PERSPEKTIVEN FÜR UMWELT & GESELLSCHAFT umweltbundesamt

NPP Fennovoima (Hanhikivi 1)

Expert Statement to the Environmental Impact Assessment Report

MINISTERIUM FÜR EIN LEBENSWERTES ÖSTERREICH

ÖSTERREICHISCHES ÖKOLOGIE INSTITUT

ENVIRONMENT AGENCY AUSTRIA umweltbundesamt

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By Order of the
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CONTENT

SUMMARY

Fennovoima Ltd. (hereinafter referred to as Fennovoima) plans to construct a 1,200 MWe nuclear power plant in the Hanhikivi headland (municipality of Pyhäjoki) at the west coast of Northern Finland. The proposed plant is a nuclear power plant of the type AES-2006/V491 from the Russian nuclear manufacturer Rosatom.

Environmental Impact Assessment

On 6 May 2010, the Council of State of Finland already granted Fennovoima a Decision-in-Principle for the construction of a nuclear power plant in accordance with the Nuclear Energy Act. The Finnish Parliament confirmed this decision on 1 July 2010.

The Environmental Impact Assessment (EIA) procedure for Fennovoima´s nuclear power plant project – a prerequisite for issuing the Decision-in-Principle – was carried out in 2008 and 2009. The Austrian Federal Ministry of Agriculture and Forestry, Environment and Water Management participated in this procedure. This original EIA evaluated the impacts of the nuclear power plant with the electric power of about 1,500–2,500 MWe, with one or two reactors, at three alternative locations. However, the AES-2006/V491 was not mentioned as one of the plant alternatives in the original Decision-in-Principal application and the original EIA, respectively.

Therefore, the Ministry of Employment and the Economy (MEE) required, among other things, an updated EIA. The government will decide on further measures after the assessments of these studies.

With reference to Art. 7 EIA Directive 2011/92/EU and Art. 3 Espoo Convention, the Austrian Federal Ministry of Agriculture and Forestry, Environment and Water Management informed the Finnish side on 26 November 2013 that Austria would take part in the transboundary Environmental Impact Assessment as the possibility of significant transboundary impacts of the project on Austria cannot be ruled out.

The EIA procedure is carried out in two main stages:

In the first phase the scope of the EIA procedure was laid down: The main document of this scoping phase was the EIA program (FENNOVOIMA 2013).

The main document of the second phase is the **environmental report** (EIA-Report). The Finnish authorities sent the EIA-Report (Environmental Impact Assessment Report for a Nuclear Power Plant, FENNOVOIMA 2014a or EIA-REPORT 2014) in February 2014 to provide the Austrian public and authorities with the possibility to comment on the EIA-Report. Furthermore, a non-technical German summary of the EIA-Report was transmitted (FENNOVOIMA 2014b).

The Umweltbundesamt (Environment Agency Austria) was commissioned by the Austrian Federal Ministry of Agriculture and Forestry, Environment and Water Management to coordinate the expert statement at hand and assist in organizational matters. The Österreichisches Ökologie-Institut (Austrian Institute of Ecology) in cooperation with Oda Becker, Helmut Hirsch and Adhipati Yudhistira Indradiningrat was assigned by the Umweltbundesamt to prepare this expert statement.

The **goal of the expert statement at hand** is to assess whether the updated EIA-Report proposing a power plant of the type AES-2006/V491 allows for making reliable conclusions about the potential transboundary impacts on the Austrian territory. Therefore, particularly safety features of the reactor type, severe accident management and the accident analysis with a focus on airborne transboundary emissions and the potential impact on Austria are discussed. Recommendations and questions are formulated.

Completeness of the documentation

In general, the EIA-Report seems to fulfill the **minimum requirements according to the EIA Directive 2011/92/EU and the Espoo-Convention**. However, some information necessary for the assessment of transboundary impacts/the comprehensibility of the given data is missing – these topics are discussed in detail in the chapters of the expert statement at hand.

No comprehensive **justification of the need** to construct another new nuclear power plant is made. According to the EIA Directive 2011/92/EU "An outline of the main alternatives studied by the developer and an indication of the main reasons for this choice, taking into account the environmental effects" is a necessary part of EIA-Report. Therefore, the applicant should clarify, whether the NPP project will be constructed to fulfill the domestic electricity demand or to export electricity – and verify the statements made.

Although the company has limited possibilities of taking significant action to conserve energy or to improve the efficiency of energy consumption (as stated in MEE (2013a)), a description of the **zero-option** and **alternatives** should be provided within any EIA procedure. The EIA-Report does not meet this requirement adequately, the missing information should be given. In the 2008 EIA-Report (FENNOVOIMA 2008), the three different reactors types and sites were the considered alternatives. The 2008 EIA-Report provided a comparison between the life-cycle CO_2 emissions of nuclear power and fossil fuel/natural gas, but the comparison with renewables was missing.

Procedure

It is the general practice in Finland, as laid down in the relevant regulations, that specific and detailed technical information concerning the reactor type(s) under consideration is not provided in the EIA-Report. After the Decision-in-Principle, a much more detailed assessment of the nuclear power plant project will be performed by STUK, in the course of the nuclear licensing procedure.

This course of action is predetermined and has to be accepted by the Austrian side. However, this does not exclude the possibility to provide more technical details already in the course of the EIA procedure. This is in particular true because the reactor type has already been chosen by the applicant and its feasibility study has already been provided to the STUK. The **questions formulated in the expert statement** at hand refer to the information that should be provided within the EIA procedure.

As the EIA procedure has to be completed before the Decision-in-Principle can be issued, most of the safety-relevant questions cannot be adequately answered within the EIA process. Whether the reactor will comply with the requirements discussed within the EIA process, can only be answered in the following approval procedure. Therefore, the **final statement of the MEE** should require the applicant to provide relevant information after the EIA procedure, especially on topics which came up during the EIA procedure but couldn't be answered at this stage.

It would be appreciated if information requested in the expert statement at hand could be provided once available.

Description of the project

It is planned to build a new nuclear power plant of approximately 1,200 MWe with a company of the Russian Rosatom Group as its supplier on Hanhikivi headland in Pyhäjoki.

The organization responsible for the project is Fennovoima, a Finnish nuclear power company established in 2007. Fennovoima´s owners are Voimaosakeyhtiö SF and RAOS Voima Oy, the Finnish subsidiary of Rosatom (34%).

In 2012, a total of about 60 industrial and commercial enterprises, as well as energy companies, were grouped under Voimaosakeyhtiö SF. During the last years, several shareholders left the consortium. At the end of March 2014, the Finnish ownership in Fennovoima sank below 50%. It is important to note that the Finnish state requires a clear majority of Finnish ownership in the project.

The nuclear power plant is scheduled to begin operation in 2024. Because of the ambitious project schedule, it has to be assumed that the date of commission will have to be postponed. The reference plant for the design of Hanhikivi 1 is the nuclear power plant Leningrad-II in Russia. The original date of commissioning of the units Leningrad-II was 2013/2014, but this date has been postponed for several years.

Reactor type

The reactor type selected for the planned new NPP at Hanhikivi is Rosatom's AES-2006/V491 (or VVER-1200/V491). With regard to the reasoning for the selection of this reactor type, appendix 2 of the EIA-Report only briefly states that Fennovoima chose AES-2006/V491 over AES-2006/V392M as the reference plant because its defense-in-depth approach in relation to redundancy and independency between system trains is closer to the Finnish regulatory requirements. No further elaboration regarding the reasoning for the **selection of AES-2006/V491** is provided.

The **WENRA safety objectives** for new power reactors are taken **as benchmark** in this expert statement. They should ensure that the NPP which will be licensed in future will fulfill higher safety standards across Europe compared to the existing plants; they reflect the current state of the art in nuclear safety.

The fulfillment of the WENRA safety objectives is not discussed in the EIA-Report. They are only briefly referred to, as being covered by the current Finnish regulations. There is no discussion of the fulfillment of the individual objectives in the context of the Hanhikivi project. In section 5.2.1 of the expert statement at hand, available information on the VVER-1200/V491 is compiled and evaluated, in connection with the application of WENRA safety objectives. From the information available, it can be recognized that considerable efforts have been undertaken in relation to the fulfillment of WENRA safety objectives. But to allow a definite assessment whether the WENRA safety objectives have been fulfilled by the reactor type in consideration, more detailed information is still needed. The expert statement at hand identifies a number of **challenges** in the discussion **regarding** the fulfillment of **WENRA safety objectives**.

For instance, it is not clear to which extent a systematic consideration of multiple failures has been performed. Separation of the I&C systems supporting different levels of defense-in-depth also has not been made clear so far in the available documents. The discussion on the fulfillment of the WENRA safety objectives also concerns the functioning and reliability of the safety systems and features of the AES-2006/V491, such as the core catcher.

Lessons learned from Fukushima are important aspects which also should be taken into consideration in new NPP projects. Therefore, the expert statement at hand also discusses the application of **lessons learned from the Fukushima accident** in AES-2006/V491. Additionally, a brief **comparison between AES-2006/V491** and one of the reactor types from earlier EIA (i.e.: EPR) is discussed in the expert statement.

It would be appreciated if information pertinent to the further course and the results of the preliminary assessment by STUK and the nuclear licensing procedure could be provided once available, with the focus on the fulfillment of WENRA safety objectives for new power reactors, on the efforts undertaken in this respect, and the challenges encountered.

Another issue of interest would be a detailed discussion of the application of lessons learned from Fukushima for the reactor type VVER-1200/V491.

In general, it is recommended that the concept of practical elimination is applied consistently in the safety requirements for the new nuclear unit. Practical elimination of accident sequences has to be demonstrated with state-of-the-art probabilistic and deterministic methods, fully taking into account the corresponding publications of WENRA.

Site evaluation incl. external hazards

According to the Decision-in-Principle 2010, based on STUK´s preliminary safety assessment, the Hanhikivi headland in Pyhäjoki is a suitable location for a nuclear power plant. In October 2013, Fennovoima submitted a report to STUK, which describes the most recent changes on site and any changed information important for the plant site's safety. STUK is currently preparing the statement, which will be given to the Ministry of Employment and the Economy in spring 2014.

Sea-related phenomena constitute the most important external hazards at the Hanhikivi plant site. It is questionable that the envisaged site elevation will ensure sufficient protection against external flooding caused by extreme sea water levels and waves; thus, it is recommended to consider the implementation of appropriate further protection of the plant site.

Considering the existing sea-related hazards that could cause the loss of the heat sink (e.g. extremely low or high sea water level; clogging caused due biological fouling or frazil ice) and the statement of WENRA (2013), it is recommended to consider the implementation of an alternative heat sink (for example a groundwater well).

In the expert statement on the EIA program (UMWELTBUNDESAMT 2013), it is recommended to perform a **comprehensive site evaluation** to enhance the safety margins of the nuclear power plant against natural hazards. This issue has been addressed in the EIA-Report, but without providing any details. A systematic consideration of all possible combinations of natural phenomena is also missing.

The **return frequency of extreme natural phenomena** according to YVL Guide B.7 complies with the state of the art – but only in case the degree of confidence of the estimated frequency is justified.

According to the EIA-Report, the nuclear power plant will be designed to withstand the impact of a **crash of a large commercial airplane**. However, the availability of the necessary safety functions after the crash of a large airplane, in particular considering the potential common vulnerability of the safety trains in the safety building of the AES-2006, are not demonstrated yet.

It has to be assumed that both a comprehensive site evaluation and the design solution concerning external hazards are not available in the course of the EIA procedure. A complete evaluation of these issues can be expected from the assessments and analyses which will be performed in the course of the licensing procedure. The results of the **preliminary safety assessment by STUK** will be helpful in this respect. **It would be appreciated if information pertinent to this topic could be provided once available**.

Accident analysis and trans-boundary impacts

The **source term** used to evaluate the consequences of a postulated severe accident has been defined according to the Government Decree on Nuclear Safety (717/2013) as a release containing 100 TBq Cs-137. The expectation value for a release bigger than this shall be less than once in 2,000,000 years (5E-7/yr).

The expert statement on the EIA program (UMWELTBUNDESAMT 2013) stated that **severe accidents with releases considerably higher than 100 TBq** of Cs-137 cannot be excluded for the AES-2006, even if their probability is required to be below 5E-7/yr. Only results of detailed safety assessments for the reactor would allow to exclude a larger source term – in case it can be proven with a high degree of confidence that such a larger source term is extremely unlikely to occur. Such safety assessments, however, are not provided in the EIA-Report and not available for the AES-2006 yet.

Rough calculations on the consequences of a severe accident of the AES-2006 at the Hanhikivi site based on source terms evaluated in the flexRISK project (54,460 TBq of Cs-137) as well as in a study of the Norwegian Radiation Protection Authority (2,800 TBq of Cs-137) presented in UMWELTBUNDESAMT (2013) show **possible consequences in Austria**. With the release of 100 TBq of Cs-137 such consequences would not be expected.

Thus, the expert statement on the EIA program recommended including a **conservative worst-case release scenario** in the EIA-Report, in addition to the limited release scenario according to Finnish regulations, since their effects can be widespread and long-lasting and even countries not directly bordering Finland, like Austria, can be affected.

This recommendation was observed to a considerable extent. On request of Ministry of Employment and the Economy (MEE), in addition to the postulated accident with a release of 100 TBq of Cs-137, a severe accident with a release of the five-fold magnitude was considered.

It is highly appreciated that the consequence of a release of more than 100 TBq Cs-137 is considered in the EIA-Report. However, a release of 500 TBq of Cs-137 represents the lower limit of a release corresponding to an INES 7 accident. Severe accidents with larger releases cannot be judged as practically eliminated on the basis of the information provided or available. Thus, the release of 500 TBq of Cs-137 does not represent a true worst-case accident scenario.

However, even the INES 7 accident as considered in the EIA-Report indicates consequences for the Austrian territory in case of a severe accident at the Hanhikivi site.

The EIA procedure in Finland does not stipulate a presentation and discussion of detailed information on the reactor type(s) in question and their technical specifications. Therefore, it has to be assumed that it will not be possible to obtain information about specific accident scenarios in the course of the EIA procedure. **It would be appreciated if information pertinent to severe accident scenarios with source terms, timing and duration of the release and calculated frequency of occurrence (including uncertainties) could be provided once available. It is recommended to perform a conservative worst-case release scenario** which is based on specific accident analyses of the AES-2006/V-491 once this information is available.

Radioactive Waste Management

Radioactive waste management is presented in the EIA-Report in a general manner as Fennovoima has not yet developed a comprehensive nuclear waste management strategy. This approach is in line with the Finnish Nuclear Energy Act -more concrete plans are currently being developed and will presumably only be finalized after the EIA procedure.

The recommendations and questions of the expert statement to the EIA program (UMWELTBUNDESAMT 2013) were only partly fulfilled/answered by the EIA-Report.

Whenever possible, additional information on RAW management should be given within the EIA procedure – information already available through the plan of final disposal of SNF currently being prepared should be made available.

The following questions should be answered within the EIA procedure:

- *When will the choice of interim storage be made? Is there a currently favored option?*
- *When can the decision about the final disposal strategy of spent fuel be made available?*

- *In case Fennovoima has to construct its own final disposal facility: (When) can the progress and timetable of Fennovoima's EIA on SNF disposal be made available?*

It would be appreciated if information pertinent to the following topics could be provided once available:

Spent Fuel

- Only a rough estimate of the quantity of spent fuel is made in the EIA-Report. Data on the expected quantities of spent fuel need to be more concrete.
- Fennovoima needs to present the planned type of interim storage for SNF (wet or dry storage), its capacity and the schedule of the construction works.
- In the EIA-Report, the stated interim storage time of SNF is a minimum of 40 years. As the duration of interim storage is important for the evaluation of the risk, concrete information need to be provided.
- The decision about the final disposal strategy of SNF is of interest from the Austrian point of view. In case Fennovoima has to construct its own final disposal facility, a time schedule as well as information on the sites envisaged should be provided and the progress and timetable of Fennovoima's EIA on SNF disposal should be made available.

LILW

- More information on the LILW waste treatment plants and on the geological suitability of the on-site LILW repository should be given.

ZUSAMMENFASSUNG

Fennovoima Ltd. plant die Errichtung eines Reaktors mit einer Leistung von 1.200 MWe auf der Halbinsel Hanhikivi (Gemeinde Pyhäjoki) an der Westküste Nordfinnlands. Das geplante Kraftwerk ist ein Kernkraftwerk vom Typ AES-2006/V491 des russischen Herstellers Rosatom.

Umweltverträglichkeitsprüfung

Am 6. Mai 2010 hat der Finnische Staatsrat dem Unternehmen Fennovoima bereits die Grundsatzgenehmigung für die Errichtung eines Atomkraftwerks gemäß dem Atomenergiegesetz erteilt. Das Finnische Parlament bestätigte diese Entscheidung am 1. Juli 2010.

Das Umweltverträglichkeitsverfahren (UVP) für Fennovoimas Atomkraftwerksprojekt wurde 2008 bis 2009 durchgeführt und stellt eine Voraussetzung für die Erteilung der Grundsatzgenehmigung dar. Das Bundesministerium für Land- und Forstwirtschaft, Umwelt und Wasserwirtschaft beteiligte sich an diesem Verfahren. Diese ursprüngliche UVP prüfte die Umweltauswirkungen von Atomkraftwerken mit einer Leistung zwischen 1.500 und 2.500 MWe, mit einem oder zwei Reaktoren auf drei verschiedenen Standorten. Allerdings wurde der AES-2006/V491 nicht als eine der Alternativen in dem ursprünglichen Antrag für die Grundsatzgenehmigung angeführt, bzw. der ursprünglichen UVP.

Daher forderte das Ministerium für Arbeit und Wirtschaft unter anderem eine aktualisierte UVP. Die Regierung will nach Prüfung dieser Berichte die weiteren Schritte beschließen.

Gemäß Art. 7 der UVP-Richtlinie 2012/92/EU und Art. 3 der ESPOO Konvention informiert das Bundesministerium für Land- und Forstwirtschaft, Umwelt und Wasserwirtschaft die finnische Seite am 26. November 2013 darüber, dass sich Österreich an der grenzüberschreitenden UVP beteiligen wird, dass mögliche bedeutende grenzüberschreitende Auswirkungen des Projekts auf Österreich nicht ausgeschlossen werden können.

Das UVP-Verfahren wird in zwei Etappen durchgeführt:

In der ersten Etappe wurde der Umfang des UVP-Verfahrens festgelegt: Das Hauptdokument dieser Scoping-Phase war das UVP-Programm (FENNOVOIMA 2013).

Das Hauptdokument der zweiten Etappe ist der Umweltbericht (UVP-Bericht). Die finnischen Behörden übermittelten den UVP-Bericht (Environmental Impact Assessment Report for a Nuclear Power Plant, FENNOVOIMA 2014a oder EIA-REPORT 2014) im Februar 2014, um der österreichischen Öffentlichkeit und den Behörden die Möglichkeit zu gewähren, Stellungnahmen zum UVP-Bericht abzugeben. Daneben wurde auch die allgemein verständliche Zusammenfassung des UVP-Berichts übermittelt (FENNOVOIMA 2014b).

Das Umweltbundesamt wurde vom Bundesministerium für Land- und Forstwirtschaft, Umwelt und Wasserwirtschaft beauftragt das vorliegende Expertengutachten zu koordinieren und Unterstützung bei organisatorischen Fragen zu leisten. Das Österreichische Ökologie-Institut wurde beauftragt in Kooperation mit Oda Becker, Helmut Hirsch und Adhipati-Yudhistira Indradiningrat dieses Expertengutachten vorzubereiten.

Das **Ziel des vorliegenden Expertengutachtens** ist es zu prüfen, ob anhand des aktualisierten UVP-Berichts für das geplante Kraftwerk vom Typ AES-2006/V491 verlässliche Schlussfolgerungen über die möglichen grenzüberschreitenden Umweltauswirkungen auf das Gebiet Österreichs gezogen werden können. Daher werden insbesondere die Sicherheitsmerkmale des Reaktortyps, Management schwerer Unfälle und Unfallanalysen mit dem Schwerpunkt auf die Emissionen und deren Ausbreitung über die Luft und die möglichen Folgen für Österreich betrachtet. Empfehlungen und Fragen werden formuliert.

Vollständigkeit der Dokumentation

Allgemein scheint der UVP-Bericht die **Minimalanforderungen laut UVP-Richtlinie 2011/92/EU und gemäß Espoo-Konvention** zu erfüllen. Dennoch fehlen Angaben für die Bewertung der grenzüberschreitenden Auswirkungen wie auch die Nachvollziehbarkeit der angeführten Angaben – diese Fragen werden jeweils in den Kapiteln des vorliegenden Expertengutachtens im Details behandelt.

Es wurde keine nachvollziehbare **Bedarfsbegründung** vorlegt. Gemäß der UVP-Richtlinie 2011/92/EU ist eine „Übersicht über die wichtigsten anderweitigen vom Projektträger geprüften Lösungsmöglichkeiten und Angabe der wesentlichen Auswahlgründe im Hinblick auf die Umweltauswirkungen“ im UVP-Bericht vorzulegen. Daher sollte der Projektträger klären, ob das KKW-Projekt errichtet wird um heimischen Strombedarf zu decken oder Strom zu exportieren – und dies mit Nachweisen zu belegen.

Wenn das Unternehmen auch nur über beschränkte Möglichkeiten verfügt, wirksamere Maßnahmen zur Energieeinsparung umzusetzen oder die Effizienz des Energieverbrauchs zu erhöhen (wie in MEE (2013a) festgehalten) so sollte dennoch im UVP-Bereich auch eine Beschreibung der **Null-Variante** und der **Alternativen** enthalten sein. Der UVP-Bericht kommt dieser Anforderung nicht nach, doch ist die Information zur Verfügung zu stellen. Im UVP-Bericht von 2008 (FENNOVOIMA 2008) wurden drei verschiedene Reaktortypen und Standorte als Alternativen betrachtet. Dieser UVP-Bericht stellte auch Vergleiche zwischen den CO_2-Emissionen von Atomenergie, fossilen Brennstoffen/Erdgas über den gesamten Lebenszyklus an, doch die Erneuerbaren fehlten.

Verfahren

In Finnland sehen die relevanten Vorschriften nicht vor, dass spezifische und detaillierte technische Informationen zum Reaktortyp(en), die in Erwägung gezogen werden, im UVP-Bericht angeführt werden. Nach der Grundsatzgenehmigung wird ein wesentlich genaueres Gutachten des Atomkraftwerkprojekts von STUK im Rahmen der atomrechtlichen Genehmigung erstellt werden.

Dies ist eine Tatsache, die von der österreichischen Seite zu respektieren ist. Dennoch ist dies kein Hindernisgrund mehr technische Details bereits während des UVP-Verfahrens zu Verfügung zu stellen. Das gilt insbesondere in diesem

Fall, weil der Reaktortyp vom Projektträger bereits ausgewählt und die Machbarkeitsstudie bereits STUK übermittelt wurde. Die **in dieser Expertenstellungnahme** formulierten Fragen beziehen sich auf Angaben, die im UVP-Verfahren zur Verfügung gestellt werden sollten.

Ein abgeschlossenes UVP-Verfahren ist die Vorbedingung für die Erteilung der Grundsatzgenehmigung. Daher kann der Großteil der sicherheitsrelevanten Fragen nicht während des UVP-Verfahrens behandelt werden. Ob der Reaktor den Anforderungen aus dem UVP-Verfahren entspricht, kann nur im anschließenden Genehmigungsverfahren beantwortet werden. Daher sollte die **Abschließende Stellungnahme des MEE** vorsehen, dass der Projektträger die relevanten Informationen nach dem UVP-Verfahren zur Verfügung stellt, insbesondere zu Fragen, die während der UVP behandelt wurden, jedoch in dieser Etappe nicht beantwortet werden konnten.

Es wäre wünschenswert der österreichischen Seite die Informationen, die diese Stellungnahme anfordert zu übermitteln, sobald sie verfügbar sind.

Projektbeschreibung

Geplant ist die Errichtung eines neuen russischen Atomkraftwerks am Standort der Halbinsel Hanhikivi im Bezirk Pyhäjoki mit etwa 1.200 MWe durch ein Unternehmen der russischen Rosatom-Gruppe und dessen Lieferanten.

Die für das Projekt verantwortliche Organisation ist Fennovoima, ein finnisches Atomenergie-Unternehmen, welches 2007 gegründet wurde. Die Eigentümer von Fennovoima sind Voimaosakeyhtiö SF und RAOS Voima Oy, die finnische Tochtergesellschaft von Rosatom (34 %).

Im Jahre 2012 schlossen sich etwa 60 Industrie- und Handelsunternehmen wie auch Energieunternehmen zu Voimaosakeyhtiö SF zusammen. In den vergangenen Jahren verließen einige Shareholder das Konsortium und Ende März 2014 fiel der finnische Anteil an Fennovoima unter 50 %. Der finnische Staat schreibt jedoch eine deutliche finnische Mehrheit in der Eigentumsstruktur des Projekts vor.

Das Kernkraftwerk sollte 2024 mit dem Betrieb beginnen. Aufgrund des ehrgeizigen Zeitplans des Projekts ist davon auszugehen, dass das Datum der Kommissionierung verschoben werden muss. Das Referenzkraftwerk für das Design von Hanhikivi 1 ist das Atomkraftwerk Leningrad-II in Russland. Das ursprüngliche Datum für die Kommissionierung der Blöcke des KKW Leningrad II lautete 2013/2014, doch wurde dieser Zeitpunkt um mehrere Jahre verschoben.

