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NPP Olkiluoto-4

Evaluation of additional information received after the Bilateral Consultation

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ÖSTERREICHISCHES ÖKOLOGIE INSTITUT

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ÖSTERREICHISCHES ÖKOLOGIE INSTITUT

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1 INTRODUCTION

In the framework of the Espoo Convention Austria participates in the transboundary EIA concerning the construction of the fourth reactor at Olkiluoto NPP.

In the EIA process the Austrian Institute of Ecology, in cooperation with Dr. Helmut Hirsch, was engaged by the Austrian Federal Environmental Agency to assess the Environmental Impact Assessment Report of TVO. The findings of this evaluation are presented in an Expert Statement (WENISCH et al. 2008a). This Expert Statement includes a list of questions resulting from the evaluation of the EIA Report. Bilateral consultations were held in Helsinki on May 26th, 2008. During these consultations the questions raised by the Austrian representatives were discussed with the relevant Finnish authorities and the applicant TVO.

After the consultation the authors of the Expert Statement summarized the result of the consultation in a report (WENISCH et al. 2008b). In autumn 2008 the authors received additional information from the Finnish authorities and were charged with assessing to which extent this information is sufficient to answer the open questions raised at the Bilateral Consultation of May 26th, 2008.

The documents under evaluation are:

- Application for a Decision-in-Principle concerning the Construction of a Nuclear Power Plant Unit – Olkiluoto 4, TVO 2008;
- Supplement to the Environmental Impact Assessment Report Extension of the Olkiluoto nuclear power plant by a fourth unit, TVO 2008.

Chapter 2 presents the Expert's conclusions. A more detailed discussion of the issues most relevant for Austria are presented in chapter 3 and 4. These chapters are structured as follows:

- conclusion from Bilateral Consultation;
- Overview and Discussion of the issue in the new documents.

2 CONCLUSION

2.1 Reactor Options

The information presented on the different reactor options provides a good overview of the basic safety functions of the five reactor types. It is, however, not complete. There are at least two fields relevant for safety, which are not discussed: instrumentation and control, and protection against external events.

Five reactor types are included in the feasibility studies and presented in the "Application for a Decision-in-Principle", submitted by the project sponsor TVO. Other types of light water reactors, however, may - as indicated in the documentation - also come into question when choosing a plant alternative which should be implemented.

It was not the intention of the authors of "Application for a Decision-in-Principle" to provide detailed information, which could serve as basis of a comparison of reactor types. They state that more detailed descriptions of the plant alternatives will be submitted to the Radiation and Nuclear Safety Authority (STUK) for safety assessment. Presumably, this safety assessment will also include some kind of weighing of the alternatives. This could be achieved by measuring the reactor types against a set of detailed deterministic criteria, as well as by comparing the results of probabilistic safety assessment (PSA) studies.

As part of the basic information provided for each reactor type, PSA results for core damage frequency (CDF) and large release frequency (LRF) could have been included in the "Application for a Decision-in-Principle". CDF and LRF results are fraught with a considerable level of uncertainty and minor differences are not of great significance. However, differences of an order of magnitude or more would be an indicator of significantly different safety levels.

Therefore, data on CDF and LRF for the considered reactor options would be of considerable interest, but they are still missing.

The presentation of safety principles appears to be reasonably complete, on a very general level. Almost exclusively, it contains principles which are generally recognized internationally.

More detailed information would have been desirable in some cases, for example a list of obligatory design basis accidents (postulated accidents).

It is noteworthy that apart from beyond design basis accidents (postulated accidents), so-called design extension conditions are to be observed as an intermediate stage between DBAs and BDBAs. These conditions constitute either events with a common-cause failure, or events involving a complex combination of faults. The latter events are not comprehensively defined; complete loss of electrical power and loss of the ultimate heat sink are provided as examples.

2.2 Severe Accident Evaluation

In the Supplement to the EIA report an accident scenario is presented which results in a smaller release than the severe accident presented in the EIA Report.

