in the Preliminary EIA Report also cover all conditions and factors. (However, it appears likely that they do, in order to be consistent with the target values.)

ASSESSMENT

If the CDF and LRF values for the VVER-1200 cover all conditions and factors – as appears likely -, the question is answered, and also some additional explanations are provided.

QUESTIONS FOR FOLLOW-UP

- Can it be confirmed that the CDF and LRF values for the VVER-1200 provided in the Preliminary EIA Report cover all plant conditions as well as internal and external initiating factors?
- Could a listing be provided of the internal and external initiating factors which have been taken into account in the probabilistic safety analysis for the VVER-1200?

Question 12

Which uncertainty is associated with the PSA results? In particular, can the 95% fractiles of CDF and LRF be provided? (UBA 2009)

Unclear aspect is connected with probability of events. In particular, whether 95% quantile of probability of serious damages of the active zone and probability of maximum permissible discharge can be provided for? (REPLIES 2010)

SUMMARY OF THE ANSWER

Dose limits as well as probabilistic targets from EUR, INSAG and the US-APWR project are listed and compared to the values for the NPP-2006. It is pointed out that a probabilistic analysis will be carried out in the course of the further development of the Belarus NPP project.

ASSESSMENT

The information listed is not relevant for the question; the question is not answered.

The reference to the probabilistic analysis which is planned in the future could be understood to imply that an answer is not possible at the moment, but could be provided later.

However, since CDF and LRF values for the VVER-1200 have been provided in the Preliminary EIA Report, probabilistic analyses for the VVER-1200 clearly have already been performed and it should be possible to obtain an answer to the question, at the present time, based on those analyses.

QUESTIONS FOR FOLLOW-UP

- A probabilistic analysis clearly has already been performed for the VVER-1200, since values for CDF and LRF are available. Is it possible to provide information on the uncertainties of this probabilistic analysis (for example, by providing the 95 % fractiles)?
- When will the results of the specific probabilistic analysis for the Belarusian NPP be available?

Question 13

It is stated in the Preliminary EIA Report that the NPP-2006 fulfills the European Utility Requirements. Can more information be provided in this respect – in particular, regarding the source term which was assumed to check compliance with the "Criteria for Limited Impact" (CLI)? (UBA 2009)

The Report affirms that the Nuclear Power Plant-2006 corresponds to the requirements of EUR. Can you submit the additional information on the given problem? In particular, on the source of discharge which, how it is supposed, meets the requirements of « Criteria on the Limited Impact»? (REPLIES 2010)

Summary of the Answer

For a reference NPP-2006 (the "Baltic Nuclear Power Plant"), it has been checked whether the EUR "Criteria for Limited Impact" are complied with in case of an accident (probability below 10^{-6}/yr). This has been done for a source term including radionuclides which account for more than 90 % of the predicted radiation dose (according to EUR, 9 nuclides have to be considered).

The results of this verification procedure are presented in a table. It is shown that the criteria B1 – B3 (concerning emergency measures in the zones beyond 800 m and 3 km) are fulfilled. For those criteria, no values are given for the emission of individual nuclides; only the value of the criterion (which constitutes a weighed sum of the emissions of the individual nuclides which are considered) is provided.

In a second table, it is shown that the criteria on economic impact (which limit emissions of I-131, Cs-137 and Sr-90) are also fulfilled.

From the latter table, the source term assumed to check compliance with the criteria (at least the economic criteria; however, it seems very plausible to assume that the same source term has been used for all criteria) becomes apparent, at least regarding the three nuclides mentioned: 100 TBq of I-131, 10 TBq of Cs-137 and 0.12 TBq of Sr-90.

It is stated that those results are completely applicable to the Belarusian NPP.

Assessment

The questions for the source term is answered regarding three important nuclides, as well as relevant additional information provided.

It would be of interest to have a more complete source term – at least including the emissions of the 9 nuclides which have to be considered to check the compliance with criteria B1 – B3 (see above). It would also be of interest to have some indication concerning how this source term has been derived.

Questions for follow-up

- The source term has been provided for I-131, Cs-137 and Sr-90. Could the assumed emission values for other nuclides be provided as well, at least for Xe-133, Te-131m, Ru-103, La-140, Ce-141 and Ba-140 (those nuclides, together with the three mentioned first, constitute the group of nuclides which has to be considered to check compliance with the EUR criteria B1 – B3)?
- How was the source term which was used to check compliance with the EUR criteria determined?