Reaktortyp

Für das neue KKW in Hanhikivi wurde der AES-2006/V491 (oder VVER-1200/V491) vom Hersteller Rosatom ausgewählt. Zu den Gründen für die Auswahl dieses Reaktortyps findet sich in Anhang 2 des UVP-Berichts die kurze Aussage, dass Fennovoima dem AES-2006/V491 gegenüber dem AES-2006/V392M als Referenzkraftwerk den Vorzug gab, weil die Tiefengestaffelte Verteidigung (Defense-in-Depth) bei Redundanz und Unabhängigkeit der Systemstränge stärker den finnischen Vorschriften entspricht. Es wird keinerlei weitere Begründung dafür angeführt, warum die Entscheidung für den **AES-2006/V491** gefallen ist.

Die **WENRA-Sicherheitsziele** für neue Atomkraftwerke werden als **Maßstab** für diese Expertenstellungnahme herangezogen. Sie sollen sicherstellen, dass die in Zukunft in Europa genehmigten Kernkraftwerke höhere Sicherheitsanforderungen erfüllen als die bestehenden, denn sie reflektieren den aktuellen Stand der Technik in der nuklearen Sicherheit.

Der UVP-Bericht geht nicht auf die Einhaltung der WENRA-Sicherheitsziele ein. Sie werden nur kurz als durch die aktuell geltenden finnischen Vorschriften abgedeckt erwähnt. Die Beschreibung der Einhaltung individueller Ziele im Zusammenhang mit dem Hanhikivi-Projekt fehlt. In Abschnitt 5.2.1 der vorliegenden Expertenstellungnahme wird verfügbare Information über den VVER-1200/V491 zusammengestellt und unter Anwendung der WENRA-Sicherheitsziele bewertet. Aus der verfügbaren Information ist erkennbar, dass beachtliche Anstrengungen unternommen wurden, um die WENRA-Sicherheitsziele zu erfüllen. Um allerdings eine endgültige Bewertung darüber zu ermöglichen, ob die WENRA-Sicherheitsziele erfüllt werden, ist mehr Detailinformation über den Reaktortyp nötig. Diese Expertenstellungnahme hat eine Reihe von **offenen Fragen** zur Diskussion darüber identifiziert, ob die **WENRA-Sicherheitsziele** erreicht werden.

So ist zum Beispiel nicht klar, in welchem Umfang eine systematische Betrachtung von Mehrfachversagen durchgeführt worden ist. Die Trennung des I&C Systems für die einzelnen Ebenen der Tiefengestaffelten Verteidigung wurde in den bisher zur Verfügung gestellten Unterlagen nicht klar dargestellt. Die Frage der Einhaltung der WENRA-Sicherheitsziele betrifft auch das Funktionieren und die Verlässlichkeit der Sicherheitssysteme und anderer Bereiche des AES-2006/V491 wie etwa den Core Catcher.

Die Lehren von Fukushima sind ebenso wichtige Aspekte, die auch in dem neuen KKW-Projekt zu berücksichtigen sind. Daher betrachtet das vorliegende Expertengutachten auch die Umsetzung der **Lehren des Unfalls von Fukushima** beim AES-2006/V491. Zusätzlich enthält das vorliegende Gutachten auch einen **kurzen Vergleich des AES-2006/V491** mit einem der Reaktortypen aus der früheren UVP (d. h. des EPR).

Es wäre wünschenswert, wenn relevante Informationen über den weiteren Verlauf und die Ergebnisse der vorläufigen Bewertung von STUK und dem nuklearen Genehmigungsverfahren sobald sie vorliegen zur Verfügung gestellt würden, nämlich mit dem Schwerpunkt auf der Einhaltung der WENRA-Sicherheitsziele für neue Reaktoren und den damit verbundenen Anstrengungen und Herausforderungen.

Von Interesse wäre ebenso eine detaillierte Betrachtung der Umsetzung der Lehren des Unfalls von Fukushima beim Reaktortyp VVER-1200/V491.

Als allgemeiner Grundsatz wird empfohlen das Konzept des Praktischen Ausschlusses bei den Sicherheitsanforderungen für die neuen Atomkraftwerke durchgehend anzuwenden. Der Praktische Ausschluss von Unfallabfolgen hat mit probabilistischen und deterministischen Methoden auf dem Stand der Technik durchgeführt zu werden, die entsprechenden Publikationen von WENRA sind dabei vollständig anzuwenden.

Standortprüfung inkl. externer Gefahren

Der Grundsatzgenehmigung von 2010 zufolge, die auf der vorläufigen Sicherheitsprüfung von STUK basiert, eignet sich die Halbinsel Hanhikivi im Bezirk Pyhäjoki als Standort für ein AKW. Im Oktober 2013 übermittelte Fennovoima einen Bericht an STUK, der die jüngsten Änderungen am Standort beschreibt, und neue Informationen zur Sicherheit des Standorts beinhaltet. STUK bereitet zurzeit seine Stellungnahme vor, die dem Ministerium für Arbeit und Wirtschaft im Frühling 2014 übergeben werden wird.

Naturereignisse durch das Meer stellen die wichtigste externe Gefährdung am Standort Hanhikivi dar. Es ist anzuzweifeln, ob die angestrebte Standorterhöhung einen ausreichenden Schutz gegen externe Überflutungen durch extrem hohe Meeresspiegel und hohe Wellen bieten kann. Daher wird empfohlen die Durchführung weiterer geeigneter Schutzmaßnahmen des Standorts in Erwägung zu ziehen.

Angesichts der bestehenden Gefährdung durch die Lage an der See, wodurch ein Verlust der Wärmesenke verursacht werden könnte (z. B. extrem niedriger oder hoher Meeresspiegel, Verstopfung durch organische Fäulnisprozesse oder Frazileis) und die Stellungnahme von WENRA (2013) wird empfohlen, die Implementierung einer alternativen Wärmesenke (z. B. ein Grundwasserbrunnen) zu erwägen.

Die Fachstellungnahme zum UVP-Programm (UMWELTBUNDESAMT 2013) empfiehlt die Durchführung einer **komplexen Standortprüfung** um die Sicherheitsreserven des KKW gegen Naturereignisse zu verbessern. Diese Frage wurde im UVP-Bericht aufgegriffen, doch ohne auf Details einzugehen. Eine systematische Betrachtung aller möglichen Kombinationen von Naturereignissen fehlt ebenso.

Die **Eintrittshäufigkeit von extremen Naturereignissen** entspricht laut YVL Guide B.7 dem Stand der Technik – allerdings nur in dem Ausmaß, wie man der Einschätzung der Eintrittshäufigkeit vertrauen kann.

Laut dem UVP-Bericht wird das KKW so ausgelegt werden, dass es die Folgen eines **Absturzes eines großen Verkehrsflugzeugs** beherrscht. Allerdings wurde die Verfügbarkeit der notwendigen Sicherheitsfunktionen nach dem Absturz eines großen Flugzeugs noch nicht nachgewiesen, vor allem in Anbetracht der Vulnerabilität der gemeinsamen Sicherheitsstränge in den Sicherheitsgebäuden des AES-2006.

Es ist davon auszugehen, dass weder die komplexe Standortprüfung noch die Designlösung für die externen Gefährdungen während des UVP-Verfahrens zur Verfügung stehen werden. Eine komplexe Evaluierung dieser Fragen wird bei den Prüfungen und Analysen zu erwarten sein, die während des Genehmigungsverfahrens durchgeführt werden. Die Ergebnisse des **vorläufigen Sicherheitsberichts von STUK** werden in dieser Frage hilfreich sein. **Es wäre sehr wünschenswert für diese Frage relevante Informationen zu erhalten, sobald diese vorliegen.**

Unfallanalysen und grenzüberschreitende Auswirkungen

Der **Quellterm** für die Bewertung der Folgen von postulierten schweren Unfällen wurde gemäß der Regierungsverordnung über die nukleare Sicherheit (717/2013) als eine Freisetzung von 100 TBq Cs-137 definiert. Die erwartete Häufigkeit für eine darüber hinaus gehende Freisetzung sollte unter einmal in 2 Millionen Jahren (5E-7/a) zu liegen kommen.

Die Expertenstellungnahme zum UVP-Programm (UMWELTBUNDESAMT 2013) hielt fest, dass **schwere Unfälle mit Freisetzungen deutlich über 100 TBq** Cs-137 für den AES-2006 nicht ausgeschlossen werden können, auch wenn deren Häufigkeit mit unter 5E-7/a angesetzt wird. Nur Ergebnisse detaillierter Sicherheitsanalysen für den Reaktor würden es erlauben einen größeren Quellterm auszuschließen, wenn nachgewiesen werden kann, dass ein so großer Quellterm nur extrem unwahrscheinlich eintreten kann. Derartige Sicherheitsbewertungen enthält der UVP-Bericht allerdings nicht und diese stehen für den AES-2006 noch nicht zur Verfügung.

Grobe Berechnungen der Konsequenzen von schweren Unfällen des AES-2006 am Hanhikivi-Standort basierend auf Quelltermen wurden im flexRISK-Projekt (54 460 TBq an Cs-137) angestellt wie auch in einer Studie der Norwegischen Strahlenschutzbehörde (2 800 TBq an Cs-137), die in der Stellungnahme UMWELTBUNDESAMT (2013) angeführt ist, zeigen **mögliche Folgen für Österreich.** Bei einer Freisetzung von 100 TBq an Cs-137 würden solche Folgen nicht erwartet werden.

Daher enthielt die Expertenstellungnahme zum UVP-Programm die Empfehlung, im UVP-Bericht zusätzlich zu den eingeschränkten Freisetzungsszenarien gemäß der Finnischen Verordnung auch ein **konservatives Worst-Case Szenario für die Freisetzung** zu erstellen, da die Folgen weitläufig und langfristig sein können und selbst Länder betroffen sein können, die nicht an Finnland angrenzen, wie etwa Österreich.

Dieser Empfehlung wurde zu einem großen Teil Folge geleistet. Auf Anforderung des Ministeriums für Arbeit und Wirtschaft (MEE) wurde zusätzlich zum postulierten Unfall mit einer Freisetzung von 100 TBq an Cs-137 ein schwerer Unfall fünffacher Größe betrachtet.

Es ist sehr begrüßenswert, dass die Folgen der Freisetzung von über 100 TBq Cs-137 im UVP-Bericht betrachtet werden. Dennoch ist die Freisetzung von 500 TBq an Cs-137 das untere Limit einer Freisetzung bei einem Unfall, der der INES-Skala 7 entspricht. Schwere Unfälle mit größeren Freisetzungen können aufgrund der zur Verfügung gestellten oder verfügbaren Information nicht ausgeschlossen werden. Daher stellt die Freisetzung von 500 TBq an Cs-137 nicht wirklich das Worst-Case-Szenario für Unfälle dar.

Dennoch zeigt der INES 7-Unfall im UVP-Bericht, dass mit Folgen für das österreichische Staatsgebiet bei einem Unfall am Hanhikivi-Standort zu rechnen ist.

Das UVP-Verfahren in Finnland sieht keine verpflichtende Präsentation und Behandlung detaillierter Information des Reaktortyps (der Reaktortypen) und deren technischer Spezifikation vor. Daher ist davon auszugehen, dass es nicht möglich sein wird Informationen über spezifische Unfallszenarien während des UVP-Verfahren zu erhalten. **Es wäre sehr wünschenswert, wenn relevante Informationen zu Szenarien für Schwere Unfälle mit Quelltermen, Dauer**

und Zeitpunkt der Freisetzung und Berechnung zu den Häufigkeiten (einschließlich Unsicherheiten) übermittelt werden, sobald diese zur Verfügung stehen. Es wird empfohlen ein konservatives Worst-Case-Scenario für Freisetzungen auszuarbeiten, basierend auf spezifischen Unfallanalysen für den AES-2006/V-491, sobald diese Informationen vorliegen.

Entsorgung radioaktiver Abfälle

Die Entsorgung radioaktiver Abfälle präsentiert der UVP-Bericht allgemein, da Fennovoima noch keine umfassende Strategie zur Entsorgung radioaktiver Abfälle ausgearbeitet hat. Diese Vorgangsweise entspricht dem Finnischen Atomenergiegesetz – konkretere Pläne werden zurzeit entwickelt und wohl erst nach Abschluss des UVP-Verfahrens fertig gestellt werden.

Die Empfehlungen und Fragen der Expertenstellungnahme zum UVP-Programm (UMWELTBUNDESAMT 2013) wurden nur teilweise im UVP-Bericht erfüllt bzw. beantwortet.

Sobald wie möglich sollte zusätzliche Information zur Entsorgung von Atommüll während des UVP-Verfahrens zur Verfügung gestellt werden – die bereits über den Plan für die Entsorgung von abgebrannten Brennstäben bekannte Information ist zu übermitteln.

Folgende Fragen sind während des UVP-Verfahrens zu beantworten:

- *Wann wird die Auswahl des Zwischenlagers erfolgen? Gibt es eine zurzeit favorisierte Option?*
- *Wann wird die Entscheidung über die Strategie für die Endlagerung von abgebrannten Brennstäben bekannt gegeben?*
- *Wann kann über Fortschritte und den Zeitplan für die UVP von Fennovoima für das Endlager für abgebrannte Brennstäbe berichtet werden?*

Es wäre wünschenswert, relevante Informationen zu den folgenden Themen übermittelt zu bekommen, sobald sie zur Verfügung stehen:

Abgebrannter Nuklearbrennstoff

- Der UVP-Bericht beinhaltet nur eine sehr grobe Schätzung der Menge an anfallendem abgebranntem Brennstoff. Die Angaben zu den erwarteten Mengen sind konkreter auszuführen.
- Fennovoima hat die Art der Zwischenlagerung für die abgebrannten Brennstäbe (Nass- oder Trockenlagerung), die Kapazität und den Zeitplan für die Errichtungsarbeiten zu präsentieren.
- Im UVP-Bericht wird die Dauer der Zwischenlagerung von abgebrannten Brennstäben mit minimal 40 Jahren genannt. Da die Dauer der Zwischenlagerung für die Risikobewertung wichtig ist, bedarf es der Nennung konkreter Angaben.
- Die Entscheidung über die Endlagerungsstrategie für abgebrannte Brennstäbe ist für Österreich von Interesse. Falls Fennovoima verpflichtet sein sollte seine eigene Endlagerstätte zu errichten, so ist ein Zeitplan als auch Information über die betrachteten Standorte anzuführen, wie auch über Fortschritt und Zeitplan von der UVP von Fennovoima für die Endlagerstätte.

Niedrig- und mittelaktive Abfälle

- Mehr Informationen über die Anlagen zur Aufbereitung von niedrig- und mittelaktiven Abfällen und die geologische Eignung für ein Endlager für die niedrig - und mittelaktiven Abfälle im Areal des KKW sind anzuführen.

1 INTRODUCTION

Fennovoima Ltd. (hereinafter referred to as Fennovoima) plans to construct a 1,200 MWe nuclear power plant in the Hanhikivi headland (municipality of Pyhäjoki) at the west coast of Northern Finland. The proposed plant is a nuclear power plant of the type AES-2006/V491 from the Russian nuclear manufacturer Rosatom.

Environmental Impact Assessment

On 6 May 2010, the Council of State of Finland granted Fennovoima a Decision-in-Principle for the construction of a nuclear power plant in accordance with the Nuclear Energy Act (990/1987). The Finnish Parliament confirmed the Decision-in-Principle on 1 July 2010.

The Environmental Impact Assessment (EIA) procedure for Fennovoima´s nuclear power plant project – a prerequisite for issuing the Decision-in-Principle – was carried out in 2008 and 2009. The Austrian Federal Ministry of Agriculture and Forestry, Environment and Water Management participated in this procedure. This original EIA evaluated the impacts of the nuclear power plant with the electric power of about 1,500–2,500 MWe, with one or two reactors at three alternative locations. However, the AES-2006/V491 was not mentioned as one of the plant alternatives in the original Decision-in-Principal application and the original EIA, respectively.

Therefore, the Ministry of Employment and the Economy (MEE) required an updated EIA, a safety assessment and Pyhäjoki municipality's view on the matter. The government will decide on further measures after the assessments of these studies (MEE 2013a).

On 27 September 2013, the Finnish Ministry of Environment notified Austria of the new EIA procedure. The Finnish Ministry of Environment is responsible for the international consultation within the Environmental Impact Assessment, the Finnish Ministry of Ministry of Employment and the Economy (MEE) act as coordinating authority for the overall EIA process.

With reference to Art. 7 EIA Directive 2011/92/EU and Art. 3 Espoo Convention, the Austrian Federal Ministry of Agriculture and Forestry, Environment and Water Management informed the Finnish side on 26 November 2013 that Austria would take part in the transboundary Environmental Impact Assessment as the possibility of significant transboundary impacts of the projects on Austria cannot be ruled out.

The EIA procedure is carried out in two main stages:

In the **first** (shorter) phase the scope of the main document of the EIA procedure – the EIA-Report – was laid down: The main document of this **scoping phase**, the **EIA program** (FENNOVOIMA 2013), contained a study on the current state of the project area, as well as a work program stating which impacts shall be studied and how the studies shall be performed within the EIA-Report.

The Umweltbundesamt (Environment Agency Austria) assigned Oda Becker, scientific consultant, to elaborate an expert statement (UMWELTBUNDESAMT 2013) to the documents presented by Finland within the scoping phase, in particular Fennovoima´s EIA program published in September 2013 (FENNOVOIMA

2013). The Expert Statement to the EIA program was submitted to Finland on 26 November 2013.

The review of the document focused mainly on the proposed safety and risk analysis. The aim was to assess if the EIA-Report will allow making reliable conclusions about the potential impact of transboundary emissions.

The Ministry of Employment and the Economy issued a statement (MEE 2013a) summarizing the statements of different organizations and giving its own statement concerning the content of the EIA-Report.

The main document of the **second phase** is the Environmental report (EIA-Report). The Finnish authorities sent the **EIA-Report** (Environmental Impact Assessment Report for a Nuclear Power Plant, FENNOVOIMA 2014a or EIA-REPORT 2014) in February 2014 to provide the Austrian public and authorities the possibility to comment on the report. Furthermore, a non-technical German summary of the EIA-Report was transmitted (FENNOVOIMA 2014b).

Expert statement at hand

The Umweltbundesamt (Environment Agency Austria) was commissioned by the Austrian Federal Ministry of Agriculture and Forestry, Environment and Water Management to coordinate the expert statement at hand and assist in organizational matters. The Austrian Institute of Ecology (Österreichisches Ökologie-Institut) in cooperation with Oda Becker, Helmut Hirsch and Adhipati-Yudhistira Indradiningrat was assigned by the Umweltbundesamt to prepare this expert statement.

The **goal** of the expert statement at hand is to assess if the updated EIA-Report proposing a power plant of the type AES-2006/V491 allows for making reliable conclusions about the potential impact of transboundary emissions. Therefore, particularly safety features, severe accident management and the accident analysis with a focus on airborne transboundary emissions and the potential impact to Austria are discussed. Recommendations and questions are formulated.

This expert statement is **structured** as follows: After a summary in English and German, comments on the completeness of the documentation and the procedure are given in chapters 2 and 3. The project is described in chapter 4. In chapter 5, the reactor type (AES-2006/V491) considered for Fennovoima´s nuclear power plant is discussed in detail. Site conditions and external hazards are presented in chapter 5.1. Chapter 7 deals with the accident analysis with focus on possible transboundary consequences. In chapter 8, the management of the radioactive waste is discussed briefly. Recommendations and questions are summarized in chapters 9 and 10.

Expert statements within the previous EIA procedure

The Austrian Federal Ministry of Agriculture and Forestry, Environment and Water Management also participated in Fennovoima´s 2008 EIA procedure and commissioned the Umweltbundesamt (Environment Agency Austria) to coordinate an expert statement on this topic. The Austrian Institute of Ecology in cooperation with Helmut Hirsch and Petra Seibert implemented this expert statement on behalf of the Umweltbundesamt. (UMWELTBUNDESAMT 2008).

A bilateral consultation was held in Helsinki on 28 January 2009. During this consultation, the questions of the Austrian side were discussed with the competent Finnish authorities and the applicant Fennovoima. Information presented at the bilateral consultation was assessed in the experts' report on the consultation (UMWELTBUNDESAMT 2010).

In summer 2009, further documents in conjunction with the ongoing decision-making process were made available to the Austrian side as an important contribution to keeping the Austrian side well-informed. The evaluation of these supplements was published in September 2009 (UMWELTBUNDESAMT 2009).

The expert statement at hand is also based on the above-mentioned reports.

2 COMPLETENESS OF DOCUMENTATION

2.1 Legal requirements

The transboundary EIA procedure is regulated within different legal bases. On the level of international law, the Espoo Convention is applied – Finland accepted the Espoo Convention in 1995, the 1st and 2nd amendments in 2014.[1] At EU level, the EIA Directive 2011/92/EU is in force. The EIA Directive as well as the Espoo Convention contain a number of provisions concerning the content of EIA-Reports.

The expert statement at hand does not aim at carrying out a comprehensive assessment on whether or not the EIA-Report contains all the necessary information according to the aforementioned regulations - only the fulfillment of selected criteria is evaluated. The following table gives an overview on the legal requirements and whether or not the topic is covered in the expert statement. If it is, the table refers to the chapters of the expert statement which deal with the topic in question or gives a short answer to the topic right away.

Table 1: Requirements according to Espoo-Convention and EIA Directive 2011/92/EU concerning the content of EIA-Reports.

Criterion	Espoo-Convention Annex II	Directive 2011/92/EU Annex IV	Chapter
Description of the project	a) A description of the proposed activity and its purpose	1. A description of the project, including in particular the physical characteristics and an estimate, by type and quantity, of expected residues and emissions resulting from the operation of the proposed project	Chapter 4 Chapter 5 Chapter 5.1 Chapter 8
Alternatives und Zero Alternative	b) A description, where appropriate, of reasonable alternatives (for example, locational or technological) to the proposed activity and also the no-action alternative	2. An outline of the main alternatives studied by the developer and an indication of the main reasons for this choice, taking into account the environmental effects	see chapter 2.2
State of the Environment	c) Description of the environment likely to be significantly affected by the proposed activity and its alternatives	3. A description of the aspects of the environment likely to be significantly affected by the proposed project	not considered within the expert statement

[1] http://www.unece.org/env/eia/ratification/convmap.html

Criterion	Espoo-Convention Annex II	Directive 2011/92/EU Annex IV	Chapter
Environmental Impact	d) A description of the potential environmental impact of the proposed activity and its alternatives and an estimation of its significance	4. A description of the likely significant effects of the proposed project on the environment resulting from e.g. the emission of pollutants or the use of natural resources	only concerning accidents and transboundary impacts: Chapter 5 Chapter 5.1 Chapter 7
Mitigation measures	e) A description of mitigation measures to keep adverse environmental impact to a minimum	6. A description of the measures envisaged to prevent, reduce and where possible offset any significant adverse effects on the environment.	only concerning accidents and transboundary impacts: Chapter 5 Chapter 5.1 Chapter 7
Methods	f) An explicit indication of predictive methods and underlying assumptions as well as the relevant environmental data used	5. The description by the developer of the forecasting methods used to assess the effects on the environment referred to in point 4.	only concerning technical solution/accidents/ transboundary impacts: Chapter 5 Chapter 5.1 Chapter 7
Gaps in knowledge and uncertainties	g) An identification of gaps in knowledge and uncertainties encountered in compiling the required information	8. An indication of any difficulties (technical deficiencies or lack of know-how) encountered by the developer in compiling the required information.	only concerning technical solution/accidents/ transboundary impacts: Chapter 4 Chapter 5 Chapter 5.1 Chapter 7 Chapter 8
Monitoring	h) Where appropriate, an outline for monitoring and management programs and any plans for post-project analysis		not considered within the expert statement
Non-technical summary	i) A non-technical summary including a visual presentation as appropriate (maps, graphs, etc.).	7. A non-technical summary of the information provided under headings 1 to 6.	A non technical summary was provided in English and German
Transboundary Impacts		Art. 7 Par. 1a of the EIA Directives stipulates that together with the description of the project, any available information on its possible transboundary impact has to be given.	Chapter 7

In general, the EIA-Report seems to fulfill the minimum requirements according to the EIA Directive 2011/92/EU and the Espoo-Convention. However, some information necessary for the assessment of transboundary impacts/the comprehensibility of the given data is missing. Details on this topic are given in the chapters referred to above.

2.2 Alternatives and zero-alternative

2.2.1 Treatment in the EIA-Report

Justification of the project

Chapter "1.3 Purpose and justification of the project" of the EIA-REPORT (2014, p. 30) gives a short justification of the project. It says that nuclear power is a cost-effective way to produce electricity, that the price is stable and foreseeable, that the project will improve the national security of supply (in 2012 20% of the electricity had to be imported), increase competition on the Finnish electricity market and has a significant positive impact on regional economy. Being a carbon dioxide-free electricity production method, nuclear power supports the achievements of the Finnish climate goals (last update of the national energy and climate strategy in 2013).

The EIA-REPORT (2014, p. 215-216) summarizes the prognosis for the **demand for electrical energy** in Finland (94 TWh by 2020, 102 TWh by 2030 vs. 88 TWh in 2010). The capacity of the units with expiring lifetime until 2030 is given. The future development of the Finnish electricity production with only one new nuclear unit (Olkiluoto-3) is presented in a graph. The EIA-REPORT (2014, p. 216) states: "If the nuclear power plant units for which Decisions-in-Principle have been made are constructed, the Finnish electricity production capacity will increase significantly more than shown in the graph." No comparison between the rise in demand and this significant increase is given.

