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Addressing off-site consequence criteria using Level 3 PSA

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Abstract

The goal of this project is to further Nordic understanding of the potential for Level 3 PSA to determine the influences and impacts of off-site consequences, the effectiveness of off-site emergency response, and the potential contributions of improved upstream Level 1 and Level 2 PSAs.
This status report summarizes the work performed during 2014, which mainly belongs to Task 4, the Pilot Application, which is separated into two parallel activities, the "Swedish" and "Finnish" Pilot projects.
The Finnish Pilot study demonstrates the application of deterministic and probabilistic methods in Level 3 PSA. The case considered applies the source term of the actual Fukushima Daiichi nuclear power plant accident, but without taking into consideration that the population was evacuated before the accident due to the tsunami
The population doses are analysed in the event trees, and uncertainty analyses are conducted on the weather variables, evacuation and sheltering success probabilities, and the effectiveness of sheltering. We find that, even under rather conservative assumptions, the radiological consequences are small. However, the results should be seen as only indicative due to simplifications made in modelling.
For the Swedish Pilot study, two major activities were completed during 2014: The Input Specification and the Scope of Analysis.
The input specification includes a discussion on the input data that will be used in this study. Here some justification for why the EPR reactor would be the focus of the source term development for the Swedish pilot project. Further input data, such as weather and population data, will be extracted from a Thesis work.
The proposed analysis scope will help the project to ultimately provide important insights related to the main project goals.
During 2015, the final year of the project, the pilot projects will be completed and the guidance document will be formulated along with the project stake holders. The working group will also remain engaged in international activities surrounding Level 3 PSA, the development of the IAEA Level 3 PSA TECDOC and the ANS/ASME Level 3 PSA Standard.

Key words

PSA, PRA, Level 3 PSA, Probabilistic Consequence Analysis, Nuclear Power Plant Safety

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Phase 2 Status Report from the NKS-R L3PSA

(Contract: AFT/NKS-R(14)109/11)

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1. Introduction

Level 3 Probabilistic Safety Analysis (Level 3 PSA) provides a probabilistic assessment of off-site consequences from radioactive releases. The input to a standard Level 3 PSA is derived from several sources. The results from the identification and assessment of the accident sequences leading to core damages, which are provided by Level 1 PSA, and the severe accidents and radioactive source term analyses, which are provided by Level 2 PSA, are combined with meteorological, population and agricultural data to estimate the off-site societal, environmental, and economic risks posed by a nuclear facility.

The typical outputs of a Level 3 PSA can vary, but often include collective radioactive doses, health effects (e.g. early fatalities, latent cancers), economic impacts, and agricultural effects. Interest and activities in Level 3 PSA have increased recently for several reasons. The primary reason for the increased interest in Level 3 PSA is to better understand and characterize off-site consequences following the findings from the Fukushima accident, the obligations utilities have from insurance companies and shareholders, and the obligations regulators have to the public's health and safety.

The potential insights that could be gained through Level 3 PSA may assist utilities with operating plants, utilities pursuing new construction, regulatory bodies, public health organizations, and emergency preparedness networks. Therefore, as a structured study of Level 3 PSA, this project seeks to determine the requirements and overall utility of such an analysis. During the project there has been close interaction with utilities, regulators, and insurers which have been able to guide and influence the project execution through participation in project planning, meetings, and seminars.

1.1. Purpose

Interest in Level 3 off-site consequence PSA has risen within the Nordic region, and around the world as a consequence of the Fukushima accidents and the continuing interest in new reactors.

This interest has been reflected in the volume of recent activity in the area of Level 3 PSA at the International Atomic Energy Agency (IAEA) and ongoing projects in the United States, the Netherlands, South Africa, Japan, and elsewhere.

The goal of this study is to further Nordic understanding of the potential for Level 3 PSA to determine the influences and impacts of off-site consequences, the effectiveness of off-site emergency response, and the potential contributions of improved upstream Level 1 and Level 2 PSAs. Level 3 PSA provides a tool to assess the risks to society posed by a nuclear plant, and could be integral in making objective decisions related to the off-site risks of nuclear facilities.