Basically, this scenario is no alternative to the worst case release scenario as it was requested by Austria and Norway. In the report from the Bilateral Consultation it was requested that a consequence calculation for a source term corresponding to a severe, unmitigated accident should be performed. This source term should be selected in accordance with the results of analyses performed for such accidents for comparable reactor types since, according to present knowledge, such an accident cannot be excluded for any of the reactor types listed.

From the Austrian point of view, the question of the worst case release scenario is still open and should be answered during the progress of safety evaluations, even if this kind of release falls below the frequency limit of 5 E-7 per year as stipulated by the Finnish regulation.

3 PLANT ALTERNATIVE OPTIONS

3.1 Conclusions from the Bilateral Consultation

In the EIA the new NPP is regarded as a black box with standard impacts which has to fulfil the regulatory requirements. Four or five reactor designs are under closer consideration. At this stage of the procedure STUK has to assess whether there are any safety issues which would prevent the plant from meeting the Finnish requirements. STUK could probably recommend the exclusion of a certain design if it comes to the conclusion that the requirements will probably not be fulfilled.

In the process of issuing a construction licence, STUK will review the plant design applied for in the construction licence and can point out possible improvements. Feasibility studies will be included in the preliminary safety evaluation prepared by STUK for the Decision-in-Principle procedure and will be made public afterwards. From the Austrian point of view, this information should be made available before the Parliament's decision on the DIP.

In order to evaluate the residual risk associated with the OL 4 project, the following information should be provided:

For the different reactor types, the core damage frequency (and, as far as results are available, the large release frequency) should be reported and discussed in the further course of the procedures, as relevant input for the decisions to be taken. In spite of the fact that concrete, specific modifications reducing CDF (core damage frequency) and LRF (large release frequency) can be implemented at the reactor to be constructed at Olkiluoto, the generic values of these frequencies are relevant, since they provide the starting point for improvements, and since the potential for improvements is limited by the basic features of a reactor type.

3.2 Overview and Discussion of the Safety Functions of the Plant Alternatives Investigated in Appendix 7 of the "Application for a Decision-in-Principle"

3.2.1 Overview

Five reactor types are included in the feasibility studies and presented in the "Application for a Decision-in-Principle". However, in the "Application for a Decision-in-Principle" TVO states that other types of light water reactors, may also come into question when choosing a plant for the implementation.

In the following table, basic information as well as information on safety functions from Appendix 7 is compiled in a compact manner to facilitate comparisons between the reactor types. The ESBWR is a reactor type with mostly passive safety systems; the other types are evolutionary with some passive features.

Table:

	ABWR	ESBWR	APR 1400	APWR	EPR
Basic information	Toshiba-Westinghouse	GE Hitachi	Korea Hydro & Nuclear Power	Mitsubishi Heavy Industries	AREVA
	Approx. 1,650 MWe	Approx. 1,650 MWe	Approx. 1,450 MWe	Approx. 1,650 MWe	Approx. 1,650 MWe
	U.S. design certification 1997	U.S. design certification applied			
	3 units in operation in Japan	No units in op. or under constr.	4 units under constr. in South Korea	No units in op. or under constr.	(units under constr. not mentioned in App. 7)
Reactor shutdown	1 passive, 2 active systems - each sufficient to shut down reactor, with single failure	2 passive systems, one with 2 x 100%	1 passive, 1 active system plus 1 operational system	1 passive, 1 active system Separate prim. circ. depress. system, allows flooding with borated water by ECCS	1 passive, 1 active system
Decay heat removal from reactor under normal operating pressure	Isolation condenser with 4 heat exchangers Active HP system 3 x 100%	Isolation condenser 4 x 33,3%, each circuit tolerating single failure Shutdown cooling system 2 x 100%	Active emergency feedwater system 4 x 100% (2 electric pumps, 2 pumps with steam turbines)	Active emergency feedwater system 4 x 100% (2 electric pumps, 2 pumps with steam turbines)	Active emergency feedwater system 4 x 100%
Emergency core cooling	Active LP system 3 x 100% 8 of the 18 r/s valves for automatic depressurization	LP system 8 x 20% 10 of the 18 r/s valves for automatic depressurization plus 8 special depress. Valves Active operational system, 2 x 100%, also for LP	HP cooling system and accumulators 4 x 50% Injection directly into RPV 4 parallel relief lines for reducing primary circuit pressure LP cooling system with two circuits	HP cooling system and accumulators 4 x 50% Injection directly into RPV No LP ECC system; HP system can operate at low pressure 2 x 100% depress. System Active combined decay heat removal and containment spray system at low pressure, 4 x 50%	4 x 50% ECCS trains (IP, accumulators, LP) Three relief lines for depress., 3 x 100% LP: Active decay heat removal system 4 x 50%
Decay heat removal from containment building	Active system, 3 x 100% If steam released: Passive system with 4 heat exchangers	If steam released: Passive system with 6 heat exchangers, 72 hrs w/out replenishment Active op. system mentioned for ECC can also be used	Containment spray system with 2 circuits, 2 parallel pumps in each circuit	Active combined decay heat removal and containment spray system at low pressure, 4 x 50% Also cooling containment atm. by spraying water through fine nozzles	Active system, 4 x 50%
Severe accident management	Core catcher with passive automatic flooding Separate depress. System Full pressure containment able to retain hydrogen Filtered venting	Core catcher with passive automatic flooding 8 special depress. valves mentioned for ECC to prevent HP melt-through of RPV	Core catcher, 2 parallel lines for flooding Primary circuit depress. System Full pressure containment able to retain hydrogen Catalytic recombiners and igniters for H_2	Core catcher Dedicated primary circuit depress. System Active system for decay heat removal from containment, separate from combined system mentioned above Full pressure containment able to retain hydrogen Hydrogen igniters	Core catcher Active separate depr. syst. (1 x 100%) Independent active system for decay heat removal from containment after severe acc., 2 x 100% (can also cool struct. below RPV) Full pressure containment able to retain hydrogen Passive catalytic recombiners for H_2

3.2.2 Discussion

The information presented in Appendix 7 provides a good overview of the basic safety functions of the five reactor types. It is, however, not complete. There are at least two aspects relevant for safety, which are not discussed: instrumentation and control (information on the I&C-system employed; particularly concerning the degree of automation), and protection against external events.

Furthermore, the information provided does not permit a complete, meaningful comparison of the reactor types. It is clear that some reactor types have more different installations for fulfilling particular safety functions than others; also, that there is often a different degree of redundancy in the safety systems.

In several cases, however, only the number of parallel circuits is provided, and not the redundancy they provide. Also, there is no information on the reliability of the systems. Some systems can serve more than one purpose; there is no discussion whether this could be disadvantageous in certain accident scenarios.

It was not the intention of the authors of Appendix 7 to provide detailed information which could serve as a basis for a comparison of reactor types. They state that more detailed descriptions of the plant alternatives will be submitted to the Radiation and Nuclear Safety Authority (STUK) for safety assessment. Presumably, this safety assessment will also include some kind of weighing of the alternatives. This could be achieved by measuring the reactor types against a set of detailed deterministic criteria, as well as by comparing the results of probabilistic safety assessment (PSA) studies.

As part of the basic information provided for each reactor type, PSA results for core damage frequency (CDF) and large release frequency (LRF) could have been included in Appendix 7. CDF and LRF results are fraught with a considerable level of uncertainty and minor differences are not of great significance. However, differences of an order of magnitude or more would be an indicator of significantly different safety levels.

3.3 Overview and Discussion of the Safety Principles to be Applied for the New NPP in Finland, According to Appendix 8 of the “Application for a Decision-in-Principle”

3.3.1 Overview

After a brief introduction explaining which documents include the safety requirements in Finland (Decisions/Decrees of Council of State and YVL Guides of the Radiation and Nuclear Safety Authority), the safety principles to be applied are discussed.

In the section on general principles, the following topics are discussed:

- General objective;
- Safety culture;
- Quality management;
- Demonstration of compliance with safety regulations.

Regarding design requirements, the following issues are discussed:

- Levels of protection;
- Technical barriers for preventing the dispersion of radioactive materials;
- Fuel integrity;
- Primary circuit integrity;
- Containment building integrity;
- Safety functions;
- Avoiding human errors;
- Protection against external events and fires;
- Safety classification;
- Monitoring and control.