The EIA-REPORT (2014, p. 216-217) gives a short overview on Finland's **Energy Efficiency** Plan: The new Energy Efficiency Directive entered into force in 2012 and aims at savings of around 1 TWh per year in addition to the already established targets. The potential of energy savings by Fennovoima's stakholders is estimated to be low when compared to their electricity demand.

Zero-option

The considered **zero-option** (zero alternative) is not implementing Fennovoima's nuclear power plant project and covering the electricity corresponding to the nuclear power plant's capacity with separate electricity production mostly in Finland (EIA-REPORT 2014, p. 31).

The EIA-REPORT (2014, p. 217) states the following on the impact of the zero-option: "part of the production volume will probably be covered with separate production of electricity in Finland. A large part of the electricity that would have been produced by the nuclear power plant will be replaced with separate production based on fossil fuels in the other Nordic countries and continental Europe. […] If the Fennovoima nuclear power plant project is not implemented, the same volume of electricity must be produced by other means. The assumption is that, in such a case, 20% of the Fennovoima nuclear power plant's planned electricity production capacity of 9.5 TWh would be replaced with separate electricity production in Finland. The remaining 80% would be produced abroad. Separate electricity production is assumed to be coal condensate production". Under these assumptions, the production to replace the Fennovoima nuclear power plant in Finland and abroad would cause a little less than seven million tons of CO_2 emissions, a little less than six thousand tons of both sulfur dioxide and nitrogen oxide emissions, and a little less than a thousand tons of small particle emissions per year.

Alternatives

The EIA implemented in 2008 studied four alternative locations for the NPP including Hanhikivi. In the current EIA, the location has already been decided upon, so the considered alternative consists of the key characteristics of the 1,800 MW plant studied in the EIA of 2008 in comparison with the current AES-2006/V491 project with 1,200 MW (EIA-REPORT 2014, p. 31).

Chapter 8 of the EIA-Report compares these two alternatives concerning their environmental impact. The EIA-REPORT (2014, p. 222) states: "According to the assessment results, the size of the plant or the specified plant type does not change the environmental impacts in any significant way."

2.2.2 Discussion

Justification of the project

The **expert statement to the EIA program** (UMWELTBUNDESAMT 2013, p. 18) recommended to include a comprehensive justification of the need to construct another new nuclear power plant in the updated EIA-Report. The reasoning behind that was the Government's statement that "nuclear power will not be constructed in this country for the purpose of permanent export of electricity" in the Finnish Climate and Energy Strategy from 2008 (MEE 2008). Although the 2013 update of this strategy (MEE 2013b) does not repeat this statement, it emphasizes the importance of self-sufficiency in electricity sourcing. The Strategy states that in the 2020s, the self-sufficiency target will be met, when the nuclear power units that have been granted favorable decisions-in-principle become operational.

The EIA-Report only gives little information on the justification of the project concerning the need for energy self-sufficiency. Concerning future capacity, the EIA-Report only states that if the nuclear power plant units for which Decisions-in-Principle have been made are constructed, the Finnish electricity production capacity will increase significantly more than illustrated in graph 7-53 of the EIA-Report which shows only Olkiluoto-3 as new capacity. No comparison of this increase with the given prognosis of the energy demand is made. A comprehensive **justification of** the **need** to construct a new NPP concerning energy self-sufficiency is therefore **missing**. An assessment of Finland's energy policy is not aim of the expert statement at hand, this topic will therefore not be further elaborated.

The topic of **energy efficiency** is addressed briefly in the EIA-Report, but **no** statement on the possible role of energy efficiency in an **alternative scenario** to the project is made.

No comprehensive justification of the need to implement the project is given. This is in accordance with the conditions for the EIA-Report made in MEE (2013a). According to the EIA Directive 2011/92/EU "An outline of the main alternatives studied by the developer and an indication of the main reasons for this choice, taking into account the environmental effects" is a necessary part of EIA-Report. Therefore, the applicant should clarify, whether the NPP project will be constructed to fulfill the domestic electricity demand or to export electricity – and verify the statements made.

Furthermore, in the Decision-in-Principle, the Government must also consider the issue from the perspective of the overall good of society, with special attention paid to e.g. the need for the nuclear facility project with respect to the country's energy supply. (EIA-REPORT 2014, p. 103)

Within the chapter "1.3 Purpose and justification of the project", the EIA-REPORT (2014, p. 30) calls nuclear power a **carbon dioxide-free electricity method**. The expert team would like to note that while nuclear power can – under some conditions – be called a low-carbon technology, it is not carbon-free, as the whole life cycle of the nuclear fuel chain has to be considered (WALLNER et al. 2011). The EIA-REPORT (2014, p. 214) states that the CO_2-emissions have been in the range of 2–40 g CO_2-equivalents/kWh – in this section of the EIA-Report the life cycle CO_2-emissions have thus been taken into account.

The statement that nuclear power is a **cost-effective** way to produce electricity (EIA-REPORT 2014, p. 30) is a currently highly disputed topic, especially considering the current rise in construction prices in European NPP projects and the discussion on long-term state guarantees on electricity prices in Great Britain concerning their nuclear new build plans. Furthermore, no evidence of this statement is given in the EIA-Report – the chapter on electricity production and cost structure in the Nordic electricity market (EIA-REPORT 2014, p. 217) gives only a comparison of the variable electricity costs (figure 7-54). This is contrary to the fact, that the fixed costs of nuclear power constitute a bigger share of the total costs (about 2/3 according to THOMAS 2010 and ROGNER 2012) than the variable costs.

Zero-Option/Alternatives

The expert statement to the EIA program (UMWELTBUNDESAMT 2013, p. 18) recommended to "include into the EIA-Report a **comprehensive comparison of all electricity production technologies and the options of saving energy, efficiency enhancement and demand side management**. The EIA-Report should also include information on the cost structure of the project and the technological alternatives."

The **zero-option** calculates additional emissions that would arise without the implementation of the Fennovoima project. In the calculations, 100% of the replaced volume of electricity is to be produced by coal-fired power plants. Thus, energy efficiency and energy savings are not considered in the zero-option at all, neither are other energy production technologies like renewables. The zero-option therefore constitutes a very limited worst-case scenario.

Also, the considered **alternatives** are very limited: only the 1,800 MW plant studied in the EIA of 2008 is used as alternative which is compared to current AES-2006/V491 project with 1,200 MW.

2.2.3 Conclusions/Recommendations

No comprehensive **justification of the need** to construct another new nuclear power plant is made. According to the EIA Directive 2011/92/EU "An outline of the main alternatives studied by the developer and an indication of the main reasons for this choice, taking into account the environmental effects" is a necessary part of EIA-Report. Therefore, the applicant should clarify, whether the NPP project will be constructed to fulfill the domestic electricity demand or to export electricity – and verify the statements made.

Although the company has limited possibilities of taking significant action to conserve energy or to improve the efficiency of energy consumption (as stated in MEE (2013a)), a description of the **zero-option** and **alternatives** should be provided within any EIA procedure. The EIA-Report does not meet this requirement adequately, the missing information should be given. In the 2008 EIA-Report (FENNOVOIMA 2008), the three different reactors types and sites were considered as alternatives. The 2008 EIA-Report provided a comparison between the life-cycle CO_2 emissions of nuclear power and fossil fuel/natural gas, but the comparison with renewables was also missing.

Furthermore, the expert team recommends not to call nuclear power a carbon dioxide-free technology, nuclear power can be referred to as low-carbon at maximum.

3 PROCEDURE

3.1 Treatment in the EIA-Report

Licenses/permits/decisions related to the construction/operation of NPPs in Finland

A number of licenses, permits, notifications and decisions is necessary related to the construction and operation of a nuclear power plant. EIA-REPORT (2014, p. 101-107) gives an overview of these processes.

EIA procedure

The EIA stands at the beginning of this process – it is carried out parallel to several other overlapping procedures. The EIA procedure has already been briefly described in the introduction (chapter 1).

Decision-in-Principle

The licensing procedure of nuclear facilities is described in the Finnish Nuclear Energy Act (990/1987). According to this Nuclear Energy Act, the construction of a nuclear power plant shall require a government Decision-in-Principle to ensure that the project is in line with the overall good of society. The EIA procedure has to be completed before the Decision-in-Principle concerning a new nuclear power plant can be issued. The EIA procedure itself does not involve any project-related decisions, but its objective is to generate information to back up decision-making.

In the original EIA procedure, three sites including the Hanhikivi site were under discussion. Furthermore, three different types of reactors were considered: Areva´s EPR; Toshiba´s ABWR and Areva´s KERENA (FENNOVOIMA 2008).

Since the AES-2006/V491 was not mentioned as one of the plant alternatives in Fennovoima’s original Decision-in-Principal application, the Ministry of Employment and the Economy (MEE) has required the following additional studies (EIA-REPORT 2014, p.103):

- Fennovoima shall update the environmental impact assessments of the project,
- STUK shall assess the safety of the current plant alternative,
- the municipality of Pyhäjoki shall make a statement on the issue, and
- the MEE shall arrange a public hearing in accordance with the Nuclear Energy Act.

After these clarifications have been completed, a statement will be made regarding the fact whether the Decision-in-Principle in force also covers the present plant alternative, or whether the Decision-in-Principle shall be reintroduced to Parliament for new parliamentary proceedings (EIA-REPORT 2014, p. 103). The Decision-in-Principle from 2010 laid down as a condition that Fennovoima must apply for a construction license within 5 years of the Parliament upholding the Decision-in-Principle, thus not later than on 30 June 2015. (MEE 2013a)

3.2 Discussion

It is the general practice in Finland, as laid down in the relevant regulations, that specific and detailed technical information concerning the reactor type(s) under consideration is not provided in the EIA-Report. Rather, the new nuclear power plant is regarded as a black box, which has to comply with the regulatory requirements. This approach was also followed in the Fennovoima´s 2008 EIA-Report. Several overlapping procedures are ongoing, besides the EIA procedure. Preparation for the Decision-in-Principle includes feasibility studies which have to be provided by the applicant. Based on these documents, the regulatory authority STUK has to assess whether there are safety issues to be foreseen which could prevent the plant meeting the Finnish requirements. After the Decision-in-Principle, a much more detailed assessment of the nuclear power plant project will be performed by STUK, in the course of the nuclear licensing procedure.

This course of action is predetermined and has to be accepted by the Austrian side. However, this does not exclude the possibility to provide more technical details already in the course of the EIA procedure. This is in particular true because the background of this EIA procedure is quite different. The reactor type has already been chosen by the applicant and its feasibility study has already been provided to the STUK. Thus, it is should be possible to present some more details on e.g. the reactor type, in particular concerning safety analysis and a plant-specific severe accident scenarios within the **EIA procedure**. The **questions** formulated in the expert statement at hand refer to this information.

An **exchange of information** between the competent authorities of Austria and Finland covering the results of feasibility studies and safety assessments to follow the still ongoing procedures was recommended in the context of the 2008 EIA procedure. Austria highly appreciated that relevant documents[2] were made available to the Austrian side as an important contribution to keeping the Austrian side well-informed.

As the EIA procedure has to be completed before the Decision-in-Principle can be issued, most of the safety-relevant questions cannot be adequately answered within the EIA process. Whether the reactor will comply with the requirements discussed within the EIA process, can only be answered in the following approval procedure.

3.3 Conclusions/Recommendations

After the Decision-in-Principle, a much more detailed assessment of the nuclear power plant project will be performed by STUK, in the course of the nuclear licensing procedure.

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[2] (a) Decision-in-Principle application by Fennovoima; b) Statement of MEE on the EIA; c) Decision- in-Principle including STUK´s report on the feasibility study of the reactor types for all applications (UMWELTBUNDESAMT 2009).

As the EIA procedure has to be completed before the Decision-in-Principle can be issued, most of the safety-relevant questions cannot be adequately answered within the EIA process. Whether the reactor will comply with the requirements discussed within the EIA process, can only be answered in the following approval procedure. Therefore, the **final statement of the MEE** should require the applicant to provide relevant information after the EIA procedure, especially on topics which came up during the EIA procedure but couldn't be answered at this stage.

It would be appreciated if information requested in the expert statement at hand could be provided once available.

4 DESCRIPTION OF THE PROJECT

Background of the project

In 2008, Fennovoima implemented an environmental impact assessment (EIA) to assess the impact from the construction and operation of a nuclear power plant of approximately 1,500–2,500 megawatts that consists of one or two reactors at three alternative locations: Pyhäjoki, Ruotsinpyhtää and Simo.

Fennovoima received the Decision-in-Principle (DiP) in compliance with the Nuclear Energy Act (990/1987) in May 2010 – the DiP was confirmed by the parliament in July 2010. According to the Decision-in-Principle 2010, Hanhikivi in Pyhäjoki and Karsikko in Simo are suitable locations for a nuclear power plant. The Hanhikivi headland in Pyhäjoki was selected as the plant site in autumn 2011.

At the same time, Fennovoima also started to assess whether a mid-sized unit of 1,000–1,300 MW_e would be a better option. Fennovoima invited Rosatom to engage in direct negotiations, in parallel with Toshiba, concerning its AES-2006. In July 2013, Fennovoima announced that it would focus on negotiations with Rosatom and end consideration of the Toshiba option. It signed a project development agreement with Rusatom Overseas, which may also take a 34% share of the project (WNA 2013a).

The nuclear power plant of approximately 1,200 MW with a company of the Russian Rosatom Group as its supplier, which is currently the object of the environmental impact assessment, was not mentioned in the original application for a Decision-in-Principle as one of the plant alternatives. Therefore, the Ministry of Employment and the Economy required Fennovoima (MEE) to update the project's environmental impact assessments.

4.1 Treatment in the EIA-Report

The organization responsible for the project is Fennovoima, a Finnish nuclear power company established in 2007. Fennovoima´s owner is Voimaosakeyhtiö SF, a company currently consisting of 46 industrial, commercial and energy companies (EIA-REPORT 2014, p. 29). The shareholders represent a variety of sectors. Negotiations regarding Rosatom Company becoming Fennovoima's minority shareholder are currently ongoing. Agreements will be made to ensure that the majority of Fennovoima remains in the ownership of Voimaosakeyhtiö SF (EIA-REPORT 2014, p. 30).

The EIA-REPORT (2014, p. 30) emphasized that Rosatom is one of the leading nuclear technology experts in the world. It is mentioned that Rosatom's impressive nuclear technology competence would be at Fennovoima's disposal during the different stages of the project.

The Fennovoima nuclear power plant will be constructed on Hanhikivi headland in Pyhäjoki. The municipality of Pyhäjoki is located on the coast of the Gulf of Bothnia in between the municipalities of Raahe and Kalajoki, in the southwestern part of the province of Northern Ostrobothnia.

Table 1-1 of the EIA-REPORT (2014, p. 31) compares the key characteristics of the approximately 1,200 MW plant studied in this EIA-Report to those of the 1,800 MW plant studied in the EIA of 2008 (see Table 2). The data of the plant of approximately 1,200 MW will be further specified as the design efforts proceed.

Table 2: Specification of the 1,200 MW_e nuclear power plant in comparison with 1,800 MW_e nuclear power plant.

Description	NPP with approximately 1200 MW_e	NPP with approximately 1800 MW_e
Reactor	Pressurized water reactor	Pressurized water reactor
Electric power	about 1,200 MWe	about 1,800 MW_e
Thermal power	about 3,200 MW_{th}	about 4,900 MW_{th}
Efficiency	about 37%	about 37%
Fuel	Uranium dioxide UO_2	Uranium dioxide UO_2
Thermal power released to the water system	about 2,000 MW_{th}	about 3,100 MW_{th}
Annual energy production	about 9 TWh	about 14 TWh
Cooling water requirement	about 40–45 m³/s	about 65 m³/s
Fuel consumption	20–30 t/year	30–50 t/year

The construction of the nuclear power plant has been estimated to take about six years. The necessary infrastructure elements on land and water as well as the necessary excavation works must be completed before the construction can start. The civil engineering work is planned to commence in 2015. The construction of the power plant will take 5–6 years; the commissioning of the plant will take 1–2 years. The nuclear power plant is planned to be in operation in 2024 as agreed in the plant supply contract signed in December 2013.

To start construction of the nuclear power plant, Fennovoima will need a construction license in compliance with the Nuclear Energy Act from the Government. Before being able to start production at the nuclear power plant, Fennovoima has to apply for an operating license according to the Nuclear Energy Act, an environmental permit, and other permits required for the power plant. (EIA-REPORT 2014, p. 33)

4.2 Discussion

Organization responsible for the project

In 2006, a group of Finnish industrial and energy companies interested in participating in a new power plant project founded Voimaosakeyhtiö SF as the main shareholder in Fennovoima. Within a couple of years, over 60 companies had joined and Germany's EON had taken a 34% stake in the project. However, in October 2012, EON withdrew from Fennovoima, with Voimaosakeyhtiö buying its share to take full control. (WNN 2014b)

Meanwhile, Russian nuclear state corporation Rosatom has bought 34% of Fennovoima for an undisclosed amount. The deal was made between RAOS Voima Oy, Rosatom's Finnish subsidiary, and Voimaosakeyhtiö SF. (NNF 2014)

The EIA-Report emphasizes that Rosatom is one of the leading nuclear technology experts in the world. In this context, the statement of Jukka Laaksonen, Vice President of Rusatom Overseas and former Director of the STUK, is remarkable. In December 2012, he emphasized "[g]etting the role as world leader in Nuclear Technology requires that Russia is also world leader in the Codes and Standards that are used to design the Russian nuclear facilities and to manufacture components installed in these facilities."

Laaksonen explained that the "[c]omparisons between the Russian Nuclear Safety Regulations and the Technical Standards used in other countries have been made in connection with licensing of Russian designed NPPs (Finland) and in specific multinational projects (EU, MDEP). These comparisons have indicated that Russian *"rules"* are in some respects more comprehensive and stringent than the foreign Technical Standards, but there are also gaps in the Russian *"rules"* (i.e. some important topics are not adequately addressed or not addressed at all)." (ROSATOM 2012)

The EIA program mentioned that under Voimaosakeyhtiö SF, 60 industrial and commercial enterprises, as well as energy companies, are grouped (UMWELTBUNDESAMT 2013). However, several companies left the consortium, the EIA-Report mentions only 46 shareholders.

At the deadline for committing to the nuclear power plant (February 28, 2014), 44 of the shareholders had confirmed to take a combined 50.2% stake in the company. Voimaosakeyhtiö announced that it aimed to increase the share held by Finnish firms up to around 66% and that it was in negotiations with potential new owners and that these negotiations would be finalized by the end of June. At that time, the final ownership shares of the current owners would be specified (WNN 2014b).

On March 27, 2014, however, the Finnish retailer Kesko announced it would drop out of the Fennovoima nuclear consortium, adding to concerns over a project that looks to be heading for a showdown in parliament between members of the ruling coalition. Kesko owned around 2% of Fennovoima shares. With Kesko's exit, Finnish ownership in Fennovoima sank below 50%, which might lead to problems as the state requires a clear majority in Finnish ownership in the project. (REUTERS 2014)

The government announced it expected to vote on the Rusatom-Fennovoima project in June and that parliament would vote on the issue in fall. (NNF 2014)

Project schedule

The construction time for the nuclear power plant is estimated to be about six years; start of operation is envisaged in 2024. The reference plant for this design (VVER-1200/V491) is the nuclear power plant Leningrad-II in Russia. The construction licenses for the two units of Leningrad-II were granted in June 2008 and July 2009, respectively; construction started soon afterwards. Startup was originally planned for 2013 and 2014. Today, the target years for begin of operation are 2016 and 2018. There is no published information on the reasons for this delay (see chapter 5.2.3 of this expert statement).

However, it has to be assumed that similar delays will occur during the construction of the new nuclear power plant Hanhikivi 1, particularly because of the ambitious project schedule.

4.3 Conclusions/Recommendations

In the EIA program, it is mentioned that Fennovoima is owned by Voimaosakeyhtiö SF, under which a total of 60 enterprises as well as energy companies are grouped. During the last years, however, several companies left the consortium. The EIA-Report mentions 46 shareholders.

At the deadline for committing to the nuclear power plant (February 28, 2014), 44 of the shareholders had confirmed to take a combined 50.2% stake in the company. RAOS Voima Oy, the Finnish subsidiary of Russian nuclear state corporation Rosatom, bought 34% of Fennovoima. Voimaosakeyhtiö said that it aimed to increase the share held by Finnish firms up to around 66%. The company said that it was in negotiations with potential new owners and that these negotiations would be finalized by the end of June. At that time, the final ownership shares of the current owners would be specified.

At the end of March 2014, the Finnish ownership in Fennovoima sank below 50%. It is important to note that the state requires a clear majority of Finnish ownership in the project.

The government announced it expected to vote on the Rusatom-Fennovoima project in June 2014 and that parliament would vote on the issue in fall.

In 2024, start of operation of the nuclear power plant Hanhikivi 1 is envisaged. Reference plant for the design of the reactor type of Hanhikivi 1 (VVER-1200/V491) is the Russian nuclear power plant Leningrad-II. The construction licenses for the two units of Leningrad-II were granted in June 2008 and July 2009, respectively; construction started soon afterwards. Commissioning of the units was originally planned for 2013/2014, but is now postponed until 2016 and 2018. It has to be assumed that delays will also occur during the construction of Hanhikivi 1, particularly because of the ambitious project schedule (in particular a construction time of only six years).

5 REACTOR TYPE

5.1 Treatment in the EIA-Report

It is stated that the reactor type chosen for the planned NPP New Build at Hanhikivi is Rosatom's AES-2006/V491 (VVER-1200/V491), which is a third generation reactor based on VVER technology (EIA-Report 2014, p. 54). The current state regarding planned or on-going construction of new AES-2006 reactors worldwide is illustrated. The construction of AES-2006 in Leningrad, Kaliningrad and Novovoronezh is mentioned (EIA-REPORT 2014, p. 54).

In the response to the Austrian question provided in appendix 2 of the EIA-REPORT (2014, p. 3), it is stated that Fennovoima chose AES-2006/V491 (versus the AES-2006/V392M[3]) as the reference plant because its defense-in-depth approach in relation to redundancy and independency between system trains is closer to the Finnish regulatory requirements.

With regard to the safety design of the reactor, it is stated that the **target of the design** of the selected reactor type (AES-2006) is to comply with the requirements of IAEA's safety guidelines and standards, European Utility Requirements (EUR) and Russia's national regulations and requirements. The Fennovoima plant will also be designed to fulfill the requirements of the Finnish authorities (EIA-REPORT 2013, p. 54–55). Furthermore, appendix 2 of the EIA-REPORT (2014, p. 1) mentions that the safety design of the AES-2006 reactor also takes advantage of WENRA regulations, and that the WENRA safety objectives and lessons learned from Fukushima have been implemented in the latest revision of STUK YVL Guides. Most of the new YVL Guides entered into force in December 2013 (EIA-REPORT 2014, p. 85).

According to the EIA-REPORT (2014, p. 55), the **safety design** of AES-2006/V491 is based on both active and passive systems. The main active systems have 4x100% redundancy which means that each one of the four system trains is sufficient to fulfill the safety function (EIA-Report 2014, App. 2, p. 2). The safety systems for reactor cooling are installed in four separate divisions within the safeguard building (EIA-REPORT 2014, p. 55). Alternatively, the reactor can be cooled down using passive systems by extracting heat from the steam generators to pools located outside the containment building (EIA-REPORT 2014, p. 55). The passive heat removal systems have a capacity of 4 x 33%, which means that the systems are able to perform their function even if one train would fail (Eia-REPORT 2014, App. 2, p. 3).

According to the EIA-REPORT (2014, p. 55), the reactor has 121 **control rods** arranged in 12 rod banks. It is mentioned that the number of the control rods is higher compared to other PWRs (EIA-REPORT 2014, p. 55). It is possible to shut down the reactor with or without using the control rods, and the reactor power can also be controlled by injecting boric acid into the reactor (EIA-REPORT 2014, p. 55).

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[3] The reactor type AES-2006 has two variants, which are the AES-2006/V491 and the AES-2006/V392M. The AES-2006/V491 is designed by JSC SPb Atomenergoproekt (based in Saint-Petersburg), and the AES-2006/V392M is developed by JSC Atomenergoproekt (based in Moscow). Significant differences between these two variants include differences in systems to cope with BDBA, differences in predicted CDF, differences in the use of passive and active systems, etc. (IAEA-ARIS 2011a, p. 2)

The **reactor containment** is also briefly described. It is stated that the reactor type features a double-shell containment building (EIA-REPORT 2014, p. 55). The inner containment shell is made of pre-stressed reinforced concrete that is capable of withstanding the tensile stresses caused by overpressure under accident conditions. The outer containment shell is a thicker structure made of reinforced concrete that is capable of withstanding external collision loads, including a large commercial airplane crash. Regarding the airplane crash, it was also mentioned that both the collision force caused by the airplane itself and the eventual fire caused by its fuel will be taken into account in the design of the buildings that are important to safety (EIA-REPORT 2014, p. 88).