1.2. Scope of project

The project will develop guidance on several significant topics. The reports and seminars will include guidance on the following topics:

1. A summary of the industrial purpose for performing Level 3 PSA
2. Recommended risk metrics for Level 3 PSA
3. Requirements on existing Level 1 & Level 2 studies set by the Level 3 PSA analysis.
4. Insights on abilities of existing Level 3 PSA tools/codes and possible needs for further development.
5. Collection of current regulations, guides and standards toward Level 3 PSA
6. Guidance document

1.3. Project organization

The project includes separate tasks that are being conducted in parallel. Several of these tasks started during 2013, while others will start-up in 2014 and will be finalized in 2015. The project tasks address the following topics:

(0) Industry and Literature Survey
(1) Appropriate Risk Metrics,
(2) Regulation, guides and standards,
(3) Development of a Guidance document
(4) Pilot Application including tools for dispersion and consequence analysis

1.4. Project interfaces

The project has had significant interaction with Nordic utilities and regulatory authorities. These include a Stakeholder Meeting where the project financiers provided input on the scope and direction of the project and the Task 0 survey. The stakeholders also responded to the questionnaire that was developed in Task 0, and then assisted in drawing conclusions from the questionnaire during a "Questionnaire Response Workshop". Finally, the working group held the second year project seminar on January 20th, 2015 to summarize the progress during the second year of the project and to receive input on the pathway forward for the project.

The project has created interest in many international organizations and has fostered Nordic participation in several international Level 3 PSA activities. Currently, the IAEA is developing Level 3 PSA guidance through the drafting of a TECDOC. This project has allowed the working group to contribute to this effort through member participation in IAEA Technical Meeting & Consultant Meetings as well as an expert lecturer an IAEA Regional Workshop on Level 3 PSA. The project has also interfaced with groups such as OECD/NEA Working Group RISK and the ANS/ASME Level 3 PSA standard writing committee.

1.5. Report contents

This report describes the developments the working group has made during the calendar year, 2014. The following sections summarize the work performed under each of the separate Tasks which were performed during 2014 (outlined in Section 1.3). For further information full task reports will be written, describing more completely the work completed for each respective task.

1.6. Acknowledgements

The working group in this project would like to acknowledge the funding organizations that stand behind this project. Funders are found in several organizations such as the Nordic Nuclear Safety Research group (NKS) and the Nordic PSA Group. NPSAG is represented by the Swedish utilities Forsmark (FKA), Ringhals (RAB) and Oskarshamn (OKG) and the Swedish Radiation Safety Authority (SSM). Funding is also provided by and the Finnish Research Programme on Nuclear Power Plant Safety (SAFIR2014). NKS conveys its gratitude to all organizations and persons who by means of financial support or contributions in kind have made the work presented in this project possible.

1.7. Disclaimer

The views expressed in this document remain the responsibility of the author(s) and do not necessarily reflect those of NKS. In particular, neither NKS nor any other organisation or body supporting NKS activities can be held responsible for the material presented in this report.

2. Finnish Pilot Study

The pilot project is separated into two parallel activities. The "Swedish" and "Finnish". Pilot projects. This section details the developments of the Finnish Pilot Study during 2014.

2.1. Goal of Finnish Pilot Study

This report presents a pilot study in level 3 PSA.

The main goal of the pilot is to study how to apply the IDPSA methodology on level 3 PSA.

There are also other goals:

1. To illustrate how to apply a particular risk measure on level 3, namely the number of cancers resulting from a radioactive release.
2. To enable comparison to the Swedish method of conducting level 3 PSA.
3. Facilitate level 3 PSA software development. It is hoped that the construction of the pilot reveals targets of development in the SPSA software, and provide experience of Level 3 analyses needed in level 3 software development.

2.2. Description of pilot case

The Finnish Pilot project is an exercise in alternate history, and seeks to answer the question; what would the consequences of the Fukushima Daiichi nuclear accident have been if a similar accident, with the source term of the actual accident of March 2011, had happened so that the population had not been decimated by the tsunami and evacuated after that, but instead had been in their places, and evacuated only after the nuclear accident.