Question 14

Can the requirements the NPP has to fulfill (apart from the EUR) be specified in more detail? (UBA 2009)

SUMMARY OF THE ANSWER

The reply provides an overview of the content of the two Technical Codes which list the requirements for nuclear installations in Belarus.

The principle of defense-in-depth is emphasized and the main safety objectives are presented. Safety classes, operational conditions and limits etc are discussed.

The main safety systems are presented and discussed:

- Control safety system (an automatic system operating without personnel intervention for 10 to 30 minutes).
- Protection system
- Localizing system
- Systems for supplying the safety systems (e.g. with electrical energy)

ASSESSMENT

The reply constitutes an adequate overview on a general level (even if the language is less than clear in some parts).

If a better English translation should be provided, the question can be regarded as answered. However, details concerning the safety requirements might come up in the discussion of various technical questions and should then be followed up in the respective context.

QUESTIONS FOR FOLLOW-UP

- None at the present stage if a better English translation of the reply could be provided.
- The issue of detailed safety requirements might become relevant in the context of various technical questions.

ACCIDENT ANALYSIS

Summary and assessment of the Preliminary EIA Report

Information concerning accidents in the NPP is distributed over different parts of the Preliminary EIA Report. There is no systematical analysis of design basis (DBA) and beyond design basis accidents (BDBA). Several BDBA source terms are presented, but without description of the initiating events and the progress of the emergency situation. Two Novovoronesh NPP severe accident scenarios are described without details and without reference. Furthermore it is unclear whether the source terms are derived from deterministic or probabilistic assessments. That is also the case for the “worst case” BDBA emission scenario in Chapter 5 “Transboundary impact”.

The conclusion of the Report that no greater source terms than the presented limited releases could occur is not sufficiently substantiated. For all existing reactors and also for the new Generation III reactors now under construction severe accidents with a release in the range of some percent of the radioactive Cesium inventory (2-20%) are not excluded. Even if the frequency of occurrence of accidents with a large release appears very small according to PSA, such severe accident source terms should be considered in the transboundary EIA.

Question 15

Which are the references for the source terms presented in the Preliminary EIA Report ? Why are larger source terms not discussed? (UBA 2009)

SUMMARY OF THE ANSWER

A list of references is presented, the documents are serially numbered.

Besides that a new source term is presented:

"The amount of discharge of the reference isotopes Iodine-131 = 3,1 E+15 and Caesium-137 =3,5E+14 to the environment has been chosen on the following basis: at out-of-design-basis accidents the integrity of a protective cover is being retained for at least 24 hours, leakings through the containment - 0,2 % per 24 hours and discharge lapses in a 24 hours period..... " (REPLIES 2010)

ASSESSMENT

Unfortunately it is not possible to match the reference papers to the information in chapter 5 of the Preliminary EIA Report.

The above cited new source term does not correspond with any of the ones used in the Preliminary EIA Report (maximum permitted release of section 5.1, emergency scenarios: Table 29 and 30, most heavy BDBA according to section 5.4). It also does not correspond to the source term which was used to check compliance with the EUR criteria (question 13). This new scenario assumes that the containment does not fail and the release from the containment to the environment is only due to containment leakage rate, which is assumed to be 2%and that after 24 hours the release is stopped.

QUESTIONS FOR FOLLOW-UP

- It would be helpful to amend the following table with the numbered documents:

Source term in EIA	**Reference Paper (no.)**
maximum design basis accident of section 5.1	
Table 29:	
Table 30:	
most heavy BDBA of section 5,4	
new source term of REPLIES 2010	

- The conditions which guarantee the limited impact are clearly defined in the Replies.
- The new source term of 3100 TBq I-131 and 350 TBq Cs-137 is about 30 times the release of table 2 (REPLIES page 12). But it is not explained, why this source term is chosen. Is this the most serious release of radioactive substances ?
- If this is the largest release due to an accident, which BDBA scenario would represent this worst case?

Question 16

Which source terms are worst case scenarios and which maximum permitted emissions? (UBA 2009)

SUMMARY OF THE ANSWER

The answer considers only the permissible discharge and presents a table with maximum permissible discharges from different institutions (Table 4).

ASSESSMENT

The question is answered regarding permissible discharge. However, the question on the potential maximum release due to an accident is still open.

QUESTIONS FOR FOLLOW-UP

Part of the question still open.