The presence of a **core catcher**, as a provision to cope with a severe accident in the case the core melts through the RPV, is mentioned. The location and function of the core catcher are described (EIA-REPORT 2014, p. 55). In the core catcher, the corium is cooled down by water spraying from above. Spraying the corium with water also reduces the dispersion of radioactive substances inside the containment building. The water vapor generated in the core catcher is cooled down using the reactor building's passive cooling system. It is stated that this allows for maintaining the integrity of the containment building even during severe accidents and, consequently, limiting the dispersion of radioactive releases outside the containment building (EIA-REPORT 2014, p. 55). It was also mentioned (EIA-REPORT 2014, App. 2, p. 3) that no human actions are required during the operation of the core catcher, and that according to the severe accident management strategy, the core catcher is held dry before RPV rupture. Additionally, the containment structures and the internal geometry of the reactor pit also contribute to minimizing the risks of steam explosion. It was stated that **further analysis** of the core catcher will be performed in the **construction license application stage** of the project (EIA-REPORT 2014, App. 2, p. 3).

It is stated that plan safety design, structural design and basic dimensioning will be described in the **Preliminary Safety Analysis Report (PSAR)** together with related safety analyses, which will be reviewed by STUK as a part of the construction license procedure (EIA-REPORT 2014, App. 2, p. 3-4). Before the power plant is commissioned, Fennovoima and the power plant supplier will draw up the **Final Safety Analysis Report (FSAR)** with safety analyses performed based on the detail design of the plant (EIA-REPORT 2014, App. 2, p. 4).

5.2 Discussion

5.2.1 WENRA Safety Objectives

Introduction

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 (WENRA 2010). The **safety objectives** were based on a report by the Reactor Harmonization Working Group of WENRA (WENRA-RHWG 2009), also considering comments received from stakeholders. They are fully in line with the Fundamental Safety Principles of the International Atomic Energy Agency (IAEA 2006).

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 (WENRA 2013).

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 **EIA-Report does not discuss the fulfillment of the WENRA safety objectives**. They are only briefly referred to, as being covered by the current Finnish regulations. There is no discussion of the fulfillment of the individual objectives concerning the Hanhikivi project.

A recent presentation on international safety standards by Rosatom (LAAKSONEN 2013, p. 31), claims that "[n]ew VVER-1200 plants have been designed to meet the most advanced safety principles and requirements that are available today". The WENRA safety objectives are only briefly mentioned and their implementation in current Russian standards is not discussed. Such a discussion could not be found by the authors in the open literature.

In this section, available information on the VVER-1200/V491 is compiled and evaluated, applying the WENRA safety objectives. The following points will be assessed:

- What can be asserted on the basis of available information, regarding the fulfillment of WENRA safety objectives by the VVER-1200/V491?
- Which issues remain unclear on the basis of available information regarding the fulfillment of the WENRA safety objectives by this reactor type?
- Are there any potential challenges which could make fulfillment of the WENRA safety objectives difficult or impossible?

Apart from the WENRA safety objectives, the above-mentioned reports by RHWG will also be taken into account in these considerations.

A comprehensive and detailed assessment of these points is beyond the scope of this expert statement. Instead, in the following **illustrative examples are provided** which are sufficient to give a rough overall picture.

WSO 1 – Normal operation, abnormal events (DiD levels 1, 2)

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.

Means to achieve objectives	Situation at VVER-1200/V491
Large operational margins	Among the basic principles and approaches of the design, the following items are mentioned (SPbAEP 2011, p. 3): Improving system and equipment characteristics by abandoning excessive conservatism and optimizing design margins; reducing capital and operating expenditures (The latter point does not refer directly to margins; however, it provides an interesting background to the first point.)
Improved I&C	Application of enhanced reliability I&C with self-diagnostic functions (IAEA-ARIS 2011a, p. 2)
Improved reactivity control	The number of control rods has been increased compared to VVER-1000; hence, re-criticality temperature is very low (about 100 °C). Due to the use of burnable absorbers, the boron content of the primary coolant can be kept relatively low. Management of boron-free water slug has been taken into account in design, in spite of lower boron content of the primary coolant and the more effective scram. Supplementary analyses and/or tests in later stages of the licensing process are required (STUK 2009a, p. 51).
Improved testing and monitoring	Application of diagnostics systems for safety-related system equipment for periodic tests during shutdown as well as for the monitoring of the operating reactor (IAEA-ARIS 2011a , p. 2).
Advanced materials and manufacture	Reactor pressure vessel (RPV) manufactured of forged shells without longitudinal welds. Number of welds is minimized to reduce the time taken by inspections (IAEA-ARIS 2011a, p. 2). The interior surface of the RPV is covered with austenite welding, protecting the main metal from corrosion influence of coolant and also providing the possibility of decontamination of the vessel's interior. (MINENERGO 2010a, p. 84) The welding of the RPV is performed using known and qualified methods (STUK 2009a, p. 47). Radiation embrittlement of RPV has been taken into consideration and is monitored during operation. Attention must be given to the analysis requirements for P, Cu and Ni to make sure embrittlement remains within limits during 60 years of service life (STUK 2009a, p. 47). Damages detected earlier in the welding joints of VVER-1000 plant's SG collectors have been taken into account through material selection of the new SG type of AES-2006 (STUK 2009a, p. 48).
High reliability of operational systems	The residual heat removal system and its subordinate cooling systems have 4 x 100% redundancy. The spent fuel pool cooling system has two trains; each train is equipped with two pumps. As a diverse system, the containment spray system pumps and ECCS heat exchangers can be used. (STUK 2009a, p. 56)

Discussion

It seems plausible that considerable **efforts** have been undertaken **to improve the design of the VVER-1200**, as compared to the forerunner types. This concerns operational systems, reactivity control, I&C, materials and other fields.

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. For example, “abandoning excessive conservatism and optimizing design margins” seems to point into the direction of cost-effective operation. However, the Reactor Harmonization Working Group of WENRA listed as an example for an improvement in the context of safety objective 1: “Larger operational margins based on design provisions in order to reduce the frequency of abnormal events.” (WENRA-RHWG 2009, p. 26)

Another **challenge** could lie in 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.

While efforts clearly have been undertaken to fulfill this WENRA safety objective, available information does not nearly permit a definite assessment. There are potential challenges (most notably the conflict between economic optimization and safety improvements) which particularly need to be followed up.

WSO 2 – Accidents without core melt (DiD levels 3a, 3b)

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 nor 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.

Means to achieve objectives	Situation at VVER-1200/V491
General design principles	Functional and design diversity, protection from common-cause failures, provisions against operator error, physical separation and assurance of high reliability of safety functions (IAEA-ARIS 2011a, p. 14). Severe accidents cannot be initiated by simple imposition of several single and additional failures. Such accidents can emerge only in case of several common-cause failures in safety system trains (SPBAEP 2011, p. 21). Safety system actions are calculated with consideration of the following failures for each safety function (SPBAEP 2011, p. 21): • system train failure because of the initiating event if such initiating event is possible for this safety system, and also as a dependent failure;

	• system train failure because of the worst single failure of one of the active elements of this train or passive elements that have moveable mechanical parts; • maintenance or service of one of the safety system trains when such maintenance of service is envisaged in this safety system design. Furthermore, according to the deterministic approach, the following is considered for each calculated initiating event (SPBAEP 2011, p. 21): • one human error independent from the initiating event; • further failures of systems which can cause violation of limits and affect the accident propagation; • selection of initial and boundary conditions that affects the results adversely.
Systematic consideration of initiating events, including hazards	In addition to design basis conditions, the following beyond design basis accidents (DECs) are considered in the project (BYKOV 2013a, p. 15): • loss of all AC power sources for 8 and 24 hours • loss of spent fuel pool cooling for 8 and 24 hours • spectrum of steam line breaks inside and outside containment, up to steam line of maximum diameter with rupture of one SG tube • complete loss of feedwater • LB LOCA with failure of low pressure ECCS • SB LOCA with failure of high pressure ECCS • long-term (up to 24 hours) loss of heat removal with reactor top head removed or with reactor sealed • multiple failure of SG tubes, or leaks over steam generator primary side collector • ATWS Protection against internal events: 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, call into question the adequate realization of physical separation. (STUK 2009a, p. 59) Lower floor of safety building contains seawater heat exchangers and pipelines of intermediate circuit cooling system. Controlling a major flood caused by breakage of these components is challenging. (STUK 2009a, p. 59) 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. (STUK 2009a, p. 59) Finnish safety requirements concerning protection from internal hazards, such as floods and fires, have not yet been demonstrated. (STUK 2009a, p. 59) Break preclusion principle is applied to primary circuit piping, part of which involves LBB principle. Nevertheless, break of pipe with largest diameter is taken into account in ECCS design. Clarifications were still needed regarding dynamic effects of pipe break and their effects on the reactor's inner components (STUK 2009a, p. 48). The building for the emergency diesel generators is divided into four parts by reinforced concrete walls, completely separating the four safety trains of the emergency power supply (SPBAEP 2011, p. 14).

	Regarding earthquake, the reference plant Leningrad-2 is designed against a peak ground acceleration of 0.25 g (SPbAEP 2011, p. 11). Natural hazards are site specific and are not discussed further in this section (see section 6 of this expert statement).
Systematic consideration of multiple failures	Protection from common-cause failures is mentioned as a design principle for the AES-2006. Furthermore, it becomes clear from the information relating to various systems that there is a certain degree of diversity in several important cases (for example, for heat removal from the containment and for emergency power supply). (Stuk 2009a; Iaea-Aris 2011a; Belarus 2013) It is also pointed out that each safety system consists of active and passive (or "practically passive") parts, each of which is able to carry out the safety function (Minenergo 2010a, p. 108). A listing of some examples for common cause failure events which are postulated is provided in a recent presentation (Kolchinsky 2013c, p. 14), including: • ATWS • station blackout • total failure of all computer I&C systems • CCF at LOCA However, no systematic discussion and consideration of multiple failures (level of DiD 3b, according to WENRA) could be found in the documents used for this expert statement. 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 sufficient to guarantee the safety function.
Safety systems	Systems for LOCA: ECCS with active and passive part – high-head and low head safety injection (4 x 100%), 4 accumulator tanks (4 x 50%). Leaking water drains back into IRWST. Design and testing of suction strainers was under way at the time of the STUK assessment. (Stuk 2009a, p. 54) Systems for accident with primary circuit intact: Removal of heat via emergency feedwater system (4 x 100%) and turbine condenser. (Stuk 2009a, p. 52) If heat removal via sec. circuit is not possible, primary feed and bleed can be performed (high-head safety injection 4 x 100%, removing hot water via safety valves back to IRWST). (Stuk 2009a, p. 53) In mid-loop operation, if the residual heat removal system (4 x 100%) cannot be used, the passive heat removal system from the containment can be used, transferring heat to pools on the containment roof for 24 hrs without measures by operator (can be lengthened to 72 hrs by pumping water from storage in the pools). (Stuk 2009a, p. 52) Details of the arrangement for filling the pools are not clear, for instance, whether it are fixed pipelines or whether it is a fixed refilling pump. For heat removal from the containment in transient and accident conditions, there is diversity: PHRS for containment (4 x 33%) and containment spray system (4 x 50%). (Stuk 2009a, p. 54-55) In case of loss of ultimate heat sink, heat can be removed via EFWS and BRU-A valves (releasing secondary steam in the atmosphere). EFWS can bring reactor in hot shutdown and maintain there for 24 hrs, lengthened to 72 hrs by pumping water in pools. Alternatively, passive heat removal system via steam generators (PHRS SG) can be used (4 x 33%). This system requires the

	opening of a valve and subsequently works without power source. Can maintain hot shutdown for 24 / 72 hrs, as stated above. (STUK 2009a, p. 55) Further systems mentioned are: Make-up and boron control system, emergency gas removal system (2 x 100%), overpressure protection systems (primary and secondary, 2 x 100%). The emergency gas removal system is destined for DEC and intended for removal of steam-gas mixture from the primary circuit pipelines, the top of the reactor vessel, the SG collectors and the pressurizer (BYKOV 2013a). 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. There is an emergency control room which can be used for control of safety-critical systems independently of the main control room. The plant can be brought into a safe state (cold shutdown) from the ECR. It is located in a separate building from the main control room. In case of failure of the SFP cooling system, cooling can be implemented by water evaporation and the PHRS for the containment. The fuel pool purification system can provide make-up water to the pools. (STUK 2009a; IAEA-ARIS 2011, BELARUS 2013; SPBAEP 2011)
Categorization of accidents	Accidents without core melt are divided into 3 categories: ● DBC-3 (Category 3 - DBA), ● DBC-4 (Category 4 – DBA), and ● DEC-A (common cause failures, extreme external impacts). There are in total 6 categories of design conditions. In previous designs, there were only three categories: normal operation, anticipated transients, and accidents. (KOLCHINSKY 2013a, p. 15) Acceptance criteria published for different accident categories belonging to the design basis appear to be roughly in line with general international practice, for example (BYKOV 2013b, p. 12): ● maximum cladding temperature does not exceed 1200 °C ● local cladding oxidation depth does not exceed 17% of the original cladding thickness ● amount of hydrogen obtained in the fuel cladding-coolant interaction should not exceed 1% of the maximum possible amount.
Limits for emergency protection	In case of DBA, the radius of the emergency protection area does not exceed 0.8 km around the reactor. This corresponds to the boundary of the site (KOLCHINSKY 2013a, p. 16). The effective dose for DBC-3 has to be below 1 mSv/event, and 5 mSv/event for DBC-4 (KOLCHINSKY 2013b, p. 4).
Requirements and results for CDF	Limit for CDF: 1.0E-6/yr. Calculated CDF for one-year fuel cycle (mean values): ● Full power operation 2.24E-7 (IAEA-ARIS 2011a, p. 13), 3.82E-7 (BELARUS 2013, p. 26), 1.36E-7 (KOLCHINSKY 2013b, p. 8) ● Low power and shutdown 3.7E-7 (the first two sources mentioned above), 4.58E-7 (KOLCHINSKY 2013b) ● CDF for all states: 5.94E-7/yr or 7.52E-7, respectively Sources provide no information to which extent internal and external hazards are included in these numbers, and no information on uncertainty of results.

	Main contributors to CDF in power operation: Intermediate circuit system for essential services, makeup systems for the first circuit, emergency heat removal systems, systems involved in the feed and bleed procedure. (BELARUS 2013, p. 26) Main contributors to CDF in low power and shutdown state: Personnel errors in the responses to initiating events, system for residual heat removal from the first circuit, low-pressure ECCS. (BELARUS 2013, p. 27)
Reduction of influence of human factor	Reduction of human factor influence is included in the main principles and approaches for the design of AES-2006 (KOLCHINSKY 2013a, p. 5). No further explanation is provided by the source. For design basis accidents, it is required that actions of automatic and/or passive systems provide a grace period of at least 8 hours (BYKOV 2013a, p. 2).

Discussion

Regarding the **general design principles**, it is not clear to which extent they go beyond those which are customary for the latest Gen II plants, like assuming additional single failure (including human errors) plus maintenance of one train in case of DBAs or selecting the most adverse initial and boundary conditions.

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

The listing of DEC events in a recent presentation does not seem to be complete and can only be taken as a compilation of examples. Events like uncontrolled boron dilution, rupture of a major pressure-retaining component, uncontrolled level drop during mid-loop operation or total loss of component cooling system are not included.

Controlling internal hazards could be a challenge as far as the safety building is concerned; in particular, separation of safety trains as well as equipment belonging to one train seems to be a challenge.

From the documents available, it is not clear to which extent a systematic consideration of **multiple failures** has been performed. Wherever the levels of defense-in-depth are addressed, there is no mention in earlier documents of the two sub-levels of level 3 (and in particular, of level 3b covering multiple failures), which have been introduced by WENRA-RHWG in 2009. However, the sub-levels are applied in a presentation in late 2013 in the framework of INPRO. The importance of systematic and comprehensive considerations of multiple failures appears to have been recognized now, but could still be a challenge.

Safety systems generally have high redundancy (4 x 100% in many cases, compared to 3 x 100% or 4 x 50% mostly found in Gen II plants). AC emergency power supply is diverse; however, one of the diverse systems has no redundancy.

The published results of **PSA studies**[4] appear to confirm that the limit of 1E-6/yr 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 number 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.

There are some differences between CDF values as recently reported. The differences are small compared to what can be expected as the range of uncertainty, and therefore of minor importance.

While efforts clearly have been undertaken to **fulfill the WENRA safety objectives**, available information does not come near to permitting a definite assessment.

There are potential significant **challenges which need to be followed up**; in particular:

- Conservative selection of assumptions for dealing with design basis accidents, beyond what is customary for Gen II plants;
- 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.

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.

[4] In this context, it has to be taken into account that there are important factors which cannot, or cannot adequately, be taken into account in PSA studies (for example safety culture, malicious human acts, aging phenomena), and that not all uncertainties with which PSA results are inevitably beset can be quantified. However, a discussion of this issue would be beyond the scope of this expert statement.

Means to achieve objectives	Situation at VVER-1200/V491
Safety features	Safety features for BDBA generally are based on passive elements, in particular (SPbAEP 2011, p. 22; KOLCHINSKY 2013a): ● hydrogen recombiners ● core catcher ● passive heat removal system from the containment (C-PHRS) ● passive heat removal system via steam generators (PHRS-SG) ● spray system for pressure and temperature decrease and suppression of volatile iodine The pressure operated relief valves of the primary circuit can work both in active and passive mode of operation. There is only one set of valves for DiD levels 3 and 4 (DBAs and BDBAs). (IAEA-ARIS 2011a; STUK 2009a) The passive heat removal systems for containment and via steam generator are passive systems which have not been used previously at NPPs (STUK 2009a, p. 46). The two systems are linked; they share the emergency heat removal tanks (BYKOV 2013a). As already mentioned in the previous section, these systems can remove the decay heat for 24 hrs without measures by operator. (This period of time can be lengthened to 72 hrs by pumping water from storage in the pools.) This also applies in case of a severe accident, when the passive heat removal system for the containment is cooling the core catcher (BELARUS 2013). No filtered venting system is foreseen. According to the supplier, no significant amount of non-condensible gases other than hydrogen is expected (STUK 2009a, p. 50). There is diverse AC emergency power supply for DiD levels 3 and 4; one of the diverse systems has no redundancy (see above). The SAM systems have dedicated DC power supply (batteries, with capacity for 72 hrs). Procedures for re-charging are not clear (STUK 2009a, p. 57). The core catcher is the central feature for controlling severe accidents and is therefore discussed in more detail separately in the following row of the table.
Core catcher	The core catcher of the VVER-1200 has been developed from the core catcher for the VVER-1000. The design is largely the same for the VVER-1200/V491 and the VVER-1200/V392M. It is a crucible-type catcher – in contrast to the other type which has been fully developed before, the catcher with melt spreading developed for the EPR. In the crucible-type core catcher, the melt is immobilized within a water-cooled steel vessel located directly below the reactor pressure vessel. The core catcher vessel has vertical walls and a conical bottom. The internal volume of the catcher is partially filled with sacrificial material, containing oxidic components and steel. Steel sheeting on the vessel top prevents water ingress before melt relocation; this considerably reduces the probability of steam explosions. The catcher vessel capacity is sufficient to accommodate the whole mass of the core melt. Heat is transferred from the molten pool through the vessel walls and bottom to the surrounding cooling water. The heat from the upper surface of the melt is at first removed by radiation. Later, the melt is flooded from above, water covering the upper surface.

	The purpose of the sacrificial material is to reduce the temperature of the melt, diluting the melt to reduce the density of heat release and increase the heat exchange surface of the melt, and to oxidize the melt. By oxidizing zirconium with, for example, Fe_2O_3, the amount of zirconium which could react with steam is significantly reduced. Some hydrogen is produced by other reactions (mainly by iron oxidation in interaction with steam), but all in all, considerably less hydrogen is expected to be produced than in case of a core melt accident without catcher. Stratification is expected in the catcher vessel; at first with a metal layer on top, and an oxide layer below. Later, inversion is expected to occur. This inversion has to be guaranteed for better distribution of the heat flux, the avoidance of steam explosions and restriction of hydrogen generation from steam-metal reactions after top flooding. Timing of the top flooding is important. Melt relocation from the RPV into the catcher is expected to occur in several stages. Top flooding must be excluded before relocation is complete, to avoid steam explosions. Compared to the spreading-type core catcher, complete melt immobilization takes a long time in the crucible-type catcher. Analysis of the efficiency of the VVER-1200 core catcher was performed analytically, on the basis of experimental and analytical results for the VVER-1000 core catcher. Important experiments for the VVER-1000 have been performed at the RASPLAV facility, with production of a simulated melt by induction and with melt masses in the crucible of up to 10 kg. The numerical analysis of the core catcher efficiency for the VVER-1200 was stated to confirm that the core catcher effectively executes the functions of reception, localization and long-term cooling of the corium. (KHABENSKY 2009; ZVONAREV 2011)
Limits for emergency protection	In case of severe accidents, the radius of area where protection measures for the population are planned doesn't exceed 3 km (KOLCHINSKY 2013a, p. 16).
Requirements and results for LRF	Limit for large radioactive release: 1.0E-7/yr. The calculated LRF (mean value) is 1.8E-8/yr (ERMOLAEV 2009, p. 40). This value, however, includes full-power operation and internal initiating events only. There is no information concerning its uncertainty (UMWELTBUNDESAMT 2012).
Practically eliminated phenomena	Physical phenomena related to severe accidents that might endanger the containment integrity are avoided as per the NPP design, namely (SPBAEP 2011, p. 27): ● 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. The term "practical elimination" is not used in this source; neither is it referred to in the other documents the authors have evaluated. However, 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.

Discussion

The VVER-1200/V491 exhibits most of the **safety features** which are considered as necessary and are usually found at NPPs of Generation III, apart from a filtered venting system. There are passive systems for heat removal which have not been used in NPP designs before. Demonstration of their functioning and reliability in severe accident conditions could present a challenge.

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.

A unique type of **core catcher** has been developed for the VVER-1200/V491 – the crucible type catcher.

This type has first been developed for the VVER-1000, and has already been installed at the Tianwan and Kudankulam plants. The results of numerical and experimental investigations for these plants could be used for the analysis of core catcher efficiency for the VVER-1200. However, there are changes in energy release and melt mass, and dynamics of melt flow into the crucible. Therefore, additional analyses have been performed for the core catcher of the VVER-1200.

Published results so far confirm the efficiency of the core catcher. However, there are a number of challenges which could render comprehensive and authoritative demonstration, of a standard as required in a licensing procedure in an EU country, rather difficult and complex:

- Reliable and accurate description of different accident progression scenarios, which can be characterized by different melt temperatures, compositions (in particular, different degree of oxidation of the melt in the RPV) and scenarios of melt relocation in the catcher – generally crucial for functioning of core catcher
- Timing of arrival of melt for different scenarios, making sure that flooding from top occurs after complete relocation – crucial for steam explosions
- Guaranteeing inversion of metallic and oxidic layers in the catcher vessel – crucial for steam explosions, heat flux and hydrogen formation

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.

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 also can 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[5].

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. Also, there is no systematic discussion on how it has been achieved for the phenomena mentioned.

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 (IAEA 2012, p. 6).

In a 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 (WENRA 2013, p. 29-32).

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.

It could be a challenge to demonstrate practical elimination for the VVER-1200/V491 for all phenomena in question, taking into account these principles.

WSO 4 – Independence of levels of DiD (DiD levels 1 – 4)

Objectives:

- Enhancing the effectiveness of the independence between all levels of defence-in-depth, in particular through diversity provisions (in addition to the strengthening of each of these levels separately as addressed in the previous three objectives), to provide as far as reasonably achievable an overall reinforcement of defence-in-depth.

[5] See footnote on PSA in the previous section.

Means to achieve objectives	Situation at VVER-1200/V491
General approach to levels of DiD	The defense-in-depth concept is regularly presented in the documents on the VVER-1200/V491. However, this is usually done in a brief and general manner (IAEA-ARIS 2011a; MINENERGO 2010a; SPBAEP 2011; BYKOV 2013b). There is no detailed discussion of the application of this concept. The introduction of two sub-levels in level 3 of DiD (3a – single initiating events, 3b – multiple failures) by WENRA (see above) has been registered by the designer. It is stated that at each of these sub-levels and at level 4, a special set of SSC for the implementation of all safety functions is provided by the design, including I&C and electrical sources (KOLCHINSKY 2013c, p. 10). However, a table displaying a matrix of levels of DiD and systems used which is contained in the same presentation shows that this principle is not rigorously implemented. For example, systems for heat removal are used both in sub-levels 3a and 3b, and partly also in level 4.
Safety systems and features used on several levels	From the available information, it is clear that there is only one primary depressurization system (one set of safety valves) for levels of defense 3 and 4. There is no independent system for severe accidents. Furthermore, it appears that the passive heat removal systems via steam generators and for the containment are destined to be used both at safety level 3b (corresponding to DEC A) and safety level 4. (Based on sections on WSO 2 and WSO 3.) The I&C systems are of particular interest in this context and are discussed separately below.
I&C structure	According to the safety requirements, the I&C systems are physically and functionally divided into safety system I&C and normal operation condition I&C. I&C has a hierarchical structure and is divided into several sub-systems (SPBAEP 2011, p. 32-34). In the assessment by STUK (2009a, p. 60), it is stated that the I&C systems of the AES-2006 are based on the principle of defense-in-depth, with several lines of defense: • 1st line – normal process automation and control systems. • 2nd line – primary protection system with two redundant, diverse parts (four redundant sub-systems in all). • 3rd line – second protection system, hardwired, initiating the most important safety functions (also four redundant sub-systems). This system is “realising diversity principle for the automation”. • 4th line – severe accident management system. The documents available for assessment by STUK did not describe to which operational state the hardwired system can bring the plant, if the other systems fail. Furthermore, the separation of I&C systems and components of different safety classes from each other and within sub-systems has not been described in the documents available for STUK; neither has the separation of I&C and monitoring systems for severe accident management from other automation systems. Therefore, STUK concluded that the consistency with Finnish requirements is not clear. (STUK 2009a, p. 61) Also, it is not clear if the diversity principle for measurements (requiring that at least two different process parameters must be measured in the reactor protection system) is realized. (STUK 2009a, p. 61)

Means to achieve objectives	Situation at VVER-1200/V491
	No clear allocation of the lines and sub-systems of I&C to the levels of defense-in-depth can be found in the documents available.