The motivation for the case study comes from the fact that the Fukushima Daiichi accident had very small radiological consequences: it has been estimated that the radioactive release will produce no extra deaths in the general public [1], and probably none even in plant and rescue workers. On the other hand, in the first few days of the release, wind blew dominantly to the Pacific Ocean, thus saving the population from exposure. Therefore it is of interest to find out whether the near nonexistence of radiological consequences was due to good luck and the deflation of the nearby areas from population after the tsunami, or was it to be expected given the weather conditions in Japan and the efficiency of the evacuation within the evacuation zone.

We assume that the release would have been much more abrupt than it was (in reality there were multiple releases over several months). We assume that the whole release would have happened in three hours. As the source term, we use the actual source term of Fukushima. Assuming such a short release time span is conservative, but can be justified on the basis that much of the release at Chernobyl happened in a few hours, and the source term there was an order of magnitude bigger than that in Fukushima Daiichi.

Table 1. The source term of the Fukushima Daiichi nuclear power plant accident [1]

Radionuclide	Total release (PBq) to the atmosphere
Te-132	29
I-131	120
I-132	29
I-133	9.6
Xe-133	7300
Cs-134	9.0
Cs-136	1.8
Cs-137	8.8

The evacuation proceeded in Fukushima as follows:

Table 2. Evacuation-related events in the Fukushima prefecture, March 2011 [1]

Event	Date	Time
earthquake	11.3.2011	14:46
tsunami	11.3.2011	15:35
evacuation within 2 km ordered	11.3.2011	20:50
evacuation within 10 km ordered	12.3.2011	5:44
evacuation within 20 km ordered	12.3.2011	18:25
sheltering within 30 km, evacuation within 20 km completed	15.3.2011	11:00

We consider population doses (and from that, the theoretical number of cancers as 0.05 x population dose) in five cities closest to the Fukushima Daiichi NPP site. The cities are given in Table 3 (population data are from Wikipedia).

Table 3. Cities considered in health consequence calculations.

City	Direction from Fukushima Daiichi	Distance from Fukushima Daiichi [km]	Population
Minamisoma	north	27	71 000
Kakuda	north	58	31 000
Fukushima	northwest	64	294 000
Koriyama	west	56	338 000
Iwaki	south southwest	48	345 000

2.3. Limitations

This study is a demonstration of how the integrated deterministic and probabilistic safety analysis (IDPSA) framework may be applied to level 3 PSA studies. The case chosen is the following:

What could the radiation doses to the population have been if the Fukushima Daiichi nuclear accident…

- would have taken place in the weather circumstances generally prevailing in the Fukushima province in March (and not the particular weather of March 2011)

- assuming the evacuation of the surroundings would not have proceeded as it actually did,

- with a release far more rapid than the actual –

This study, therefore, does not reflect what actually happened in Fukushima, March 2011. The results obtained should be seen as indicative and not as reliable estimates of release consequences even in this alternative scenario.

2.4. The model

The general architecture of the model is as follows. The deterministic part covers atmospheric dispersion and population dose calculation in given weather conditions, which was implemented in ARANO [2], VTT's consequence analysis code. The probabilistic part covers the assessment of the probabilities of various consequences, and incorporates the probabilities of different weather conditions, and evacuation and sheltering success probabilities. The probabilistic part is modelled by an event tree; the population dose resulting from each sequence in the event tree was calculated in ARANO. The probabilistic part was implemented in SPSA, VTT's code for level 2 PSA.

The number of cancer deaths caused by the ionizing radiation of the release was calculated from population dose as 0.05 times the population dose (manSv). This is the estimate used generally.

In the rest of the sections of this chapter, the event tree, weather model, evacuation, and shielding models are presented

2.4.1. Event tree model

The event tree model includes five sections:

- Wind speed: 16 m/s, 8 m/s or 0 m/s
- Wind direction: northwest, west, north, south southwest or other
- Precipitation: 5 mm/hour or 0 mm/hour
- Population sheltering: in time or not
- Evacuation: in time or not (for north direction, only Kakuda might be evacuated)

For each end point of the event tree without evacuation and sheltering, the population dose was calculated by ARANO software. For the end points with sheltering but without evacuation, the population doses obtained from ARANO were multiplied by a 'sheltering factor' (see section 2.4.3). For the end points with evacuation or wind direction 'other', the population dose was assumed to be 0. The population dose from the release was assumed to be 0, when the wind speed was under 4 m/s. The justification for this is that 1) a mild wind does not carry the radionuclides far from the site, and 2) there will be plenty of time for evacuation, and therefore the cities in the direction of the wind would be void of people when the radioactive plume would finally arrive.