3.3.2 Discussion

The presentation of safety principles appears to be reasonably complete, on a very general level. Almost exclusively, it contains principles which are generally recognized internationally.

More detailed information would have been desirable in some cases, for example a list of obligatory design basis accidents (postulated accidents).

Also, it is not quite clear which degree of redundancy is required for safety systems. In the section on safety functions, the alternatives of 4 x 50% and 3 x 100% redundancy are cited, but as examples only. In the section on human errors, the N-2 principle is mentioned (simultaneous occurrence of maintenance and single failure), but again, it is not completely clear whether this principle has to be applied in all cases.

Similarly, it is mentioned that diversity is a principle observed in the design of safety systems without specifying to which extent this principle has to be implemented. It appears that only in case of the reactor shutdown system, two diverse systems are definitely required.

It is noteworthy that apart from beyond design basis accidents (postulated accidents), so-called design extension conditions are to be observed as an intermediate stage between DBAs and BDBAs. These conditions constitute either events with a common-cause failure, or events involving a complex combination of faults. The latter events are not comprehensively defined; complete loss of electrical power and loss of the ultimate heat sink are provided as examples.

Depressurization of the containment by filtered venting appears to be a requirement. Venting is not to begin earlier than 24 hours after the beginning of the accident.

For operator actions, the 30-minute rule generally applies. There is, however, the precondition that the safety systems operate automatically at least at their minimum capacity. If this is not the case, operator actions might be required earlier.

4 SEVERE ACCIDENT EVALUATION

4.1 Conclusions from the Bilateral Consultation

In the EIA the new NPP is regarded as a black box with standard impacts which has to fulfil the regulatory requirement. This requirement is satisfied if the possibility of a Cs-137 release of more than 100 TBq caused by a severe accident is extremely small (< 5 E-7/a). In order to assess the fulfilment of this requirement the applicant has to provide STUK with sufficient information according to the YVL-Guides[1].

The exemplary source term considered in the EIA Report (corresponding to a mitigated accident with limited releases, according to Finnish regulations) clearly is non-conservative.

In the further course of the procedures, a consequence calculation for a source term corresponding to a severe, unmitigated accident should be performed.

The source term should be selected in accordance with the results of analyses performed for such accidents for comparable reactor types, since according to present knowledge such an accident cannot be excluded for any of the reactor types listed.

The method and input data for the dose assessment based on the exemplary source term should, in the further course of the procedures, be documented in more detail than they are documented in the EIA Report, particularly regarding the dispersion model and the weather data. It should be ascertained that the dose assessment is based on a well documented, suitable program yielding meaningful results for distances up to 1,000 km, and going beyond mere extrapolation for large distances. For example, FLEXPART could be such a program.

4.2 Overview and Discussion of the Severe Accident Source Term According to Appendix 12 of the “Application for a Decision-in-Principle” and the Supplement to the Environmental Impact Assessment Report

4.2.1 Overview

Release during accidents is treated in chapter 2.3 of Appendix 12. The basis for the selection of the source term is explained in the same manner as it was in the EIA report. The limit posed by the Regulation GD 395/91 is chosen as the source term for the impact assessment of severe accidents.

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[1] YVL-Guides = Finnish Regulatory Guides on Nuclear Safety

The basic assumption is a severe damage of the reactor core, releasing a major part of the radioactive material into the containment. According to the design requirements, the containment building must keep the amount of radioactivity released into the environment below the limit specified in Regulation GD 395/91.

Detailed analyses are used to prove that the plant fulfils these requirements. These analyses are scheduled in connection with the application for a construction licence and operating licence.

In the Supplement to the EIA Report, an Assessment of the environmental impact of an accident less severe than the severe accident presented in the EIA Report is presented. An accident description is given in section 4.2.1 for an EPR type reactor as follows:

The initiating event is the break of the pressurizer surge line connected to a hot leg; failure of several systems is assumed to result in a core melt, making this a severe accident beyond plant design basis conditions. Melting of the reactor core, failure of the pressure vessel and relocation of the core melt within the spreading area inside the containment are assumed to occur during the accident.