Discussion

The independence of the **levels of DiD** is an important and constitutive element of the concept of defense-in-depth (WENRA 2013, p. 9):

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. (WENRA 2013, p. 16)

It is particularly emphasized by WENRA that safety features required in postulated core melt accidents (DiD level 4) should be independent, to the extent reasonably practicable, from the SSCs of the other levels of DiD (WENRA 2013, p. 16).

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.

It has already been noted that systematic and comprehensive considerations of multiple failures, and consistent separation of DiD sub-levels 3a and 3b, only seems to have been introduced quite recently according to the information at hand.

Furthermore, there are a number of features foreseen for severe accidents (DiD level 4) which are also used on lower levels of DiD: The primary depressurization system, and the passive heat removal systems from the containment, and via the steam generators.

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

WENRA expectations do not exclude the allocation of systems/features to several levels of DiD. However, independence between levels is expected to the extent reasonably practicable. Thus, it could represent a challenge to demonstrate that a higher degree of independence than foreseen in the current design would not be reasonably practicable, for the features required at DiD level 4 mentioned above as well as for the I&C and possibly also for other cases.

WSO 5 – Safety and security interfaces

Objective:

- Ensuring that safety measures and security measures are designed and implemented in an integrated manner. Synergies between safety and security enhancements should be sought.

Means to achieve objective	Situation at VVER-1200/V491
Airplane crash	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 (SPBAEP 2011, p. 7). Furthermore, there is protection against the impact of an (unspecified) large commercial airplane (STUK 2009a, p. 58). It can be assumed that this event is considered under DEC A. Protection against airplane crash is to be provided by the strength of buildings and the principle of separation (STUK 2009a, p. 58). The reactor building constitutes a robust enclosure in the form of a double ferro-concrete cover. (The space between the covers is connected to the ventilation system which provides for controlled discharge.) It 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 - 1200 mm 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. (MINENERGO 2010b, p. 117-118; SPBAEP 2011, p. 14) Apart from the containment building, the fresh fuel storage building is also designed against the impact of a large aircraft. The tanks for radioactive waste are located underground. The buildings for the main steam valves, the safety systems, the control rooms and the emergency diesel generators are separated by distance, and to some extent also shielded by other buildings (in particular, the containment building). They are not designed to withstand the impact of a large airplane (STUK 2009a, p. 58). Furthermore, it has to be noted that 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. The same applies to the four diesel generators (SPBAEP 2011, p. 20). 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 (STUK 2009a, p. 59).
	Other issues related to security are beyond the scope of this expert statement.

Discussion

The protective design of the reactor building (double ferro-concrete cover) 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.

However, it has to be noted that the **safety building is not designed against airplane crash**. 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.

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 (WENRA 2013, p. 40). 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.

It is mentioned in the EIA-REPORT 2014 (p. 88) that a fuel fire will be taken into account in the design of buildings important to safety. However, this is not discussed or specified further.

It could constitute a **challenge** to demonstrate the availability of the necessary safety functions after the crash of a large airplane, in particular considering the potential common vulnerability of the safety trains in the safety building.

Furthermore, the issue of airplane fuel combustion and/or explosion could constitute a challenge in the safety demonstration for the plant.

WSO 6 – Radiation protection and waste management

Objectives:

- Reducing as far as reasonably achievable by design provisions, for all operating states, decommissioning and dismantling activities :
  - individual and collective doses for workers;
  - radioactive discharges to the environment;
  - quantity and activity of radioactive waste.

The first two items are important, as are all items addressed in the WENRA safety objectives. However, they have little relevance for possible adverse effects on Austria, from the operation of the NPP.

Regarding discharges during normal operation, the dose limit for the reference plant (Leningrad-2) is 0.1 mSv/yr (SPBAEP 2011, p. 9). This corresponds to the dose limit in Finland (EIA-REPORT 2014, p. 79).

It is expected that actual emissions of the reference plant will only cause a radiation exposure of 1–2% of this value in normal operation. Anticipated operational occurrences are expected not to exceed the limit (SPBAEP 2011, p. 10).

The issue of radioactive waste management is addressed in Chapter 8 of this expert statement.

WSO 7 – Leadership and Management for Safety

Objectives:

- Ensuring effective management for safety from the design stage. This implies that the licensee:
  - establishes effective leadership and management for safety over the entire new plant project and has sufficient in house technical and financial resources to fulfil its prime responsibility in safety;
  - ensures that all other organizations involved in siting, design, construction, commissioning, operation and decommissioning of new plants demonstrate awareness among the staff of the nuclear safety issues associated with their work and their role in ensuring safety.

This safety objective goes beyond design aspects of the plant, and beyond the scope of this expert statement. It is not discussed here.

5.2.2 Lessons Learned from Fukushima

There is no specific and detailed discussion of the application of lessons learned from the Fukushima-accident in the available documents on the VVER-1200/V491.

It is reported, however, that a **stress test** was performed **for Leningrad-II** (the reference plant for the VVER-1200/V491) and a full list of external impacts (including earthquake, flooding, tsunami and tornado) was considered.

In particular, an engineering evaluation of the seismic strength of the inner containment was performed to determine the threshold for the seismic impact. The conclusion was that the threshold value of maximum acceleration at ground level is 0.51 g.

Furthermore, complete loss of electric power, loss of ultimate heat sink and combination of both were analyzed (KOLCHINSKY 2013a, p. 17).

Thus, the first two topics of the EU stress tests (natural hazards, and loss of safety functions) appear to have been covered by this stress test for Leningrad-II. The third topic of the EU stress tests (severe accident management) is not mentioned.

For a similar **reactor type**, the **VVER-TOI** (an advanced version of the VVER-1200/V392M), safety analyses for Fukushima accident conditions (as well as an analysis of more severe conditions – station blackout with large-break LOCA) have been performed. It was concluded that the design of this plant type can withstand external impacts combined with internal initiating events and additional failures.

Nevertheless, some measures are considered to enhance plant stability in case of hypothetical events with low probability, to improve spent fuel pool heat removal, to assure long-term make-up for the primary circuit and to improve parameter monitoring. These measures include installation of additional equipment such as air-cooled mobile diesel generators.

The lessons learned from Fukushima are summarized as follows:

- combine active and passive safety systems

- ensure safety functions' performance at different accident stages (redundant power supply sources with guaranteed connections, long-term stable functioning of passive safety systems)
- ensure safety systems' self-sufficiency and diversity, protection against dependent failures under extreme conditions
- ensure NPP accessibility for emergency services during accidents and disasters
- ensure the possibility of replenishment of media and energy sources in case of destruction and blockage
- develop recommendations to enhance operational stability and safety of water-cooled reactors.

(ATOMENERGOPROEKT 2012, p. 18-19)

These items are of rather general nature and mostly correspond to the conclusions which have been drawn in the EU after the Fukushima accident.

5.2.3 Stage of Design Development of VVER-1200

The design of the VVER-1200/V491 was developed by the General Designer "Atomenergoproekt" (St. Petersburg), based on the VVER-1000.

The **reference plant** for this design is **Leningrad-II**. The construction licenses for the first two units of this NPP were granted in June 2008 and July 2009, respectively; construction started soon afterwards (WNN 2010).

Startup was originally planned for 2013 and 2014 (WORLD NUCLEAR INDUSTRY HANDBOOK 2009). Today, the target years for begin of operation of the first two units of Leningrad-II are 2016 and 2018 (WNN 2014a).

There is no published information on the reasons for this delay. There was an incident on 17 July 2011 when the concrete for the outer protective shell of unit 1 was poured. Shortly after the concrete was placed, the reinforcement cage began deforming. However, it is not likely that this mishap led to relevant delays (UPI 2011).

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 (E&T 2013; YLE 2014). 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. In Finland, where this is required, the outer shell of the reactor containment building will have to be thicker (ROSATOM 2014).

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. A **preliminary safety assessment** has been performed by STUK in 2009, when several reactor types were under discussion for a new unit at the Loviisa site (these plans were later abandoned). However, this was a rather summary assessment of how the design objectives and principles of the various plant alternatives met the Finnish safety requirements, and not an extensive review of the design, as will be part of the nuclear licensing procedure.

It is also noteworthy that there are plans to submit a similar reactor type, the VVER-TOI (an advanced version of the VVER-1200/V392M), to a Generic Design Assessment, starting in the beginning of 2015 (ROSATOM 2014).

5.2.4 Discussion of VVER-1200/V491 in Relation to Reactor Types from Earlier EIA

In the **EIA procedure** for the **Fennovoima** NPP project **2008/2009**, the following **reactor types** had been **under consideration**: EPR (Evolutionary Power Reactor), ABWR (Advanced Boiling Water Reactor) and SWR-1000 (Siedewasserreaktor-1000, renamed KERENA in 2009).

The EPR is a pressurized water reactor, like the VVER-1200/V491, whereas the two other types are boiling water reactors. Furthermore, EPR appears to be more relevant since several EPRs are under construction in Europe and worldwide. While a number of ABWRs have been built in Japan, no ABWR is under construction in Europe (although there has been an application for General Design Assessment in the UK in 2013), and no KERENA-type NPP is planned or under construction anywhere in the world.

The **discussio**n in this section is therefore **focused on the EPR**. This discussion can only be performed in a rough and general manner. A more systematic comparison of the two reactor types would be beyond the scope of this expert statement.

The discussion is based mainly on information from the Advanced Reactors Information System (ARIS) of the IAEA (IAEA-ARIS 2011a; IAEA-ARIS 2011b), and the sections on AES-2006 and EPR in the Preliminary Safety Assessment of Loviisa 3 Nuclear Power Plant Project, Appendix 1, of the Finnish regulatory body STUK (STUK 2009a). In cases in which other sources have been used, a reference is provided at the appropriate place in the text.

It has to be noted that considerably more detailed information is available for the EPR. In particular, a Pre-Construction Safety Report has been submitted in the course of the procedure for Generic Design Assessment in the UK, which has been published by AREVA and EdF. No comparable document has been made public for the VVER-1200/V491.

The **designs** of both EPR and VVER-1200/V491 represent evolutionary developments of forerunner types – the EPR has been developed from the German Konvoi and the French N4 PWRs, the VVER-1200/V491 from the VVER-1000.

EPRs are **under construction** in Finland (Olkiluoto 3) and France (Flamanville 3), as well as in China (Taishan 1 and 2). VVER-1200/V491s are being constructed at the Leningrad-2 site and the Baltic site in the Kaliningrad region (two units in each case).

The **electrical capacity** of the EPR (1770 MW) is considerably higher than that of the VVER-1200/V491 (1170 MW). Both plants are capable of load following operation.

Both types display a combination of active and passive **safety systems** and features. The redundancy of safety systems is generally comparable. The concept of defense-in-depth with four levels of defense taken into account in the plant design is applied in both cases. The further development of this concept as proposed by WENRA-RHWG (WENRA 2009), including a separation of level of DiD 3 into two sub-levels (3a – single initiating events, 3b – selected multiple failure events) has not yet been explicitly taken into account in the design of the two reactor types. However, for the EPR, it is reported that a systematic analysis of multiple failures in redundant systems was conducted by AREVA. In the documents evaluated for VVER-1200/V491, the issue of common-cause failures is addressed, but no systematic analysis of multiple failures is reported.

The number of large welds in the **reactor pressure vessel** is minimized in both cases, to reduce the effort required for in-service-inspection. The **steam generators** are vertical in case of EPR, horizontal in case of VVER-1200/V491. Horizontal steam generators make it possible to reduce the height of the reactor building, thus improving seismic resistance.

In both plants, **reactive management** is performed by means of control rods, boron contained in the reactor coolant, and burnable poisons. For **shutting down** the reactor, the diversity principle is implemented with an emergency borating system. In order to prevent re-criticality in case of cooling, the efficiency of the control rods has been improved in the VVER, by increasing the number of control rods in the reactor. As a result, the reactor's re-criticality temperature during a cooling accident without boron addition is exceptionally low, about 100 °C. (No comparable feature is reported for the EPR.)

Regarding **severe accident management**, the EPR is equipped with dedicated valves for depressurization of the primary circuit, to be used exclusively on level 4 of DiD. (VVER has only one set of valves for level 3 and level 4 situations.) Furthermore, filtered venting of the containment is foreseen for the Finnish EPR, but not for the VVER-1200/V491.

Both reactor types are equipped with **core catchers**, but the designs of the core catchers differ significantly: EPR has a catcher with melt spreading, whereas VVER is furnished with a crucible-type catcher. The melt-spreading type of catcher features a pre-catcher in the reactor pit, which contains sacrificial material to secure melt retention for the whole period of melt relocation from the reactor pressure vessel. Subsequently, the melt penetrates through a steel plug and flows into the catcher proper – a large horizontal surface – via a sloping channel. The surface of the catcher is covered by protective and sacrificial materials. The discharge into the spreading area triggers the flooding valves. The coolant flows to the cooling ducts running under the floor and behind the wall elements of the spreading area, and then also on the surface of the melt.

The crucible-type core catcher of the VVER-1200/V491 is described above, in section 5.2.1.

A clear advantage of the spreading-type core catcher is faster melt solidification. The advantages of the crucible type lie in the more compact structure of the solidified corium and – as far as can be deduced from the information at hand – the smaller amounts of hydrogen produced in the course of the accident.

The **containment heat removal system** to be used for severe accidents is passive for VVER, active for EPR.

Their **I&C systems** are digital in both reactor types. In both cases, there are four lines of defense. The third line of defense consists of two systems for the Finnish EPR (one of them hardwired); there is only one (hardwired) system in the third line for the VVER-1200/V491.

For the EPR, the design objectives and principles associated with I&C systems were found to be consistent with Finnish requirements by STUK, whereas the consistency with Finnish requirements could not be established for the VVER-1200/V491 due to lack of information, in particular regarding the separation principle.

The protection of reactor buildings of the two reactor types against **external events** appears to be comparable.

The four redundant safety trains are located side-by-side for the VVER-1200/V491, and not designed to withstand the impact of a large airplane. The building with the emergency diesel generators also contains the four emergency diesels side-by-side and is not designed against the impact.

For the EPR, the four redundant safety trains are separated by distance, and two trains are designed to withstand the impact of a large airplane. Emergency diesel generators are housed in two spatially separated buildings designed against earthquakes and explosions, but not against aircraft crash.

STUK found the design objectives regarding protection against crash of a large commercial airplane to be consistent with Finnish requirements for the EPR; for the VVER-1200/V491 it was found that the fulfillment of these requirements was not yet demonstrated; more detailed designs and analyses as well as plant modifications were seen as required.

The **PSA-results** reported for both plant types are similar, although comparability is limited due to differences in the inclusion of plant states and hazards in the analyses.

Regarding **core damage frequency**, mean values of 5.94E-7/yr and 7.52E-7/yr 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. For the EPR a mean value of 6.4E-7/yr is given (full power and shutdown, including internal and external hazards), and a value of 1.24E-6/yr for the 95%-fractile (AREVA-EDF 2012, p. 12).

For the large release frequency, a mean value of 1.8E-8/yr is provided for VVER-1200/V491 (full power, internal initiating events only), without specification of uncertainty. The large release frequency of the EPR is reported to be 3.94E-8/yr (median value, all plant states and internal and external events except earthquake), with the 95%-fractile at 1.41E-7 (UMWELTBUNDESAMT 2012).

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 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 hazard, 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.

5.3 Conclusions/Recommendations

In section 5.2 of this expert statement, it has become clear that available information, including the information in the EIA-Report, does not permit an assessment as to whether the WENRA safety objectives are fulfilled by the reactor type VVER-1200/V491. It is obvious that relevant efforts have been undertaken. On the other hand, many questions remain open and a number of challenges have been identified.

At present, STUK is preparing a **preliminary safety assessment** of the VVER-1200/V491 which is to be submitted to the Ministry of Employment and the Economy during spring 2014. Subsequently, in case a positive decision on the project is taken on the political level, an assessment which is considerably more detailed will be performed by STUK in the course of the nuclear licensing procedure.

In the **Expert Statement to the EIA program** (UMWELTBUNDESAMT 2013), **recommendations** were listed concerning the discussion of the reactor type, addressing a number of issues which should be included in the EIA-Report (concerning the choice of reactor type, the description safety systems and features and the application Finnish safety requirements and the WENRA safety objectives for new reactors). These points have been addressed in the EIA-Report, but only in a very brief and summary manner, without providing any details.

The EIA procedure in Finland does not stipulate a presentation and discussion of detailed information on the reactor type(s) in question, and their technical specifications. Therefore, it has to be assumed that it will not be possible to resolve the open questions and obtain a definite evaluation of the identified challenges (and, if possible, a resolution of the challenges) in the course of the EIA procedure. The results of the preliminary safety assessment by STUK will be helpful in this respect, and complete clarification can be expected from the assessments and analyses which will be performed by the authority in the course of the licensing procedure.

It would be appreciated if information pertinent to the further course and the results of the preliminary assessment by STUK and the nuclear licensing procedure could be provided once available, with the focus on the fulfillment of WENRA safety objectives for new power reactors, on the efforts undertaken in this respect, and the challenges encountered.

The information provided should **focus** on the **fulfillment of WENRA safety objectives** (WSO) for new power reactors, on the efforts undertaken in this respect, and the challenges encountered. It would be appreciated if, inter alia, the following items are covered in the information provided:

WSO 1:

- operational safety margins;
- selection of materials.

WSO 2:

- conservative selection of assumptions for dealing with design basis accidents, beyond what is customary for Gen II plants;
- 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.

WSO 3:

- demonstration of functioning and reliability of passive safety systems and features;
- reliability of primary depressurization;
- 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;
- demonstration of practical elimination for steam explosion in the reactor pressure vessel, hydrogen detonation and other phenomena.

WSO 4:

- 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.

WSO 5:

- Demonstration of the availability of the necessary safety functions after the crash of a large airplane, taking into account the potential common vulnerability of the safety trains;
- demonstration that buildings or parts of buildings containing nuclear fuel and housing key safety functions are designed to prevent airplane fuel from entering.

Another issue of interest would be a detailed discussion of the application of **lessons learned from Fukushima** for the reactor type VVER-1200/V491.

Recommendation:

It is recommended that the **concept of practical elimination** is applied consistently in the safety requirements for the new nuclear unit. Practical elimination of accident sequences has to be demonstrated with state-of-the-art probabilistic and deterministic methods, fully taking into account the corresponding publications of WENRA.

6 SITE EVALUATION INCL. EXTERNAL HAZARDS

6.1 Treatment in the EIA-Report

In the EIA-Report of 2008, the impacts from a nuclear power plant at **three alternative locations** were evaluated. According to the Decision-in-Principle 2010, Hanhikivi in Pyhäjoki and Karsikko in Simo are suitable locations for a nuclear power plant. (EIA-REPORT 2014, p. 29)

The studies and surveys relating to the plant site have been presented to the Radiation and Nuclear Safety Authority (STUK) in conjunction with the submittal of the original application for Decision-in-Principle in 2009. According to the STUK statement, no issues that would prevent the construction of a new nuclear power plant in compliance with the safety requirements were observed at the new plant site. In October 2013, Fennovoima submitted a report to STUK, which describes the most recent changes concerning the site and any changed information important for the plant site's safety. STUK is currently preparing a statement concerning this matter which will be given to the Ministry of Employment and the Economy in spring 2014. (EIA-REPORT 2014, p. 56)

In the EIA-REPORT (2014, p. 55) it is mentioned that several different factors were taken into account in **selecting the site location**. The assessment revealed no significant differences between the Hanhikivi headland and Karsikko. In the end, the selection of the Hanhikivi headland was supported by, among other things, higher integrity of the bedrock and lower seismic design values.

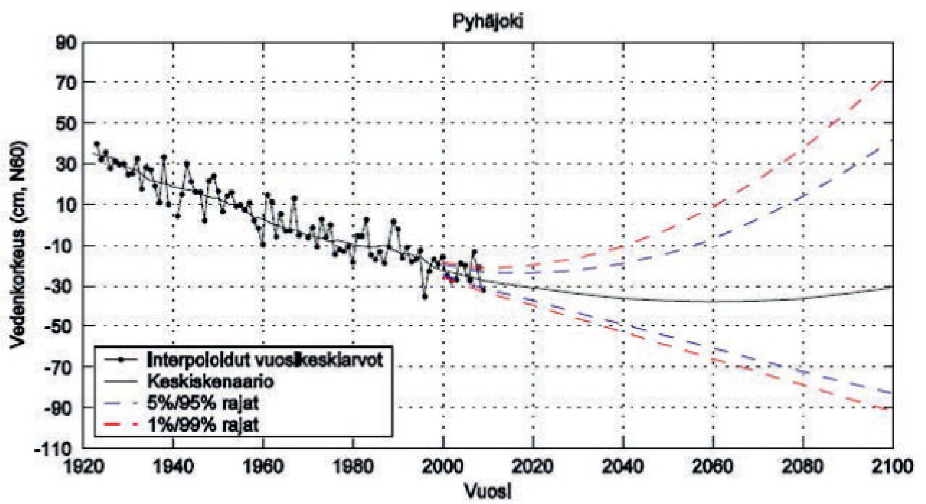
According to chapter 4.3 of the EIA-REPORT (2014, p. 88) the nuclear power plant will be **designed to withstand** the loads resulting from various **external hazards** including extreme weather conditions, sea and ice-related phenomena, earthquakes, various missiles, explosions, flammable and toxic gases, as well as intentional damage. It is mentioned that experience gained from the Fukushima accident has also been utilized in the design.

In chapter 3.3 of the EIA-REPORT (2014, p. 55), it is mentioned that the conditions occurring at the plant site have been examined in numerous different studies and surveys conducted to ensure sufficient consideration of all factors in the design of the nuclear power plant.

Sea-related hazards

In 2008, Fennovoima ordered a **study on the variation of the average water level as well as the extreme values of water level in the project area** from the Finnish Institute of Marine Research. The study was updated in 2010 to include even more rare extreme phenomena. According to the most likely scenario, the effect of land uplift will continue to be greater in the Gulf of Bothnia than the effect of the global rise of the sea levels. The updated study states that the average sea water level in Pyhäjoki will decrease by approximately ten centimeters by the middle of the century and will then return to the current level by the end of the century. However, uncertainties are significant (EIA-REPORT 2014, p. 124).

Figure 7-10 of the EIA-REPORT (2014, p. 125) illustrates the interpolated annual sea level average with uncertainties until the year 2100 (see Figure 1). Figure 1 shows that in 2100, compared to the average value, the sea level average could be 50 cm lower or 70 cm higher (95% fractile)[6].



Source: EIA-REPORT (2014, p. 125)

Figure 1: Interpolated annual sea level averages and the average water level scenario with uncertainties in Pyhäjoki until 2100.

According to the EIA-REPORT (2014, p. 56), in 2075 the sea level will fluctuate between -179 cm and +201 cm. In 2008, the sea water level fluctuation was between -152 cm and 228 cm.

Due to the open coastline, waves have a major impact on the shore. Continuous wave measurements have been carried out in the project area from November 2012 to October 2013 at two locations. According to model simulations by the Finnish Meteorological Institute, significant wave heights of two meters in the summer season and more than four meters in the fall and winter seasons are possible. The highest single waves are approximately twice as high as the significant wave height. (EIA-REPORT 2014, p. 124)

Low and high levels of seawater will be taken into account in the design. In accordance with the YVL Guide B.7, the design value for the nuclear power plant with regard to the sea water level (i.e. the construction elevation) shall be determined by adding a wave margin plus additional two meters to the highest sea water level occurring at the location once in a hundred years.

According to the EIA-REPORT (2014, p. 88), the construction elevation determined for the Fennovoima plant (approximately +4.9 meters according to the N2000 system) fulfills the YVL Guide requirement by a "good margin".

Flooding has also been taken into account in the design of the access roads; the plant site can be accessed via two separate roads, of which at least one will remain available even when the sea water level is exceptionally high (EIA-REPORT 2014, p. 88).

[6] or even 60 cm lower or 100 cm higher (99% fractile)

Sea-related phenomena that may have an effect on the plant's cooling water intake have been investigated. These include pack ice, frazil ice dam (a dam effect caused by formation of ice crystals in subcooled water), migration of sediments, impact of algae, sand and a severe oil spill accident at sea. The intake structure will be dimensioned so that the flow at the intake opening remains low (at approximately 0.2–0.3 m/s). This will minimize the amount of solids carried along by the cooling water. The various impurities will be removed from the seawater by successive screens and filters. The mouth of the cooling water tunnel will be protected with a 15–17 meter wide concrete structure (EIA-REPORT 2014, p.59/60).