2.4.2. Weather data

The weather data used has been collected from a variety of sources.

The wind speed statistics are from Onahama, which is in the Fukushima prefecture, some 60 kilometers to the south of the Fukushima Daiichi nuclear power plant site. http://www.windfinder.com/windstatistics/onahama

The site contains wind direction distributions in the form of Rose diagrams for each month, and also on the yearly level. We used the distribution for March, for reasons stated in the description of the problem.

The wind direction statistics were obtained from the Rose diagram of directions on the web page, and are approximately as follows:

Table 4. Average wind direction distribution in Onahama, Fukushima prefecture, Japan in March. Only the directions that point to land from Fukushima Daiichi are shown.

Wind direction (from)	Wind direction (to)	Approximate proportion, %
North	South	7.6
North northeast	South southwest	9.7
Northeast	Southwest	5.9
East northeast	West southwest	2.7
East	West	2.7
East southeast	West northwest	2.7
Southeast	Northwest	3.8
South southeast	North northwest	6.5
South	North	11.4

The wind speeds considered in the deterministic analyses were 0, 8 and 16 m/s. These wind speeds each represent a range of actual wind speeds in the model. It was decided that wind speed 0 m/s represents actual wind speeds of 0-4 m/s, wind speed 8 m/s represents actual wind speeds of 4-12 m/s, and 16 m/s presents any wind speed over 12 m/s.

The distribution of wind speeds was postulated as a log-normal distribution. From the probabilities of wind speed intervals were determined as shown in Table 5.

Table 5. Probabilities of wind speed ranges in Onahama, Fukushima prefecture, Japan, from the postulated log-normal model

Wind speed used in deterministic calculations [m/s]	Wind speed interval [m/s]	Probability of wind speed interval from postulated log-normal distribution

0	0-4	0.73
8	4-12	0.19
16	12+	0.07

Precipitation statistics for Onahama were obtained from:

http://www.yr.no/place/Japan/Fukushima/Onahama/statistics.html.

The average number of days with precipitation for March is 8, and therefore the probability of rain at the time of the release was set to 8/31≈0.258. It was assumed that the amount of rainfall (if it rains) concentrates on the value used, namely 5 mm/hour. It is evident that a more sophisticated analysis would take into account the probability distribution of rainfall, and even its dynamic nature.

2.4.3. Evacuation and shielding models

We define evacuation success to mean that evacuation has been completed before the release plume arrives.

If the release plume arrives before the population has been evacuated or sheltered, the population is assumed to be outdoors 10% of the time and indoors 90% of the time.

Evacuation success probability is calculated as follows. With a given wind speed v and given distance x from the site, it takes t1=x/v seconds for the plume to reach the city. This time is compared to the time it took to empty the evacuation zone in the Fukushima prefecture from people in March 2011. The time it takes for the plume to arrive from the site to the city is divided by the time it takes to evacuate the city, and this ratio is taken as the evacuation success probability (if it takes more time for the plume to reach the city than the evacuation in Fukushima in 2011, the evacuation is considered a success with probability 1).

If such an acute and large release as postulated in this report would have actually happened, it is natural to assume that evacuation would have been ordered at the latest when the release started. As seen in Table 2, the evacuation of the 20 kilometer zone in Fukushima was ordered on 12.3.2011 at 18:25, and was completed on 15.3.2011 at 11:00. Thus it took 2 days, 16 hours and 35 minutes, or 232 500 seconds. In the calculations, this reference evacuation time is rounded to 3 days.

In the calculations, it is assumed that the population is 10% of the time outdoors. Considering this sheltering does not decrease the population dose much. It is assumed that with sheltering the population dose is 70% of the population dose without sheltering. The probability of sheltering is set to 0.8 by expert’s judgement.

2.5. Results

The expected number of cancer deaths was 16. Table 6 presents the results for different wind directions and other conditions. Figure 1 presents Farmer’s curve representing the probability for having at least considered number of cancer deaths. With probability 0.927 there are no cancer deaths at all.