It is assumed that radioactive fission products are released from the core to the containment building, both when the core melt is in the pressure vessel and when it has spread to the spreading compartment. Noble gases and volatile chemical elements (iodine and caesium) are typical substances released from a damaged fuel assembly and molten core.

In the case of the EPR-type reactor, the key activities in the management strategy for a severe reactor accident are: depressurization of the primary circuit before the pressure vessel fails; transport of the molten core material to a special spreading compartment inside the containment building, followed by solidification and long-term cooling ; removal of hydrogen by means of passive catalytic recombiners; removal of residual heat from the containment building by means of a separate cooling system .

The final state foreseen in the severe accident management strategy is that the core melt is solidified and coolable in the long term. The sooner the core is solidified, the smaller the amount of radioactive substances is released into the containment. In the accident scenario analysed it is assumed that the ventilation of the containment building is not in operation and filtered venting is not required.

The model and the results presented are based on a release analysis provided by the plant supplier.

The release presented for the EPR is based on the final safety analysis report currently under preparation.

The determined source term is given as:

Nuclide	TBq
Xe-133	400
Cs-137	0.0002
Cs-134	0.0003
I-131	0.003

4.2.2 Discussion

The accident description relies on several assumptions, and is intended to prove that if the severe accident management strategy works as foreseen, even a core melt will not result in a large release of radioactive substances. The description it is not detailed enough, however, to understand the sequences and duration of the phases of the presented scenario.

Furthermore, the scenario presented is no alternative to the worst case release scenario as it was requested by Austria and Norway. Therefore, the request formulated in the report of the Bilateral Consultation, namely that a consequence calculation for a source term corresponding to a severe, unmitigated accident should be performed, is still open.

This source term should be selected in accordance with the results of analyses performed for such accidents for comparable reactor types since, according to present knowledge, such an accident cannot be excluded for any of the reactor types listed.

From the Austrian point of view, the question of the worst case release scenario is still open and should be answered during the progress of safety evaluations, even if this kind of release falls below the frequency limit of 5 E-7 per year as stipulated by the Finnish regulation.

4.3 Overview and Discussion of the Method and Input Data for the Dose Assessment According to the Supplement to the Environmental Impact Assessment Report

4.3.1 Overview

The Supplement to the EIA Report gives a more specific presentation of the methods used for the accident analyses, and a brief assessment of an accident less severe than the severe accident presented in the EIA report.

In the description of the methods used for accident reviews, a more detailed description of the Gaussian plume model which has been used in the field near the plant is presented, and more details of the dose calculation.

Furthermore it is mentioned that the TRADOS model which had been used for the assessment of long range transport has already been abandoned and has been replaced by the SILAM model.

4.3.2 Discussion

The replacement of TRADOS by SILAM will certainly be an improvement for the assessment of transboundary impacts, because SILAM is a modern state-of-the-art dispersion calculation model, which is used with historical weather data.

5 REFERENCES

WENISCH, A.; HIRSCH, H.; MRAZ, G. & KROMP, R. (2008a): NPP Olkiluoto-4. Expert Statement to the EIA Report. REP-0162 Umweltbundesamt, Wien. Ordered by the Federal Ministry for Agriculture, Forestry, Environment and Watermanagement, Project management Department V/6 "Nuclear Coordination".

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Within the framework of the cross-border Environmental Impact Assessment (EIA) undertaken for the construction of new nuclear power plants in Finland, an Expert Statement was elaborated on behalf of the Umweltbundesamt.

The Expert Statement presents a review of additional information provided by the Finnish authorities and the project sponsor after the bilateral consultation. The main conclusion is: The information presented is not complete. Relevant issues concerning safety are not discussed properly. From the Austrian point of view, the question of the worst case release scenario still remains unanswered.

The Expert Statement concludes with open questions which should be resolved during the upcoming Finnish decision-making process at government level and the nuclear licensing process, respectively.

Documents for download:
http://www.umweltbundesamt.at/olkiluoto

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