It is pointed out, that in addition to the main channel, cooling water can be taken through the auxiliary cooling water intake channel or from the outlet side, if required. Despite these design solutions, provision will also be made for situations in which seawater cooling is totally lost. (EIA-REPORT 2014, p. 88)

Extreme weather phenomena

A study conducted by the Finnish Meteorological Institute assesses the **probability of extreme weather phenomena** relating to temperature, rainfall, snow load and wind velocity in Pyhäjoki, as well as the effect of climate change on the occurrence of these phenomena. Table 3-1 of the EIA-REPORT (2014, p. 56) presents the extreme values of selected natural phenomena with an estimated recurrence level of 1,000 years (see Table 3).

Table 3: Extreme values of weather phenomena with average occurrence of 1/1,000 per year.

Extreme values of weather phenomena		
Temperature (°C)	min	-42.8
	max	33.9
Rainfall (mm)	24 h	84.6
	7 days	126.7
Wind velocity (m/s)	gust, 3 s.	34.7
	average, 10 min min.	31.2
Snow load (kg/m²)		190.5

Regarding extreme natural phenomena, the design values of safety systems will be determined in accordance with the requirements of the YVL Guide B.7, so that they are expected to be exceeded with a frequency lower than once in 100,000 years. Conditions occurring at an even lower frequency will be prepared for. (EIA-REPORT 2014, p. 56)

Earthquakes

In the EIA-REPORT (2014, p. 88), it is mentioned that Finland is located in the central part of the Eurasian continental platform, thus intense earthquakes are very rare and highly improbable. Earthquakes will nevertheless be taken into account in the design of the nuclear power plant. The design basis earthquake was determined in accordance with the requirement of the YVL Guide B.7, so that extremer earthquakes are estimated to occur with a frequency lower than once in 100,000 years. The performance of the most important safety functions will be possible even in the case that the design basis earthquake (DBE) is exceeded.

To assess the occurrence and probability of earthquakes at the plant site and in its vicinity, soil and bedrock surveys have been conducted to examine the seismic properties of the plant site, and to map any faults occurring at the plant site with the help of various soundings and bedrock analyses. Several other surveys have been performed concerning matters such as the geological and geophysical properties at the plant site. (EIA-REPORT 2014, p. 56)

Man-induced events

The probability of an **airplane crash** has been assessed to be extremely small. A no-fly zone will be defined for Fennovoima's nuclear power plant (EIA-REPORT 2014, p. 104/105). According to the EIA-REPORT (2014, p. 56), the nuclear power plant will be constructed so that it will withstand a large commercial airplane crash without significant emissions into the environment. Both, the collision force caused by the airplane itself and the eventual fire caused by its fuel, will be taken into account in the design of the safety important buildings.

There are no heavy industry sites, gas pipelines, railroads, airports or harbors in the immediate vicinity of the plant site. Thus, the risks relating to the transport, handling, and storage of **hazardous substances** are very small. (EIA-REPORT 2014, p. 56)

The nuclear power plant and the nuclear materials used will be protected against **illegal activities**, such as vandalism and sabotage. Threats caused by terrorism or other illegal activities will be addressed through continuous implementation of comprehensive security arrangements. Furthermore, the backgrounds of the personnel working at the nuclear power plant will be checked, and employee access in the plant area will be restricted. (EIA-REPORT 2014, p. 89)

6.2 Discussion

In its Preliminary Safety Assessment 2009, STUK pointed out, the **sea level variation** is relatively large at the site (STUK 2009b). Both low and high see water levels can trigger a dangerous situation for the nuclear power plant.

In the EIA-Report, it is mentioned that according to the most likely scenario - the effect of land uplift - will be greater in the Gulf of Bothnia than the effect of the global rise of the sea levels. However, in the EIA-Report it is also illustrated that the uncertainties of the interpolation of the seawater level average are significant.

Fennovoima states that the construction elevation of the plant is conservatively chosen in relation to most extreme sea water levels. (EIA-REPORT 2014, App. 2. p. 2) However, in the EIA-Report the explanation about the elevation of the site and the additional safety margin is not comprehensible. It is questionable that the envisaged site elevation will ensure sufficient protection against external flooding caused by the extreme sea water level and waves.

Recently, the results of a comprehensive **study on external hazards related to events at nuclear power plants performed on behalf of European Commission** were published. The study addresses both natural origin and man-induced external events.[7] (EC 2013)

For the study, three different event report databases have been searched in order to analyze the operating experience of events related to external hazards:

- The International Reporting System for operating Experience (IRS), operated jointly by the IAEA and the OECD/NEA, a worldwide system containing 3,700 incident reports;
- The SAPIDE database, operated by the IRSN in France, which contains more than 10,000 events reported by French nuclear power plants.
- The VERA database, operated by GRS in Germany. It includes about 6,000 events reported by German plants.

After screening, the relevant events were classified into one of the following groups:

- Extreme weather conditions
- Extreme heat sink conditions
- Flooding
- External fires
- Lightning strikes
- Fouling of the water intake
- Corrosion or chemical fouling caused by external environment
- Man-induced events
- Solar magnetic disturbances

In all three databases, four categories of external events comprise most of the incidents, namely fouling events, extreme weather conditions, lightning strikes and extreme heat sink conditions.

In particular, the issues of **fouling of the water intake** and **extreme heat sink conditions** are of concern for the Hanhikivi nuclear power plant.

The expert statement on the EIA program stated that clogging of the water intake could be a safety issue at the Hanhikivi site. According to the EIA-Report, the seawater intake is designed to minimize the risk of blockage due to low velocity of the water flow and successive screens and filters.

According to the above-mentioned study (EC 2013), one of most important lesson learned for NPPs located in cold areas is that the design should take account of the formation of frazil ice in the water intake channel and include adequate protection and warming systems.[8] According to the EIA-Report, the formation of frazil ice is considered in the design of the water intake. Further information is not provided.

[7] An important exception is the earthquake hazard, which has not been studied in this report as other extensive reports have already covered the matter.

[8] In addition to design provisions, the plant operation should consider that the formation of frazil ice can be a very fast phenomenon and thus should provide appropriate surveillance and operational procedures, for instance early recirculation of hot water to reheat water in cold conditions before ice formation.

Means against biological fouling (e.g. seasonal inrush of fish or massive arrival of vegetable material, mud, etc.) which may clog the water intake, are not mentioned in the EIA-Report.

In the current WENRA document concerning the design of new nuclear power plants (WENRA 2013) it is stated**:** **"**The safety assessment for new reactors should demonstrate that threats from external hazards are either removed or minimized as far as reasonably practicable."

Considering the existing sea-related hazards that could cause the loss of the heat sink (flooding, drought, biological fouling, frazil ice) and the statement of WENRA (2013), it is recommended to consider the implementation of an alternative heat sink (for example a groundwater well).

Solar magnetic disturbances are identified as a relevant event for nuclear power plants (EC 2013), but this natural phenomenon is not mentioned in the EIA-Report. Solar magnetic disturbances[9] can affect the power systems in different manners: power transformers excessive heating, spurious actuation of protective relays or voltage drops. These events can thus be precursors to loss of offsite power or station blackout sequences (EC 2013). Thus, it is recommendable to consider the possible impact of solar magnetic disturbances on the safety of the plant.

According to appendix 2 of the EIA-Report, the **impacts of combined natural phenomena** are also assessed. It is mentioned, for example, that high winds combined with snow storm might have adverse impacts on ventilation openings (EIA-REPORT 2014, App. 2. p. 7). Further details about possible combined phenomena are not provided. In particular, there is no systematic consideration of all possible combinations. The above-mentioned study also recommended to perform a consideration of all combinations of natural phenomena. It is emphasized that the combination of several "minor" phenomena can lead to an "important" external hazard. Therefore, even minor phenomena should be characterized and all possible combinations of these phenomena should be considered.

It is recommended to perform a systematic consideration of all possible combinations of natural phenomena at the Hanhikivi site.

In the EIA-Report, it is mentioned that intense **earthquakes** are very rare and highly improbable. The frequency of occurrence for the design basis earthquake (DBE) of the Hanhikivi 1 nuclear power plant is E-5/yr, in accordance with the requirement of the YVL Guide B.7.

Despite the fact that the Hanhikivi headland is part of a seismically low-active area, it is recommended that the nuclear power plant should be designed against a minimal horizontal peak ground acceleration (PGA) of 0.1 g according to the international state of the art.

The EIA-Report mentioned that the performance of the most important safety functions would be possible even in the case the DBE was exceeded. However, information about the envisaged seismic margins is not provided.

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[9] Solar magnetic disturbances may induce voltage potentials in the earth's crust, which in turn cause geomagnetically induced currents (GIC) to flow in transmission lines

In the **expert statement on the EIA program, it was recommended to provide** a comprehensive **site evaluation** that reflects the international efforts, in particular in the frame of EU stress tests, to enhance the **safety margins of nuclear power** plants against natural hazards.

The required information about site conditions was provided in the EIA-Report. Safety margins are mentioned, but no details are provided.

As regards extreme natural phenomena (extreme weather conditions, earthquakes and sea water level), the design values will be determined in accordance with the requirements of the YVL Guide B.7 so that they are expected to be exceeded with a frequency lower than once in 100,000 years. The use of this **return frequency** complies with the state of the art – but only if the degree of confidence of the estimated frequency of occurrence is justified taking into account the related uncertainties according to the state of knowledge (WENRA 2013).

Fennovoima emphasized that the plant shall be designed according to the latest revision of YVL Guides, which already implement the WENRA safety objectives and lessons learnt from Fukushima (EIA-REPORT 2014, App. 2, p. 1).

According to the EIA-Report, the plant will be designed to withstand the impact of a **crash of a large commercial airplane**. However, it could constitute a challenge to demonstrate the availability of the necessary safety functions after the crash of a large airplane, in particular considering the potential common vulnerability of the safety trains in the safety building (see chapter 5 of this expert statement).

6.3 Conclusions/Recommendations

The **sea level variation** is relatively large at the site. Additionally, the uncertainties of the trend of the average seawater level are significant. Furthermore, wave heights of at least four meters are possible. Altogether, it is questionable that the envisaged site elevation will ensure sufficient protection against external flooding caused by the extreme sea water level and waves. It is recommended to consider the implementation of appropriate further protection of the plant site.

The expert statement on the EIA program stated that **clogging of the water intake** could be a safety issue at the Hanhikivi site. This issue has been addressed in the EIA-Report, but only in a very brief and summary manner, without providing any details.

Regarding the existing sea-related hazards that could cause the loss of the heat sink (e.g. extreme low or high sea water level; clogging caused due biological fouling or frazil ice) – taking into account the statement of WENRA (2013) – it is recommended to consider the **implementation of an alternative heat sink** (for example a groundwater well).

It is recommended to perform a systematic consideration of all possible **combinations of natural phenomena**. In this context, even “minor” phenomena should be characterized, because several “minor” phenomena can lead to an “important” external hazard.

Despite the fact that the Hanhikivi headland is part of a seismically low-active area, the nuclear power plant should be designed against an **earthquake** with horizontal peak ground acceleration (PGA) of at least 0.1 g according to the international state of the art.

In the expert statement on the EIA program, it is recommended to perform a comprehensive site evaluation to enhance the **safety margins of the nuclear power** plant against natural hazards. This issue has been addressed in the EIA-Report, but without providing any details.

As regards extreme natural phenomena (extreme weather conditions, earthquakes and sea water level), the design values of the safety system will be determined, so that the return frequency is expected to be once in 100,000 years. The use of this **return frequency** complies with the state of the art – but only, in case the degree of confidence of the estimated frequency is justified taking into account the related uncertainties.

According to the EIA-Report, the nuclear power plant will be designed to withstand the impact of a **crash of a large commercial airplane**. However, the availability of the necessary safety functions after the crash of a large airplane, in particular considering the potential common vulnerability of the safety trains in the safety building of the AES-2006, are not demonstrated yet.

It has to be assumed that both a comprehensive site evaluation and the design solution concerning external hazards will not be available in the course of the EIA procedure. However, the results of the preliminary safety assessment by STUK will be helpful in this respect. A complete clarification of the issues concerning external hazards can be expected from the assessments and analyses which will be performed by the authority in the course of the licensing procedure.

It would be appreciated if information pertinent to the following topics could be provided once available:

- Evaluation of the design basis flood (DBF) and of cliff-edge effects in case of a beyond design basis flood
- Evaluation of natural events that can cause clogging of the water intake (in particular biological fouling and frazil ice formations) and protection against those events
- Evaluation of the design basis earthquake (DBE) and of cliff-edge effects of the plant in case of a beyond DBE
- Evaluation of safety margins of the nuclear power plant against natural hazards
- Systematic consideration of all possible combinations of natural phenomena
- Degree of confidence of the return frequency of natural phenomena
- Demonstration of the resistance of the plant against a crash of a large commercial airplane (including the impact of the resulting fuel fire)

6.4 Questions

The following questions should be answered within the EIA procedure:

- *Can the determination of the site elevation including safety margins and its justification regarding the sea level variation, wave heights and the respective uncertainties be explained?*
- *Would the implementation of an alternative heat sink (e.g. a ground water well) be possible at the site? Has the implementation of an alternative heat sink, which is independent of the sea water, been considered?*

7 ACCIDENT ANALYSIS AND TRANSBOUNDARY IMPACT

7.1 Treatment in the EIA-Report

Chapter 7.13 of the EIA-REPORT (2014, p. 199) deals with the impacts of accidents. It is stated that the radiation dose to the population was calculated on the basis of a postulated severe accident scenario. It is pointed out that the modeling results are indicative only, and that they are based on assumptions in which the radiation doses were overestimated. More detailed accident analyses will be prepared in the construction and operating license application phases. Then the dose calculations will utilize plant type-specific accident scenarios based on the detailed analyses presented in the safety report, as well as the associated source terms.

The **Government Decree on the Safety of Nuclear Power Plants** (717/2013), which superseded the previous Government Decree (733/2008), distinguishes postulated design basis accidents as follows:

- For accidents with an expected frequency of occurrence lower than once in a hundred years but equal to or higher than once in a thousand reactor operating years, the annual radiation dose limit for the most exposed individual is 1 mSv.
- For accidents with an expected frequency of occurrence lower than once in a thousand reactor operating years, the annual radiation dose limit for the most exposed individual is 5 mSv.

According to the Government Decree (717/2013) the plant will be required to withstand design extension conditions without severe fuel damage. The maximum annual radiation dose for the most exposed individual of the local population is 20 mSv. Design extension conditions are defined as

- accidents involving a combination of anticipated operational transients or a postulated accident (Class 1) and the occurrence of a common-cause failure in the safety system;
- accidents caused by a combination of failures identified as significant on the basis of a probabilistic risk analysis or
- accidents caused by a rare external event (design extension conditions).

The Government Decree (717/2013) sets limits for the radiation exposure of the general public and radioactive emissions due to severe accidents of the nuclear power plant. The limit for the release of cesium-137 (Cs-137) into the environment is 100 terabecquerels (TBq). The release of radioactive substances resulting from a severe accident shall not cause the need for extensive civil protection measures. The matter is further specified in the Radiation and Nuclear Safety Authority (STUK) guidelines.

Table 7-17 of the EIA-REPORT (2014, p. 199) shows the **limits set by authorities for the initiation of civil protection measure**s (see Table 4).

Table 4: Limits set by the authorities for the main protection measures.

Protection measure	Dose limit	Maximum distance within which the need for the measure is allowed
Taking shelter indoors	10 mSv/2 days	20 km
Evacuation	20 mSv/week	5 km
Ingestion of iodine tablets (thyroid dose)	For children 10 mGy, for adults 100 mGy	Not specified

According to YVL Guide A.7 issued by STUK, the expected

- core damage frequency (CDF) shall be less than once in a hundred thousand years (< 10^{-5}/yr).
- frequency of occurrence of a release exceeding the 100 TBq limit for a Cs-137 release shall be less than once in two million years (<$5*10^{-7}$/yr).

Magnitude and timing of the release

In the EIA-REPORT (2014, p. 200) it is explained that the analysis utilized a postulated release corresponding to the severe accident limit value of a Cs-137 release of 100 TBq laid down in the Government Decree (717/2013).

It is emphasized that the AES-2006 plants have several active and passive reactor cooling systems, and therefore the probability of a severe accident is very small. It is clarified that, nevertheless, the plant will be equipped with a core catcher, a passive containment cooling system and a hydrogen removal system to ensure the integrity of the containment even in the case of a severe accident.

In the case of a severe accident, the release would be primarily caused by containment leakage through the inner and outer containment shell or via the filtered exhaust through the vent stack. The release routes would contain the radioactive substances and significantly decrease the amounts released into the environment. Furthermore, the core melt caught in the core catcher would be sprayed with water, which prevents the dispersion of radioactive substances.

In addition to the release of 100 TBq of Cs-137, the release of the most significant radioactive substances (44 different nuclides including 1,560 TBq of radioactive iodine-131) was considered in the calculations. The source term was determined in accordance with the NUREG-1465 report and according to the reactor core inventory, which the burnup influences. The maximum burnup will be 60 MWd/kgU.

The release was postulated to commence six hours after the beginning of the accident, which is seen as a very conservative assumption for a radioactive release of this magnitude.

As the safety solutions of the AES-2006 would keep the release rate low, reaching a total release of 100 TBq of Cs-137 would require the release to continue for up to several weeks. However, it is assumed that the entire release will occur within 72 hours. It is mentioned that with this release rate, the doses will be overestimated . Furthermore, a sensitivity analysis (examining a release occurring within 24 hours) was performed for the release rate to ensure that a change will not immediately cause the exceeding of the limits set by the authorities. (EIA-REPORT 2014, p. 200)

Table 7-18 of EIA-REPORT (2014, p. 202) shows the assumptions made for the modeling (see Table 5).

Table 5: A comparison between the assumptions made for the accident modeling carried out for the 2008 EIA and the 2014 EIA.

Variable/ assumption	2008 EIA	2014 EIA	Justification
Definition of the composition of the source term	SSK 2002	NRC 1995, NRC 1988, NRC 1975	The source term definitions by the US Nuclear Regulatory Commission (NRC) are globally accepted.
The amount of cesium-137 in the release	100 TBq	100 TBq	The maximum release laid down in the Government Decree.
The amount of iodine-131 in the release	960 TBq	1,560 TBq	Based on the reference "NRC 1995"
The amount of xenon-135 in the release	1,570 TBq	180,000 TBq	Based on the reference "NRC 1995"
Weather conditions	Average and adverse weather conditions determined by expert judgment based on weather observations made in 2004–2006.	Dilution factors calculated on the basis of the weather observations made between October 2010 and October 2013 (corresponding to the 95% fractile).	Distributions calculated directly on the basis of the weather conditions offer more accurate radiation dose values.
Start of release from the start of the accident	6 hours and 24 hours	6 hours	Only the more unfavorable release, i.e. the one that starts earlier, has been studied in this EIA.
Duration of the release	1 hour and 6 hours	72 hours	The durations of release used in this EIA have been selected in order to assure overestimation of the doses. However, the selected durations are more realistic than in the EIA of 2008

It is stated that the major release examined corresponds to an INES level 6 accident because the magnitude of the release is approximately 10,000 TBq of iodine-131 equivalents.

According to the EIA-REPORT (2014, p. 200), the consequences of an INES level 7 accident are assessed in conjunction with the presentation of the analysis results. For this purpose, a release five times larger than the one resulting from the accident analyzed, was assumed.[10]

Dispersion calculation model

The calculation of the dispersion of the radioactive release was based on the Gauss dispersion model, which is commonly used around the world. Because the Finnish authority guidelines do not include regulations on calculation pa-

[10] For the postulated INES 6 accident, it was assumed that 100 percent of the noble gases of the reactor inventory will be released, the same amount of noble gases is assumed to be released during the INES 7 accident.

rameters, the modeling utilized dispersion and fallout parameters complying with the German authority guidelines. For distances exceeding 20 kilometers a long-range model was used, and for distances exceeding 150 kilometers the results were extrapolated using a fitted value that overestimates the doses. (EIA-REPORT 2014, p. 200)

A term representing the wind direction and speed, rainfall, and mixing of air currents determined on the basis of weather observation documentation (Pasquill stability class) is applied. Measurement data from the Finnish Meteorological Institute weather observation stations were used. The weather observations were recorded over a period of three years between October 2010 and October 2013. (EIA-REPORT 2014, p. 200)

The release is assumed to occur at a height of 100 meters (EIA-REPORT 2014, p. 201).

Dose calculation

The radiation dose from the radioactive release was calculated separately for children (1–2 years) and adults. Regarding eating habits, the typical Finnish diet was taken into account. The doses were calculated using dose factors determined by the International Commission on Radiological Protection (ICRP). For the purposes of the modeling, it was assumed that no civil protection measures would be implemented. Furthermore, it was assumed that people stay outdoors and consume only locally produced foodstuff. It is claimed that the applied method and assumption overestimate the radiation doses. (EIA-REPORT 2014, p. 201)

Radiation doses from severe accidents

The radiation dose was calculated for three different periods: the first two days, the first seven days, and the entire lifetime. The results are presented in table 7-19 of the EIA-REPORT (2014, p. 202) (see Table 6).

Table 6: Radiation doses in case of a severe accident (95% fractile).

	Radiation dose for a child [mSv]			**Radiation dose for an adult [mSv]**		
Distance [km]	**2 days**	**7 days**	**Lifetime (70 years)**	**2 days**	**7 days**	**Lifetime (50 years)**
1	22.8	29.3	690	14.8	19.0	336
2	12.3	15.7	386	7.9	10.4	189
3	7.7	9.9	253	5.2	6.5	126
4	5.7	7.3	189	3.8	4.8	96
5	4.3	5.6	149	2.8	3.7	76
10	1.9	2.4	71	1.2	1.6	37
15	1.2	1.5	48	0.7	1.0	25
20	0.8	1.1	36	0.6	0.7	19
50	0.4	0.5	15	0.2	0.3	7.4
100	0.2	0.3	10	0.1	0.2	4.8
150	0.2	0.2	8	0.1	0.2	3.8
500	<0.1	<0.1	<3	<0.1	<0.1	<2
1,000	<0.1	<0.1	<2	<0.1	<0.1	<1

It is emphasized that the results of the modeling show that the doses fall below the limits set by the authorities. Also the results of the sensitivity analysis (24 hour release) are below the limits set by the authorities (EIA-REPORT 2014, p. 203). Table 7-20 of EIA-REPORT (2014, p. 204) presents the results (including the results of the sensitivity analysis) compared with the limits set by the authorities (see Table 7).

Table 7: Required civil protection measures and the distances within which the measures must be implemented (defined on the basis of the 95% fractile).

	Dose limit	**Distance within which the protection measure is necessary**		**Maximum distance within which the protection measures may be necessary according to the Decree**
		72 h release	**24 h release (sensitivity analysis)**	
Evacuation	20 mSv/week, children	2 km	3 km	Approx. 5 km (the protective zone)
	20 mSv/week, adults	1 km	3 km	
Taking shelter indoors	10 mSv/2 days, children	3 km	5 km	Approx. 20 km (the emergency planning zone)
	10 mSv/2 days, adults	2 km	4 km	
Ingestion of iodine tablets	10 mGy/2 days, thyroid dose to children	5 km	15 km	Not specified
	100 mGy/2 days, thyroid dose to adults	No need	1 km	

The results of the considered INES 7 accident are shortly mentioned: In the case of a release of five-fold magnitude occurring over a period of 72 hours (corresponding to an accident of INES level 7), evacuation would be required within a radius of five kilometers and taking shelter indoors within a radius of ten kilometers from the plant. (EIA-REPORT 2014, p. 203)

Impacts of radiation exposure

In the EIA-REPORT (2014, p. 203), it is pointed out that the severe accident being studied does not cause any direct health impacts. Even without any protection measures, the radiation dose in the first two days is no more than 23 mSv. It is emphasized that the additional risk of cancer caused by the accident is statistically insignificant at all distances. For a child, the radiation dose (app. 150 mSv) can be estimated to increase his or her risk of getting cancer before the age of 70 by approximately 0.8%. For an adult, the corresponding additional risk of cancer is approximately 0.4%.[11] (EIA-REPORT 2014, p. 204)

[11] The increased risk of cancer is estimated by using the risk factor of the International Commission on Radiological Protection (ICRP). The ICRP (2007) has estimated that exposure to a radiation dose of 1,000 mSv at small doses and dose rates increases the risk of cancer by 5.5%.

Impacts of a radioactive fallout

Table 7-22 of the EIA-REPORT (2014, p. 206) presents the fallout of the most important nuclides (iodine-131, cesium-134, cesium-137 and strontium-90) in the spreading direction of the release at various distances from the power plant.

The EIA-REPORT (2014, p. 206) describes the possible restrictions on the use of agricultural foodstuff and on various kinds of natural products. It is mentioned that extensive long-term restrictions on the use of agricultural foodstuffs will not be necessary. It is also mentioned that long-term restrictions on the consumption of some mushrooms, for example, may be required in areas at a distance of 50 km.