Table 6. Results of the event tree calculations

Conditions	Northwest	West	North	South southwest
Wind 8 m/s, no rain, no sheltering	Prob = 1.1E-3	Prob = 7.7E-4	Prob = 3.3E-3	Prob = 2.8E-3
	Cancers = 220	Cancers = 320	Cancers = 210	Cancers = 410
Wind 8 m/s, no rain, sheltering	Prob = 4.3E-3	Prob = 3.1E-3	Prob = 1.3E-2	Prob = 1.1E-2
	Cancers = 150	Cancers = 220	Cancers = 150	Cancers = 290
Wind 8 m/s, rain, no sheltering	Prob = 3.7E-4	Prob = 2.7E-4	Prob = 1.1E-3	Prob = 9.6E-4
	Cancers = 180	Cancers = 270	Cancers = 190	Cancers = 350
Wind 8 m/s, rain, sheltering	Prob = 1.5E-3	Prob = 1.1E-3	Prob = 4.6E-3	Prob = 3.8E-3
	Cancers = 120	Cancers = 190	Cancers = 140	Cancers = 240
Wind 16 m/s, no rain, no sheltering	Prob = 3.9E-4	Prob = 2.8E-4	Prob = 1.2E-3	Prob = 1.0E-3
	Cancers = 220	Cancers = 320	Cancers = 210	Cancers = 410
Wind 16 m/s, no rain, sheltering	Prob = 1.6E-3	Prob = 1.1E-3	Prob = 4.7E-3	Prob = 4.0E-3
	Cancers = 150	Cancers = 220	Cancers = 150	Cancers = 290
Wind 16 m/s, rain, no sheltering	Prob = 1.4E-4	Prob = 9.6E-5	Prob = 4.1E-4	Prob = 3.5E-4
	Cancers = 180	Cancers = 270	Cancers = 190	Cancers = 350
Wind 16 m/s, rain, sheltering	Prob = 5.4E-4	Prob = 3.9E-4	Prob = 1.6E-3	Prob = 1.4E-3
	Cancers = 120	Cancers = 190	Cancers = 140	Cancers = 240
Total	Prob = 1.0E-2	Prob = 7.2E-3	Prob = 3.0E-2	Prob = 2.6E-2
	Cancers = 150	Cancers = 230	Cancers = 160	Cancers = 300
Expected cancers	1.5	1.7	4.8	7.8