Question 17

Are results from preliminary safety reports of the NPP Leningrad 2 and Novovoronesh 2 both NPP-2006 (VVER 1200/491) – under construction available to the authors of the Preliminary EIA Report? IS THERE A LEVEL 2 PSA FOR THESE REACTORS? (UBA 2009)

Are the authors of the Report on EIA aware of the results of preliminary reports on safety at the Leningradskaya Nuclear Power Plant-2 and the Novovoronezhskaya Nuclear Power Plant-2 (Nuclear Power Plant- (Water- moderated water-cooled power reactor-1200/491)) which are at a stage of construction? (REPLIES 2010)

SUMMARY OF THE ANSWER

The material studied in course of preparation of the Preliminary EIA Report for the Belarusian NPP included environmental impact assessments and radiation protection, but no safety assessment.

ASSESSMENT

The lack of appropriate documents is likely to be the reason, why the question concerning the residual risk of large releases cannot be answered at present.

QUESTIONS FOR FOLLOW-UP

- Is it correct that no preliminary risk assessment , preliminary safety report and PSA was available as background for the Preliminary EIA Report ?

Question 18

Which DBA and BDBA scenarios have been analyzed by the designers of the NPP?

SUMMARY OF THE ANSWER

In the replies it is explained that the consequences of the most serious BDBA (beyond design base accident) have been considered. Among four types of BDBA the most serious consequences, from the point of view of the radiation damage result in BDBA of the third type: This is described as a station blackout, failure of the core cooling, which leads to serious damage of the fuel, but without containment breach. It is said to be an accident of level 5 on the international nuclear event scale (INES).

ASSESSMENT

Question is not answered. The analyzed design base accidents are unclear, and the three types of BDBA which are said to be considered are not even mentioned, besides the station black-out. Without information on the other scenarios considered, it is not comprehensible that early containment failures can be excluded.

QUESTIONS FOR FOLLOW-UP

- Is it possible to present a systematical listing of considered DBA and BDBA scenarios ?
- Is it possible to present more details on the types of BDBA scenarios (besides station blackout) ?
- Is the source term presented in the reply to question 15 th result of the most serious BDBA of the worst case scenario in the reply to question 18?

Question 19

Is it possible to describe the accident management features and procedures which shall guarantee the limited emission in case of a BDBA? (UBA 2009)

SUMMARY OF THE ANSWER

The answer describes the development of the BDBA scenario from above in more detail, defines the conditions which must be guaranteed in order to achieve a save state of the nuclear power plant within 7 days and lists the systems which are used for mitigation of the consequences.

ASSESSMENT

The decisive point of the answer is that the final lists of BDBA and their realistic analysis including estimation of probabilities are being established in the project of the Nuclear Power

Plant and in the Report on substantiation of safety of the Nuclear Power Plant. The given documents will be developed at the subsequent stages of designing of the Belarusian Nuclear Power Plant.

If there are no preliminary results of safety and risk analysis, important questions concerning the risk of transboundary impacts cannot be answered at the moment. For other generation 3 reactors (EPR, AP 1000), accidents with large releases are analyzed in safety reports, with the result that such events have a low probability of occurrence, but could have a substantial release of radionuclides.

QUESTIONS FOR FOLLOW-UP

- From the Austrian point of view it will be necessary to come back to the question of large release and large release frequency on a later stage of the plant design process.

RADIOACTIVE WASTE

Summary and assessment of the Preliminary EIA Report

Regarding waste management, only the volumes of solid and liquid radioactive wastes are provided, there is no information on the radioactive inventory.

The radioactive waste handling system is described without details. No interim storage for the spent fuel and no plans for radioactive waste disposal in Belarus are mentioned in the Preliminary EIA Report.

Question 20

What radioactivity levels do you use for the classification of radioactive wastes (high level, medium level, low level waste)? (UBA 2009)

SUMMARY OF THE ANSWER

Table 7 gives an overview on the classification of radioactive waste according to the Belarusian regulations. Concerning the regulation of surface contamination the dimensions are mistranslated: the first row – low activity is correct, but in the following it must be mSv not µSv (as it is in the table and in the Russian version of Replies).

QUESTIONS FOR FOLLOW-UP

Question is answered.

Question 21

Are there any plans for construction of an interim storage for spent fuel? (UBA 2009)

Summary of the Answer

The cooling pond in the reactor building provides storage for the spent fuel of ten years operation plus room for unloading the whole core. Finally the spent fuel will be removed from the NPP and reprocessed in the Russian Federation.

Assessment

Question is answered sufficiently

Question 22

Are there plans for the construction of a disposal facility for operational nuclear waste in Belarus? (UBA 2009)

Summary of the Answer

A regional center for radioactive waste management and storage is planned.

Assessment

Question is answered sufficiently