Civil protection measures would be necessary in an area extending max. 15 km from the plant in case of such a release. Short-term restrictions on the use of agricultural and natural products could be necessary. The use of freshwater fish as nourishment may have to be restricted in an area extending around 300 km. The use of reindeer meat may have to be restricted in an area extending up to 1,000 km. (EIA-REPORT 2014, p. 227)

It is explained that the consequences of a release caused by an accident can be clearly minimized by means of civil protection measures, such as seeking shelter indoors, administering iodine tablets, evacuating residents of local areas, and restricting access. Protection measures influencing the food industry and restrictions on the use of foodstuffs can clearly reduce the radiation dose due to food ingestion. (EIA-REPORT 2014, p. 233)

Social impacts of a severe accident

The social impacts of a potential nuclear accident are one of the themes being studied in the CEEPRA[12] project, funded by the EU. First results of this project are presented. (EIA-REPORT 2014, p. 207)

Uncertainties of environmental impact assessment

Chapter 8.5 of the EIA-REPORT (2014, p. 229) deals with the uncertainties of environmental impact assessment. It is explained that several of the technical solutions that were still unfinished when the EIA of 2008 was implemented have been specified (such as the cooling water arrangements and the plant layout). However, the plant's detailed technical design is not complete yet. The description of assessment methods includes an evaluation of the related uncertainties. It is mentioned that all of the assumptions used as the basis of the assessment have been determined by selecting the worst-case scenario in terms of the environment.

Transboundary environmental impacts

Chapter 7.14 of the EIA-REPORT (2014, p. 210) deals with transboundary environmental impacts. As mentioned above, the studied radioactive release caused by a severe reactor accident was the Cs-137 release of 100 TBq laid down in the Government Decree (717/2013). This release corresponds to an

[12] Collaboration Network on EuroArctic Environmental Radiation Protection and Research

INES 6 accident. It is emphasized that the consequences of a release, which is five times larger and which would be categorized as an INES 7 accident, have also been assessed.

According to the EIA-REPORT (2014, p. 210), the analyses included several assumptions to verify that the calculation results and the radiation doses will be conservative. It is highlighted that civil protection measures would not be necessary outside of Finland. Furthermore, it is pointed out that, however, the radiation dose from food could easily be limited through various restrictions on the use of foodstuffs.

Table 7-25 of the EIA-REPORT (2014, p. 211) presents the calculated radiation doses of children and adults at four distances between 100 and 1,000 km. The integration times are 2 and 7 days as well as the entire lifetime (i.e. 70 years for children and 50 years for adults). The maximal doses are 10 mSv (see Table 6).

According to the EIA-REPORT (2014, p. 210), the release would cause a maximum lifetime dose of 8 mSv for a child and of 4 mSv for an adult living on the coast of Sweden (distance 150 km). At the Norwegian border (distance about 450 km), the release would cause a dose of a maximum of 4 mSv for children and 2 mSv for adults. On the coast of Estonia (distance about 550 km), the maximum lifetime dose would be 3 mSv for children and 2 mSv for adults. The dose on the coast of Poland (distance about 1,100 km) would remain below 2 mSv. It is pointed out that even if the weather conditions were unfavorable, the release would cause a lifetime dose of maximal 1 mSv for a resident of Austria (distance about 1,850 km).

A release that is five times higher than the analyzed release (classified as an INES 7 accident) would cause a dose of approximately 37 mSv for children and of 18 mSv for adults on the coast of Sweden. The radiation dose at the Norwegian border could be a maximum of 14 mSv for children and 7 mSv for adults. The radiation doses in the other countries bordering the Baltic Sea would remain below 12 mSv for children and 6 mSv for adults. The lifetime radiation dose in Austria would not exceed 5 mSv for children and 2 mSv for adults. (EIA-REPORT 2014, p. 212)

Table 7-26 of the EIA-REPORT (2014, p. 212) illustrates the fallout of the most important nuclides at four different distances between 150 km and 1,000 km from the plant (see Table 8). The calculated ground deposit of Cs-137 at a distance of 1,000 km is 0.79 kBq/m².

Table 8: Nuclide fallout at the distances of 100–1,000 km from the plant (95% fractile).

	Fallout at various distances (kBq/m^2)				
	100 km	**150 km**	**300 km**	**500 km**	**1000 km**
Cs-134	8.5	5.8	3.0	1.9	0.97
Cs-137	6.0	4.2	2.3	1.4	0.79
Sr- 90	0.67	0.46	0.24	0.15	0.08
I-131 (aerosol)	78	53	28	17	9.0
I-131 (elemental)	11	7.3	3.3	1.9	0.87

It is mentioned that a severe accident could increase the radioactivity of reindeer meat (Sweden, Norway, and the northwestern part of Russia) or freshwater fish (Sweden[13]) to a level that will require temporary restrictions on their use. It is pointed out that, following the restrictions, the radioactivity in reindeer meat or freshwater fish would not pose any danger to people. (EIA-REPORT 2014, p. 211)

7.2 Discussion

In the context of safety, severe accidents are the issue of foremost interest from the Austrian point of view, since such accidents can potentially lead to adverse effects on Austrian territory.

According to the EIA-Report, the radiation dose to the population was calculated on the basis of a postulated **severe accident scenario**. More detailed accident analyses will be prepared in the construction and operating license application phases. Then the dose calculations will utilize plant type-specific accident scenarios based on the detailed analyses presented in the safety report, as well as the associated source terms.

However, despite the fact that plant type-specific accident scenarios are not available, several specific assumptions concerning the type-specific capability to cope with accidents are used. For example, it is emphasized that the AES-2006 has several active and passive reactor cooling systems, and therefore the probability of a severe accident is very small.

Furthermore, it is claimed in the EIA-Report that the release would be primarily caused by containment leakage through the inner and outer containment shell or via the filtered exhaust through the vent stack. The release routes would contain the radioactive substances and significantly decrease the amounts released into the environment. Additionally, the core melt caught in the core catcher would be sprayed with water, which prevents the dispersion of radioactive substances.

However, the EIA-Report does not contain information about the effectiveness of these measures. Thus, no justification of the statements concerning the accident scenarios of the AES-2006 is given in the EIA-Report. The evaluation of the available information concerning this reactor type (see chapter 5 of this expert statement) shows that this justification is not possible with the currently available information.

Many open questions and a number of challenges concerning the capability to cope with accident situations are identified, among others:

- Demonstration of the functioning and reliability of the passive systems for heat removal in severe accident conditions could present a challenge.

[13] In case of the considered INES 7 accident; also in Norway, north-western Russia, and the Baltic states

- Furthermore, the passive heat removal systems are not for the exclusive use in case of a severe accident; the same applies for the only set of valves for primary circuit depressurization.[14]
- 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 a significant challenge.
- The considered design extension conditions (DEC) events seem to be not complete, events like uncontrolled boron dilution, rupture of major pressure-retaining components, uncontrolled level drop during mid-loop operation or total loss of a component cooling system are not included.
- In particular, it could be a challenge to demonstrate “practical elimination” for all phenomena in question (e.g. for steam explosion in the reactor pressure vessel, hydrogen detonation).

Probability of a severe accident

The source term used in the model has been defined according to the Government Decree on Nuclear Safety (717/2013) as a release containing 100 TBq Cs-137. According to STUK’s safety guides, the expectation value for a release bigger than this shall be less than once in 2,000,000 years (5E-7/yr).

In the expert statement on the EIA program (UMWELTBUNDESAMT 2013), it was stated that an accident with a release of not more than 100 TBq Cs-137 does not constitute a worst-case scenario. Severe accidents with releases considerably higher than the limit of 100 TBq Cs-137 cannot be excluded for the AES-2006, even if their calculated probability is required to be less than 5E-7/yr. Moreover, for rare events the probability of occurrence as calculated by a Probabilistic Safety Analysis (PSA) should not be taken as face value, but as an indicative number only. Such analyses are beset with considerable uncertainties, and some risk factors are difficult to include in a PSA.

The published results of PSA studies of the reactor type appear to confirm that the limit of 5E-7/yr for the large release frequency is not exceeded; they are 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, considering the hazards of 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. Thus, it is not clear whether the limit for LRF set by the authority could not be exceeded (see chapter 5 of this expert statement).

During the bilateral consultation in Helsinki on January 28, 2009, Fennovoima argued that even the source term of 100 TBq is an overestimation of the worst case. Fennovoima estimated a probability of less than 5E-9/yr for a release of 10,000 TBq of Cs-137. However, Fennovoima did not deny that, according to the present state of knowledge, the probability for a large release could be higher than its own estimate. In any case, Fennovoima was confident that because of technical improvements, they would reach their goal in the end (probability below 5E-9/yr for a 10,000 TBq of Cs-137 release). The Austrian expert

[14] It is particularly emphasized by WENRA that safety features required in postulated core melt accidents should be independent, to the extent reasonably practicable.

team pointed out that published results of current safety studies did not support Fennovoima's claim of a probability below 5E-9/yr for a large release. Fennovoima's statement could only be taken as a statement of intent to reach such a low probability. (UMWELTBUNDESAMT 2009)

The summarized discussion referred to the plant alternatives which were under consideration in the 2008 EIA-Report. However, the statement of the experts is also true concerning the AES-2006 design: A large release exceeding the limit of 100 TBq of Cs-137 (or even of 500 TBq of Cs- 137) could not be excluded. From today´s state of knowledge, it remains open whether this can indeed be achieved. As explained in chapter 5 of the expert statement at hand, several “challenges” remain.

Source term

In the expert statement on the EIA program, it is recommended that the EIA-Report should present the maximum release in case of a severe accident and more detailed information on the design and safety features of the AES-2006. In addition, parameters that are relevant for the assessment of potential source terms should be given in the EIA-Report: the radioactive core inventory, the average and maximum burn-up of the fuel and a description of the severe accident sequences envisaged.

Besides the maximum fuel burn-up, the mentioned information is not presented in the EIA-Report. Fennovoima only states that in the model the size, timing and duration of the release have been chosen so that they are significantly conservative taken into account the technical design solutions of the AES-2006 power plant. (EIA-REPORT 2014, App. 2, p. 4) However, as mentioned above, a justification for this statement is not given.

In 2012, the Norwegian Radiation Protection Authority published a report concerning the potential consequences in Norway after a hypothetical accident at the nuclear power plant Leningrad II (Russia). 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 (STATENS 2012). This source term is higher compared to those used in the EIA-Report.

In the expert statement on the EIA program, it was recommended to include a conservative worst-case release scenario in the updated EIA-Report, in addition to the limited release scenario according to Finnish regulation, since their effects can be widespread and long-lasting and even countries not directly bordering Finland, like Austria, can be affected. This recommendation was observed to a considerable extent.

In Finland, the government decree on the safety of nuclear power stations (717/2013) sets a release of 100 TBq of Cs-137 as the threshold for a serious accident – this value has been used as the source term that describes an INES 6 class accident in Finnish environmental impact assessments. Several comments and opinions concerning the EIA program received within the consultation phase suggested that the assessment should cover an INES 7 class accident.[15] Therefore, the Ministry of Employment and the Economy (MEE) finds it appropriate that the organization responsible for the project should present a comparison between the assessment used in Finland and an assessment covering an INES 7 class accident. (MEE 2013a, p. 19)

Therefore, the updated EIA-Report also includes the impact assessment of the INES 7 accident. For this purpose, a release which is five times larger than the one resulting from the postulated severe accident is considered in the EIA-Report. (500 TBq of Cs-137) (EIA-REPORT 2014, App. 2, p. 13)

The release of the postulated severe accident corresponds to an INES level 6 accident, because the magnitude of the release is approximately 10,000 TBq of I-131 equivalents[16]. An accident will be categorized as INES level 6 in case of a release of some 1,000 to a couple of 10,000 TBq I-131 equivalents..

The consideration of a release of more than 100 TBq of Cs-137 within the EIA-Report is highly appreciated. However, the considered release of five-fold magnitude (i.e. the release of about 50,000 TBq of I-131 equivalents) represents the lower limit of a release categorized as an INES 7 accident: An accident will be categorized as an INES level 7 accident in case a release is more than a couple of 10,000 TBq I-131 equivalents. It has to be noted that the considered release of 500 TBq of Cs-137 is considerable smaller compared to the release of the INES 7 accident at Fukushima. According to estimations, about 10,000–20,000 TBq of Cs-137 were released during the Fukushima accident

The source term of 100 TBq or of 500 TBq of Cs-137 for severe accidents can only be seen as justified if severe accident scenarios with higher releases can be considered as "practically eliminated", but this is not proven yet. Only results of detailed safety assessments for the reactor would permit to exclude a larger source term – in case it can be proven with a high degree of confidence that such a larger source term is extremely unlikely. However, as mentioned above, such safety assessments are not available for the AES-2006 yet (see chapter 5 of this expert statement).

In addition to the release of Cs-137, the release of the most significant radioactive substances (44 different nuclides) was considered in the EIA-Report. It is stated that information provided by NUREG-1465 has been used. Further details are not provided. It has to be assumed that the concentration of radionuclides in the containment is calculated using the transfer coefficient of radionuclides from the reactor core to the containment on the basis of NUREG-1465, but this should be clarified.

Timing and duration of the release

The release was postulated to start six hours after the beginning of the accident, which is seen as a very conservative assumption for a radioactive release of this magnitude. However, an earlier start of release, which would result in

[15] For example, Denmark ask for including the impacts of INES 7 type severe accident to the EIA-Report.

[16] Specific factors for different nuclides are given to specify the I-131 equivalents, e.g. the factor for Cs-137 is 40, which means a release of 100 TBq Cs-137 is equal to a release of 4,000 TBq Iodine-131

higher radiation doses, could not be excluded. For example, in the above-mentioned study of the Norwegian Radiation Protection Authority it is mentioned that the release starts “instantaneously” after accident initiation (STATENS 2012).

Furthermore, it is assumed that the entire release will occur within 72 hours. According to the EIA-Report, the safety solutions of the AES-2006 would keep the release rate low, thus reaching a total release of 100 TBq of Cs-137 would last several weeks. This assumption is only true in case the containment remains intact during a severe accident. It is not justified for worst-case accident scenarios (loss of containment integrity of containment by-pass).

The duration of release used in the EIA-Report (72 hours) is prolonged compared to the duration of release used in the EIA of 2008 (1 hour and 6 hours). It is claimed that the longer duration is more realistic than the release duration used in the EIA of 2008. However, a recently published report concerning severe accidents in different new reactor estimates release durations of 2 to 14 hours, depending on the reactor type and accident scenario. It is assumed that the release starts 0.25 hours to about 8 hours after the beginning of the accident (SEIBERT et al. 2014)

A sensitivity analysis, which was performed to ensure that a change of the release rate would not immediately cause the exceeding of the limits, considered a release occurring within 24 hours.

The duration of release used in the EIA-Report is not at all conservative. This applies also for the duration time of the sensitivity analysis performed. Shorter release times (e.g. 1 hour or 6 hours as used in the previous EIA procedure) would cause considerably higher radiation doses.

The results of the release with the assumed duration of 24 hours are not presented in the EIA-Report. However, the comparison of the distance within which protection measures are necessary indicates considerably higher results. For example, the ingestion of iodine tablets for children is required within 5 km in case of the 72 hours release and within a distance of 15 km in case of the 24 hours release.

Dispersion calculation model

According to the EIA-Report, for distances exceeding 20 kilometers a long-range model was used. In the 2008 EIA-Report, a Gaussian Puff Modell is used to calculate the dispersion for long range (more than 20 km). It is assumed that the Gaussian Puff Modell is applied again.

For the dispersion calculation in the frame of the EIA 2008, a deterministic approach was used. In addition to the average (“typical”) weather conditions, adverse ("unfavorable”) weather conditions were used as input parameters for the calculation. The consequences of the different weather conditions (in terms of deposition and doses) differ by a factor of more than four.[17]

[17] In the 2008 EIA-Report, the calculated Cs-137 deposition was about 1.3 kBq/m² for “unfavorable” and 0.28 kBq/m² for “typical” weather conditions (UMWELTBUNDESAMT 2008, 2010)

The use of a Gaussian model for the assessment of transboundary impacts with unfavorable weather conditions as a worst-case scenario is an acceptable approach (UMWELTBUNDESAMT 2010). This conclusion of the expert statement on the EIA 2008 does not apply to the approach used in the updated EIA. The approach of dispersion calculation has been changed: a probabilistic approach is applied. According to the EIA-Report, the justification for the new approach is that distributions calculated directly based on the weather conditions offer more accurate radiation dose values.

However, the use of a probabilistic approach is not conservative. Using both a deterministic and probabilistic approach is to be considered appropriate according to the state of the art.

The use of a release height of 100 meters is appropriate to calculate transboundary impacts. The assumption of the EIA-Report concerning dose calculation (no implementation of civil protection measures, people staying outdoor for 24 hours and consumption of local food) presents a state of the art approach.

A comparison of the results of the EIA 2008 and the EIA 2014 shows a decrease of the calculated impacts of a severe accident, because of the changes of the input parameter (prolonging of the duration time) and of the approach of dispersion calculation (probabilistic). According to the EIA 2014, the Cs-137 ground deposition at the distance of 1,000 km is 0.79 kBq/m². In the 2008 EIA-Report, the calculated Cs-137 deposition was about 1.3 kBq/m² for "unfavorable" weather conditions (UMWELTBUNDESAMT 2010).

In the EIA-Report, it is mentioned that all of the assumptions used as the basis of the assessment have been determined by selecting the worst-case scenario in terms of the environment. However, as explained above, this is in fact not true; not all of the assumptions have been determined by selecting the worst-case scenario.

Transboundary impacts

The Swedish meteorological institute (SMHI*)* notes in its statement on the EIA program that in case of serious reactor accidents, radioactive emissions spread over a very large area. Restricting the examination proposed in the EIA program to a radius of 1,000 kilometers from Pyhäjoki is thus inadequate, and the examination of the geographical distribution of radioactive substances should be extended. (MEE 2013a, p. 12)

In the expert statement on the EIA program, it was recommended to present all results of the dispersion calculation as well as results at different large distances.

However, only results up to the distance of 1,000 km are presented. The EIA-Report mentioned only the lifetime doses at different distances corresponding to the countries participating in the transboundary EIA procedure.

Furthermore, the presentation of the possible impact of the considered INES 7 accident is not at all sufficient. Some results are lacking, other results are only mentioned in the text, but not presented in tables.

However, up-scaling of the calculated Cs-137 ground deposition at the distance of 1,000 km (0.79 kBq/m²) for the INES 6 accident by a factor of five, a Cs-137 ground deposition of about 4 kBq/m^2 in case of the considered INES 7 would result. A rough extrapolation of this value to a distance of 1,850 km (boarder of Austria) yields an estimated value of 1 - 2 kBq/m² of the possible Cs-137 ground deposition in Austria.

If a ground deposition beyond a certain threshold can be expected in Austria, a set of agricultural intervention measures is triggered. The measures include earlier harvesting, closing of greenhouses and covering of plants, putting livestock in stables etc. For these measures, Austrian and German authorities defined a threshold for Cs-137 ground deposition of 0.650 kBq/m². (FLEXRISK 2013; SKKM 2010; SSK 2008).

The calculations of the EIA-Report indicate wide spreading transboundary effects in case of a severe accident (categorized as an INES 7 accident). The values of Cs-137 ground contaminations exceed the threshold that triggers agricultural intervention measures in Austria.

Austrian analyses of transboundary impacts

In the expert statement on Fennovoima´s 2008 EIA-Report, possible transboundary effects were evaluated. The Austrian experts used a source term (25,000 TBq of Cs-137) that corresponds to about 5% of the EPR core inventory to analyze the possible transboundary impacts after a severe accident in a nuclear power plant at the Hanhikivi site.[18] The scenario with the most negative consequences for Austria was described as follows: The central part of the country would be contaminated with more than 40 kBq/m² and the whole area to the east of the line Salzburg – Klagenfurt would be contaminated with more than 10 kBq/m² (UMWELTBUNDESAMT 2008).

The results show that even if the source term is smaller by one magnitude (as used in the above-mentioned study of the Norwegian Radiation Protection Authority) the calculated contaminations (1–4 kBq/m²) are above the threshold that triggers agricultural intervention measures in Austria (0.65 kBq/m²) .

In the expert statement on the 2008 EIA-Report, the experts mentioned a study performed in 2004. This study carried out on behalf of the Austrian Federal Ministry of Agriculture and Forestry, Environment and Water Management analyzed the probability of weather conditions in Europe that a severe accident could affect Austrian territory to an extent that would require protection measures. The calculated risk that a severe accident in Finnish NPP would cause a significant impact on Austria territory is in the range of 1–5 percent.[19] (ÖÖI 2008)

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[18] The Austrian experts pointed out that even this source term does not constitute the maximum conceivable release. Other accident scenarios (failure of reactor pressure vessel at high pressure or containment bypass via uncovered steam generator tube leakage) can lead to cesium releases of more than 50% of the core inventory

[19] The source term of 67,500 TBq of Cs-137 was used, which was assumed to be a large release due to a severe accident at a 1,000 MW_e pressurized water reactor

Although the probability of such weather situations is small, an impact on Austria due to a severe accident at a Finnish nuclear power plant cannot be excluded.

In the expert statement on the EIA program, calculations of the recently published FlexRISK project were used for the estimation of possible impacts of a severe accident at the proposed nuclear power plant Hanhikivi 1 (FLEXRISK 2013). Using source terms and accident frequencies as input[20], for each reactor in Europe an accident scenario with a large release of nuclear material was selected. The accident scenarios are core melt accidents with containment bypass or containment failure. Using the Lagrangian particle dispersion model FLEXPART, both radionuclide concentrations in the air and their deposition on the ground were calculated and visualized in graphs. The Cs-137 ground deposition is used as the contamination indicator.

To estimate the Cs-137 deposition after a severe accident at the Hanhikivi site, a source term of 54,460 TBq of Cs-137 was used. This source term was assumed for a severe accident[21] at an AES-2006 plant[22]. The results of this estimation have shown that for several weather conditions[23] the calculated Cs-137 deposition in Austria is above 1 kBq/m². These values are higher than the threshold that triggered agricultural intervention measures, thus Austria would be affected. The maximal value of the Cs-137 deposition is 30 kBq/m² (UMWELTBUNDESAMT 2013).

7.3 Conclusions/Recommendations

Severe accidents with releases considerably higher than 100 TBq of Cs-137 cannot be excluded for the AES-2006, even if their probability is required to be below 5E-7/yr. Only results of detailed safety assessments for the reactor would allow to exclude a larger source term – in case it can be proven with a high degree of confidence that such a larger source term is extremely unlikely occur. Such safety assessments, however, are not provided in the EIA-Report and not available for the AES-2006 yet.

Rough calculations on severe accidents of the AES-2006 at the Hanhikivi site based on source terms evaluated in the flexRISK project (54,460 TBq of Cs-137) as well as in a study of the Norwegian Radiation Protection Authority (2,800 TBq of Cs-137) presented in UMWELTBUNDESAMT (2013) show possible consequences in Austria. With the release of 100 TBq of Cs-137 such consequences would not be expected.

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[20] Data was collected from plant-specific probabilistic safety analyses (PSA), report of the International Atomic Energy Agency (IAEA), publications in journals, etc.

[21] Concerning the selected accident scenario, it is only mentioned that it is steam generator tube rupture (SGTR).

[22] In the flexRISK project, it was assumed that an EPR is in operation at the Hanhikivi site, thus the results were converted.

[23] These correspond to nine of 88 real weather situations in 1995 (1 January; 9 April, 7 May, 11 May, 21 June, 28 July, 29 August, 2 November, 27 November)

Thus, the expert statement on the EIA program recommended to include a conservative worst-case release scenario in the EIA-Report, in addition to the limited release scenario according to Finnish regulations, since their effects can be widespread and long-lasting and even countries not directly bordering Finland, like Austria, can be affected.

This recommendation was observed to a considerable degree. On request of Ministry of Employment and the Economy (MEE), in addition to the postulated accident with a release of 100 TBq of Cs-137, a severe accident with a release of the five-fold magnitude was considered.

It is highly appreciated that the consequence of a release of more than 100 TBq Cs-137 is considered in the EIA-Report. However, a release of 500 TBq of Cs-137 represents the lower limit of a release corresponding to an INES 7 accident. Severe accidents with larger releases cannot be judged as practically eliminated on the basis of the information provided or available.i. Thus, the release of 500 TBq of Cs-137 does not represent a true worst-case accident scenario.

However, even the INES 7 accident as considered in the EIA-Report indicates consequences for the Austrian territory in case of a severe accident at the Hanhikivi site.

The EIA procedure in Finland does not stipulate a presentation and discussion of detailed information on the reactor type(s) in question and their technical specifications. Therefore, it has to be assumed that it will not be possible to obtain information about specific accident scenarios in the course of the EIA procedure.

It would be appreciated if information pertinent to severe accident scenarios with source terms, timing and duration of the release, calculated frequency of occurrence (including uncertainties) could be provided once available. It is recommended to perform a conservative worst-case release scenario which is based on specific accident analysis of the AES-2006/V-491 once this information is available.

7.4 Questions

- Can you provide the interpolated results of the Cs-137 ground deposition in case of the considered INES 7 accident at the distance of 1,850 km from the Hanhikivi site (distance to the Austrian border)?
- Is it possible to perform a dispersion calculation of the considered INES 7 accident with a release time (1 hour) which corresponds to a conservative worst-case release scenario?

8 RADIOACTIVE WASTE MANAGEMENT

8.1 Treatment in the EIA-Report

In addition to the nuclear power plant itself, the project comprises the on-site interim storage of spent nuclear fuel (SNF), as well as the treatment, storage, and final disposal of low and intermediate level radioactive waste, the dismantling of the nuclear power plant, and handling and disposal of dismantling waste. (EIA-REPORT 2014, p. 31)

Quantity of spent fuel

According to the EIA-REPORT (2014, p. 73) , approximately 20–30 tons of uranium will be removed as spent fuel from the reactor of the nuclear power plant each year - an approximate of **1,200–1,800 tons of spent nuclear fuel** will be generated over the course of the 60 years of operation of the nuclear power plant.

Interim storage of spent fuel

After removal from the reactor, the spent fuel will cool down in the reactor hall water pools for 3-10 years. After that, the spent fuel will be transferred to interim storage, where it will remain for a **minimum of 40 years**.

The type of intermediate storage has not been chosen yet: **Water pools or dry storage** will be used. In the dry storage, spent fuel will be stored in capsules that have been designed for this particular use and that are cooled down passively by utilizing the circulation of air. Different concepts of dry storage are mentioned. The water pools are typically located in buildings made of steel-reinforced concrete or equivalent structures. In this type of storage water acts as a radiation shield and cools down the spent fuel.

The spent fuel interim storage facility will be **built in the power plant area** similarly to the currently existing power plants in Loviisa and Olkiluoto, where interim storage takes place in water pools. The interim storage concept will be presented in the power plant construction license application. The facility will be constructed within approximately ten years of the commissioning of the plant. (EIA-REPORT 2014, p. 74)

Final storage of spent fuel

According to the Nuclear Energy Act, the producer of nuclear waste is responsible for the management of the spent nuclear fuel it has produced until the sealing of the repository. The producer shall make financial provision for the costs arising from the management of nuclear waste by making make an annual payment to the National Nuclear Waste Management Fund, administered by the Ministry of Employment and the Economy. (EIA-REPORT 2014, p. 73)

According to the Finnish Nuclear Energy Act, all nuclear fuel spent in Finland must be processed in Finland – **reprocessing** therefore is **no option**, as there are no reprocessing facilities in Finland. (EIA-REPORT 2014, p. 75)

A **separate EIA procedure** and a Decision-in-Principle by the Government are required for the final disposal of spent nuclear fuel. According to the EIA-Report (2014, p. 33), Fennovoima's primary plan for the spent nuclear fuel is to join the current Finnish nuclear power plants' spent nuclear fuel final disposal system. In March 2012, the Ministry of Employment and the Economy appointed a work-group to control the joint studies of the nuclear power companies on the available alternatives for storing spent nuclear fuel. The Ministry published the workgroup's final report in January 2013. The report's most important recommendations were that an optimized solution would be the most cost-efficient way to handle the final disposal and the expertise gained by Posiva Ltd in its project should be utilized.

The 2010 Decision-in-Principle for Fennovoima requires that Fennovoima shall at the latest on June 30, 2016 present to the Ministry of Employment and the Economy either an agreement on the cooperation with the parties currently in charge of nuclear waste management or an environmental impact assessment program relating to Fennovoima's own spent nuclear fuel disposal plant." (EIA-REPORT 2014, p. 33)

The EIA-Report (2014, p. 76) mentions that Fennovoima is **currently preparing an overall plan on the final disposal of SNF**: "The matters discussed in the plan including a preliminary schedule and interests in common with the current operators regarding their final disposal project. One of the main goals of the overall plan is to determine an optimal final disposal solution which can, for its part, promote cooperation between Fennovoima and the other parties under the nuclear waste management obligation."

The current understanding is that the spent fuel generated in Fennovoima's nuclear power plant will be disposed of in the Finnish bedrock via **geological final disposal**. According to the EIA-REPORT (2014, p. 76), "The disposal would utilize the KBS-3 (Kärn Bränsle Säkerhet) technology developed in Sweden (SKB Svensk Kärnbränslehantering AB) and Finland (Posiva). As the disposal of spent fuel will not begin until the 2070s, the technological developments in the field can be taken into account in the planning of Fennovoima's final disposal operations."

Low and intermediate level waste (LILW)

The estimated volume of waste requiring final disposal generated over the entire service life of the plant (60 years) is approximately 5,000 m³. A more detailed estimation and break-down in different waste categories is given in Table 3-5 of the EIA-REPORT (2014, p. 71). According to the EIA-Report, the table "shows an estimate of the volumes of low and intermediate level waste generated at a plant with a power of about 1,200 MW" after treatment and packaging.

The EIA-REPORT (2014, p. 70) states the classification system used for LILW:

- very low level waste (VLLW)
  activity concentration does not exceed 100 kBq/kg
- low level waste (LLW)
  activity concentration does not exceed 1 MBq/kg
- intermediate level waste (ILW)
  activity concentration 1 – 10,000 MBq/kg

Very low level waste could be disposed of in a separate surface repository – the decision whether the construction of this facility is feasible (and thus whether VVLW will be used as separate waste category) will be made after the estimates on the waste presented by the plant supplier have been confirmed. VLLW can be released from supervision once the radioactivity has been reduced to an adequately low level (annual dose to the general population or the personnel does not exceed 10 µSv). (EIA-REPORT 2014, p. 71)

The EIA-REPORT (2014, p. 71-72) describes the methods for processing of solid, wet and liquid operating waste as well as their environmental impact (p. 194-199). No details on the waste treatment plants are given.

Interim storage of LILW

According to the EIA-REPORT (2014, p. 72), "Packed and characterized waste will be stored under supervision in a storage building located in the immediate vicinity of the solid waste treatment facilities in the plant area. According to the plan, enough storage capacity for 10 years will be built for very low, low, and intermediate level waste."

Final storage of LILW

An operating waste repository for the disposal of LILW will be constructed in the bedrock of the plant area, at a depth of approximately 100 meters. The repository will be taken into operation no earlier than 10 years after the first startup of the NPP. The repository will be either of the rock silo or the tunnel type, the tunnel type being more probable. (EIA-REPORT 2014, p. 72-73)

Fennovoima is considering the construction of a surface repository for very low level waste. The VLLW repository would be ready two years after the first startup of the NPP. The alternative would be to dispose of the VLLW in underground repositories together with the other LILW. (EIA-REPORT 2014, p. 72-73)

Decommissioning

A separate EIA procedure will be applied to the decommissioning of the nuclear power plant (EIA-REPORT 2014, p. 31). The delayed dismantling strategy will be applied - the dismantling will only commence after a long period of conversation. The decommissioning plan and the cost estimate will be further specified for the power plant operating license application. After this, the plan shall be updated every six years. (EIA-REPORT 2014, p. 76-77)

8.2 Discussion

The expert statement on the **2008 EIA-Report** concluded that **radioactive waste management** was **presented** in the EIA-Report **in a very general manner**. Different technological options for interim storage, final disposal of spent fuel and high and intermediate level radioactive waste were described, but without concrete decisions on technology and location of the facilities. The same appears to be true for the 2014 update of the EIA-Report: Fennovoima has not yet developed a comprehensive nuclear waste management strategy.

The Finnish Radiation and Nuclear Safety Authority (STUK) required in their statement within the Scoping Phase of the Fennovoima EIA in 2013, that the EIA-Report should examine questions related to nuclear waste management options. Nuclear waste management measures implemented in the power plant area must be comprehensively described, including their environmental and radiation impacts.

Quantity of spent fuel

According to Directive 2011/92/EU Annex IV, a description of the project, including an estimate, by type and quantity, of expected residues and emissions resulting from the operation of the proposed project is a requirement of an EIA-Report.

Nevertheless, only a rough estimation on the quantity of spent fuel is made in the EIA-Report: approximately 1,200–1,800 tons of spent nuclear fuel will be generated over the course of the 60 years of operation of the nuclear power plant, although the reactor type has already been decided upon. This estimation covers a range of 50%. The expected quantities of spent fuel should be concretized or reasons for the uncertainties of the expected quantities should be given.

Interim storage of spent fuel

The **expert statement to the EIA program** (UMWELTBUNDESAMT 2013) recommended to include the following topics into the updated EIA-Report: „For the demonstration of a proper waste management and to evaluate the possible risk due to a possible accident at the interim storage facility, it is recommended that in the updated EIA-Report. Fennovoima should declare the planned type of interim storage, its capacity and the schedule of the construction works. The intended duration of interim storage should also be clarified."

The **EIA-Report does not give the information required by the expert statement to the EIA program (UMWELTBUNDESAMT 2013)** – although the choice of interim storage and its capacity are essential for assessing the risk as already explicated in UMWELTBUNDESAMT (2013, 2008). "The risk of wet storage facilities compared to dry storage facilities is much higher. Generally, a severe accident in a wet spent fuel storage facility at the Hanhikivi site could affect Austrian territory. Among other issues, the enhanced vulnerability of wet storage facilities to terrorist attacks has been criticized by the IAEA (2007): An attack that partially or completely drained a spent fuel pool could lead to a propagating zirconium cladding fire and to the release of large quantities of radioactive materials to the environment.

Loss of coolant and a subsequent fire can also occur accidentally, either due to earthquakes of very large magnitude or the drop of spent fuel casks – although the probability of this kind of accident is considered to be very low. In addition, the source term in case of a severe accident is higher for wet storage – as it stores a large inventory of radioactivity under a relatively vulnerable shielding." Therefore, this information should be provided to the Austrian side.

The EIA program assumed the interim **storage time** with 20-40 years, which was short considering that Fennovoima probably has to develop its own final disposal of spent fuel. In the EIA-Report, the stated storage time is now a minimum of 40 years, thus leaving open more time for the solution of the final storage problem. As the duration of interim storage is important for the evaluation of the risk, this information should be concretized.

Final storage of spent fuel

The **expert statement to the EIA program** (UMWELTBUNDESAMT 2013) stated that a **decision about the final disposal strategy of spent fuel** is of interest from the Austrian point of view. In particular, in case it is intended to construct an own final disposal by Fennovoima, a **time schedule** as well as information on the **sites** envisaged and its timely availability should be provided in the EIA-Report.

The **EIA-Report does not contain this information**. This is permitted according to the Finnish Nuclear Energy Act, a separate EIA must be carried out for the final disposal.

Fennovoima is currently preparing an overall plan on the final disposal of SNF including a preliminary schedule. This plan has to be finalized before June 30, 2016 when an agreement on the **cooperation** described in the application for the Decision-in-Principle with the parties currently in charge of nuclear waste management **or** an environmental impact assessment program relating to Fennovoima's **own spent nuclear fuel disposal plant** has to be presented to the Ministry of Employment and the Economy.

One approach for Fennovoima would be to **join Posiva**, which would then **expand** plans for **ONKALO**. However, there was no agreement on this due to apparent limits to expansion and Fennovoima was left at risk of failing to present an available disposal route as required for a construction license (NEI 2013b). Posiva is planning a final disposal of spent fuel, which was set up in 1995 as a joint venture company of TVO (60%) and Fortum (40%). Posiva has well advanced plans for a deep geological repository for encapsulated used fuel at the Olkiluoto island. In 1999, Posiva applied for a decision in principle for the final disposal. The decision in principle was issued by the Government at the end of 2000 and ratified by Parliament in May 2001. Construction on the ONKALO underground rock characterization facility commenced in 2004. Posiva applied for a construction license for the final repository for 9,000 tons of used fuel from Olkiluoto and Loviisa and the encapsulation plant in December 2012. The operating license application is expected in 2020, with a view to operation from 2022. Current plans envisage the sealing of the repository in 2120, although this depends on whether the repository accepts waste from reactors built after Olkiluoto 3 and the operational lifetime of those reactors. Construction of new disposal tunnels will continue progressively in parallel with operation. Posiva proposed that the final size of the repository should be increased from the planned capacity of 6,500 tons of used fuel to 12,000 tons – large enough to accommodate waste from Olkiluoto 4 and the proposed Loviisa 3. In July 2010, Parliament voted in favor of an expansion to 9,000 tons to accommodate the used fuel from Olkiluoto 4. (WNA 2014).

Posiva claims that it will have no space in the planned repository for fuel from Fennovoima (WNA 2014). **Posiva's plans do not include** accommodation for spent fuel from **Fennovoima**'s nuclear power plant, and Posiva, TVO and Fortum have routinely said they will not accept Fennovoima as a partner. Early in 2012, the government threatened to use its legal authority under the Nuclear Energy Act if necessary to ensure that Fennovoima fuel would be included, but when this did not break the impasse they set up a working group to make recommendations. According to the working group's final report in January 2013, Posiva and Fennovoima's Hanhikivi should continue negotiations to find a solution for final storage of spent fuel that takes advantage of Posiva's experience. It declined to take a position on whether one or two repositories should be built, but said that the difference in cost would be insignificant.

In its statement to the EIA program on request of the MEE, *Posiva Oy* notes that rather than assume responsibility for the final disposal of all spent nuclear fuel produced in Finland, it is only tasked with managing the final disposal of spent nuclear fuel produced by its owners Fortum Power and Heat Oy (FPH) and Teollisuuden Voima Oyj (TVO) (MEE, p. 11).

In the 2008 EIA-Report, Fennovoima did not clarify whether they intended to use Posiva's final disposal. During the bilateral consultation in Helsinki (2009), Fennovoima clarified that they definitely would prefer an agreement with Posiva (UMWELTBUNDESAMT 2010). However, today **it seems that Fennovoima is forced to develop its own final disposal of spent fuel**.

According to a media release in January 2013, Fennovoima stated that it will continue the preparation of an **environmental impact assessment program of its own nuclear waste final disposal solution**, which will present a number of alternative final disposal sites (NEI 2013a). Geological final disposal is considered the safest long-term method of storing high level radioactive waste and spent fuel at present. However, no country worldwide is yet operating such a geological repository. Thus, it is an ambitious task of Fennovoima to develop such a final disposal in a relatively short time frame.

The progress and timetable on Fennovoima's EIA on SNF disposal are not presented in the EIA-Report.

A time schedule as well as information on the sites envisaged in the case Fennovoima has to construct its own final disposal facility should be provided.

Low and intermediate level waste (LILW)

According to Directive 2011/92/EU Annex IV a, description of the project, including an estimate, by type and quantity, of expected residues and emissions resulting from the operation of the proposed project is a requirement of an EIA-Report.

According to the EIA-REPORT (2014, p. 71), table 3-5 "shows an estimate of the volumes of low and intermediate level waste generated at a plant with a power of about 1,200 MW". It is, however, not clear, to which reactor type the table refers to and to what extent this data is applicable to the AES-2006/V491 waste volumes.

The given **total annual volume of 77.3 m³ of LILW** after treatment and packaging seems to be in accordance with the EUR, which require generation of not more than 50 m³ of LILW per 1000 MW per year.

Interim/final storage of LILW

The **expert statement to the EIA program** (UMWELTBUNDESAMT 2013) recommended to include the following topics into the updated EIA-Report:

"Fennovoima should give details about the **site** of the final repository of low and medium-level waste and its **depth**. Information about the **geological suitability** of the considered sites for the storage should be provided as well."

The EIA-Report states, that the repository will be constructed in the bedrock of the plant area, at a depth of approximately 100 meters, but doesn't give information on the site and its geological suitability.

8.3 Conclusions/Recommendations

Radioactive waste management is presented in the EIA-Report in a general manner as Fennovoima has not yet developed a comprehensive nuclear waste management strategy. This approach is in line with the Finnish Nuclear Energy Act -more concrete plans are currently being developed and will presumably only be finalized after the EIA procedure.

Whenever possible, additional information on RAW management should be given within the EIA procedure – information already available through the plan of final disposal of SNF currently being prepared should be made available. A timetable should be provided which states when open questions can be answered.

It would be appreciated if information pertinent to the following topics could be provided once available:

Spent Fuel

- Only a rough estimation on the quantity of spent fuel is made in the EIA-Report. The expected quantities of spent fuel should be concretized.
- Fennovoima should declare the planned type of interim storage for SNF (wet or dry storage), its capacity and the schedule of the construction works.
- In the EIA-Report, the stated interim storage time of SNF is a minimum of 40 years. As the duration of interim storage is important for the evaluation of the risk, this information should be concretized.
- The decision about the final disposal strategy of SNF is still of interest from the Austrian point of view. In the case Fennovoima has to construct its own final disposal facility, a time schedule as well as information on the sites envisaged should be provided and the progress and timetable of Fennovoima's EIA on SNF disposal should be made available.

LILW

- More information on the LILW waste treatment plants and on the geological suitability of the on-site LILW repository should be given.

8.4 Questions

The following questions should be answered within the EIA procedure:

- When will the choice of interim storage be made? Is there a currently favoured option?
- When can the decision about the final disposal strategy of spent fuel be made available?
- In case Fennovoima has to construct its own final disposal facility: (When) can the progress and timetable of Fennovoima's EIA on SNF disposal be made available?

9 SUMMARY OF RECOMMENDATIONS

After the Decision-in-Principle, a much more detailed assessment of the nuclear power plant project will be performed by STUK, in the course of the nuclear licensing procedure. As the EIA procedure has to be completed before the Decision-in-Principle can be issued, most of the safety-relevant questions cannot be adequately answered within the EIA process. Whether the reactor will comply with the requirements discussed within the EIA process, can only be answered in the following approval procedure.

Therefore, the final statement of the MEE should require the applicant to provide relevant information after the EIA procedure, especially on topics which came up during the EIA procedure but couldn't be answered at this stage.

It would be appreciated if information pertinent to the topics described below could be provided once available.

In the current chapter also recommendations to reduce the risk of severe accidents are given.

Chapter 5 "Reactor type"

Recommendation:

It is recommended that the **concept of practical elimination** is applied consistently in the safety requirements for the new nuclear unit. Practical elimination of accident sequences has to be demonstrated with state-of-the-art probabilistic and deterministic methods, fully taking into account the corresponding publications of WENRA.

Request for information:

It would be appreciated if information pertinent to the further course and the results of the preliminary assessment by STUK, and the nuclear licensing procedure, could be provided once available.

The information provided should **focus** on the **fulfillment of WENRA safety objectives** (WSO) for new power reactors, on the efforts undertaken in this respect, and the challenges encountered. It would be appreciated if, inter alia, the following items are covered in the information provided:

WSO 1:

- operational safety margins;
- selection of materials.

WSO 2:

- conservative selection of assumptions for dealing with design basis accidents, beyond what is customary for Gen II plants;
- 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.

WSO 3:

- demonstration of functioning and reliability of passive safety systems and features;
- reliability of primary depressurization;
- 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;
- demonstration of practical elimination for steam explosion in the reactor pressure vessel, hydrogen detonation and other phenomena.

WSO 4:

- 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.

WSO 5:

- Demonstration of the availability of the necessary safety functions after the crash of a large airplane, taking into account the potential common vulnerability of the safety trains;
- demonstration that buildings or parts of buildings containing nuclear fuel and housing key safety functions are designed to prevent airplane fuel from entering.

Another issue of interest would be a detailed discussion of the application of **lessons learned from Fukushima** for the reactor type VVER-1200/V491.

Chapter 6 "Site Evaluation incl. external Hazards"

Recommendations:

- Altogether, it is questionable that the envisaged site elevation will ensure sufficient protection against **external flooding** caused by the extreme sea water level and waves. It is recommended to consider the **implementation of appropriate further protection** of the plant site.
- Regarding the existing sea-related hazards that could cause the loss of the heat sink (e.g. extreme low or high sea water level; clogging caused due biological fouling or frazil ice) - taking into account the statement of WENRA (2013) - it is recommended to consider the **implementation of an alternative heat sink** (for example a groundwater well).

It is recommended to perform a **systematic consideration of all possible combinations of natural phenomena**. In this context, even "minor" phenomena should be characterized, because several "minor" phenomena can lead to an "important" external hazard.

Request for information:

It would be appreciated if information pertinent to the following topics could be provided once available:

- Evaluation of the design basis flood (DBF) and of cliff-edge effects in case of a beyond design basis flood

- Evaluation of natural events that can cause clogging of the water intake (in particular biological fouling and frazil ice formations) and protection against those events
- Evaluation of the design basis earthquake (DBE), and of cliff-edge effects of the plant in case of a beyond DBE
- Evaluation of safety margins of the nuclear power plant against natural hazards
- Systematic consideration of all possible combinations of natural phenomena
- Degree of confidence of the return frequency of natural phenomena
- Demonstration of the resistance of the plant against a crash of a large commercial airplane (including the impact of the resulting fuel fire)

Chapter 7 "Accident Analysis and transboundary Impact"

Recommendation:

It is recommended to perform a conservative worst-case release scenario which is based on specific accident analysis of the AES-2006/V-491 once this information is available.

Request for information:

It would be appreciated if **information** pertinent to severe accident scenarios with source terms, timing and duration of the release, calculated frequency of occurrence (including uncertainties) could be provided once available.

Chapter 8 "Radioactive Waste Management"

Request for information:

It would be appreciated if information pertinent to the following topics could be provided once available:

Spent Fuel

- Only a rough estimate of the quantity of spent fuel is made in the EIA-Report. Data on the expected quantities of spent fuel need to be more concrete.
- Fennovoima needs to present the planned type of interim storage for SNF (wet or dry storage), its capacity and the schedule of the construction works.
- In the EIA-Report, the stated interim storage time of SNF is a minimum of 40 years. As the duration of interim storage is important for the evaluation of the risk, concrete information need to be provided.
- The decision about the final disposal strategy of SNF is of interest from the Austrian point of view. In case Fennovoima has to construct its own final disposal facility, a time schedule as well as information on the sites envisaged should be provided and the progress and timetable of Fennovoima's EIA on SNF disposal should be made available.

LILW

- More information on the LILW waste treatment plants and on the geological suitability of the on-site LILW repository should be given.

10 SUMMARY OF QUESTIONS

The following questions should be answered within the EIA procedure:

Chapter 6 "Site Evaluation incl. external Hazards"

- *Can the determination of the site elevation including safety margins and its justification regarding the sea level variation, wave heights and the respective uncertainties be explained?*
- *Would the implementation of an alternative heat sink (e.g. a ground water well) be possible at the site? Has the implementation of an alternative heat sink, which is independent of the sea water, been considered?*

Chapter 7 "Accident Analysis and transboundary Impact"

- *Can you provide the interpolated results of the Cs-137 ground deposition in case of the considered INES 7 accident at the distance of 1,850 km from the Hanhikivi site (distance to the Austrian border)?*
- *Is it possible to perform a dispersion calculation of the considered INES 7 accident with a release time (1 hour) which corresponds to a conservative worst-case release scenario?*

Chapter 8 "Radioactive Waste Management"

- *When will the choice of interim storage be made? Is there a currently favored option?*
- *When can the decision about the final disposal strategy of spent fuel be made available?*
- *In case Fennovoima has to construct its own final disposal facility: (When) can the progress and timetable of Fennovoima's EIA on SNF disposal be made available?*

11 LIST OF TABLES

12 LIST OF FIGURES

13 REFERENCES

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14 GLOSSARY

ABWR .... Advanced Boiling Water Reactor

AC .... Alternating Current

App. .... Appendix

ASUNE............... Act on Safe Use of Nuclear Energy

ATWS................. Anticipated Transients Without Scram

BDBA .... Beyond Design Basis Accident

Bq....................... Becquerel

CCF.................... Common Cause Failure

CDF.................... Core Damage Frequency

Chap. .... Chapter

Cs....................... Cesium

CWS................... Cooling Water System

DBA.................... Design Basic Accident

DBE.................... Design Base Earthquake

DC...................... Direct Current

DEC .... Design Extension Conditions

DiD .... Defense-in-Depth

ECCS .... Emergency Core Cooling System

ECR .... Emergency Control Room

EFWS................. Emergency Feedwater System

EIA .... Environmental Impact Assessment

EPR.................... Evolutionary Power Reactor

EUR .... European Utility Requirements

FSAR .... Final Safety Analysis Report

g .... Acceleration of free fall

GIC..................... Geomagnetically Induced Currents

GRS .... Gesellschaft für Anlagen und Reaktorsicherheit mbH

hrs .... hours

I .... Iodine

I&C .... Instrumentation and Control

IAEA................... International Atomic Energy Agency

ICRP……………….International Commission on Radiological Protection

INES................... International Nuclear and Radiological Event Scale

INPRO................ International Project on Innovative Nuclear Reactors and Fuel Cycle

IRS .... International Reporting System for operating Experience

IRSN .... Institut de Sûreté Nucléaire et de Radioprotection

IRWST................ In-Containment Refueling Water Storage Tank

LB LOCA ....Large Break, Loss of Coolant Accident
LBB principle ......Leak Before Break Prnciple
LILW ....Low and Intermediate Level Waste
LLW ....Low Level Waste
LRF....................Large Release Frequency
mGy...................milli-Gray
mSv ....Milli-Sievert
MW ....Megawatt
NEA ....Nuclear Energy Agency
NPP ....Nuclear Power Plant
OECD ....Organisation for Economic Co-operation and Development
PGA...................Peak Ground Acceleration
PHRS ....Passive heat removal system
PSA ....Probabilistic Safety Analysis
PSAR.................Preliminary Safety Analysis Report
PWR ....Pressurized Water Reactor
RAW ....Radioactive Waste
RHWG ....Reactor Harmonization Working Group
RPV ....Reactor Pressure Vessel
SAM...................Severe Accident Management
sec.....................seconds
SFP ....Spent Fuel Pool
SG ....Steam Generator
SGTR ....Steam Generator Tube Rupture
SMHI ....Swedish Meteorological Institute
SNF ....Spent Nuclear Fuel
SSC ....Structure, Systems and Components
SSK ....Strahlenschutzkommission
STUK.................Finnish Nuclear Regulatory Authority
SWR ....Siedewasserreaktor (boiling water reactor)
TBq....................Tera-Becquerel
VLLW.................Very Low Level Waste
VVER.................Voda Voda Energo Reactor, Russian reactor type
WENRA ....Western European Nuclear Regulators Association
WSO..................WENRA Safety Objectives
yr ....year(s)
YVL....................Finnish Regulatory Guide on Nuclear Safety

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