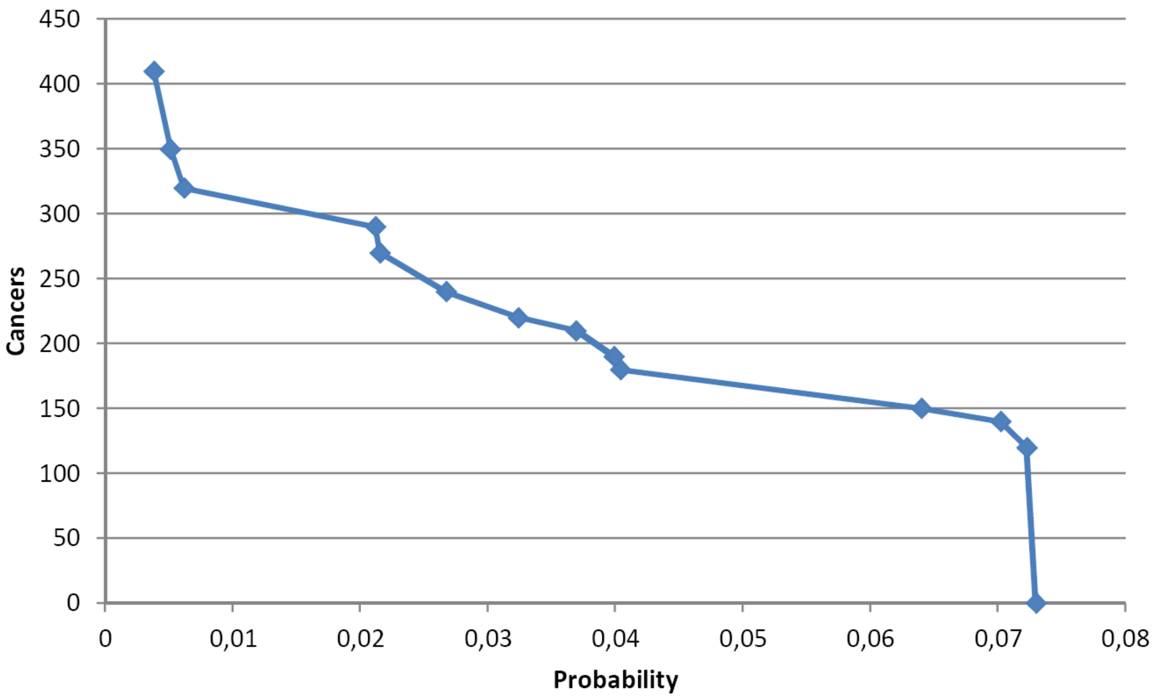

Figure 1. Farmer's curve

The sensitivity of the expected number of cancer deaths to the evacuation probabilities was also studied. The chosen evacuation probabilities in this study were so small (< 0.05) that the results were almost same as when assuming evacuations impossible. However, choosing larger evacuation probabilities reduced the expected number of cancers. When evacuation probabilities were multiplied by 10, the expected number of cancers was 13. When evacuation probabilities were multiplied by 20, the expected number of cancers was 9.8. When evacuation probabilities were multiplied by 30, the expected number of cancers was 6.8. When evacuation probabilities were set close to 1, the expected number of cancers was close to 0.

2.6. Conclusions

We have modelled and analyzed a case of alternative history – what would have happened if the source term of the Fukushima Daiichi NPP accident would have been released rapidly and the population of the big towns near the NPP site would have been in place (instead of evacuated or killed by the tsunami), under weather conditions in that part of Japan in March – in order to assess what the radiological consequences would have been in terms of cancer deaths.

The overall number of cancer deaths resulting from the release is very low considering the number of people in the area. There were approximately 1 079 000 inhabitants in the cities considered in March 2011 prior to the earthquake and the tsunami. The expected number, given by our model, of cancer deaths resulting from the release is 16, with very high probability (0.927) there will be no cancer deaths, and the maximum expected number of cancer deaths under the most adverse conditions is 410. Even the largest number of cancer deaths due to the release is well below what can be detected as an increase in a population of that size when random fluctuations in cancer deaths is taken into account. Approximately 1/5 of the population will die of cancer due to reasons not related to the radioactive release; in the case of the towns considered, this amount to 216 000 cancer deaths.

The chosen methodology – using an event tree model for probabilistic considerations, and calculating atmospheric dispersion and population dose deterministically – seems to be fit for the purpose of level 3 PSA analyses. It makes the heavy computational load of atmospheric dispersion calculations manageable, while at the same time it provides the benefits of probabilistic analysis in terms of uncertainty handling (probability distributions). The size of the event tree will remain moderate even if a more detailed model is constructed, and the parameters needed in the model can either be calculated from weather data, or – in the case of countermeasure (evacuation, sheltering) success probabilities – be estimated from evacuation models or be assessed by expert judgment.

The model developed is rather coarse and can be considered to give indicative results at best. There are several ways in which to improve the model's accuracy. Concerning the modeling of weather, wind direction cannot be changed in ARANO (wind direction remains the same during the release and atmospheric dispersion); however, some codes, such as CALPUFF, are freely available that can handle dynamic weather conditions during the atmospheric dispersion. In these codes, also precipitation can be modelled in a more accurate way.

The actual release of Fukushima might be modelled more accurately in other ways, too. The release took place over an extended period of time (several months, with small releases even after that), and varied in both intensity and isotope content. This could be modelled by several releases that could follow a stochastic process in the model.

Evacuation has been taken into account in our model in a rudimentary manner that does not take into account the size of the population to be evacuated, the existence (or not) of evacuation plans, the quality of official actions in conducting the evacuation, possibly adverse weather and other conditions, the risks involved in evacuation etc. More refined evacuation models might shed light on the effects of these factors.

Due to practical reasons, a comprehensive sensitivity and uncertainty analysis, covering both the deterministic and probabilistic parts of the model, was not possible. It is evident that a comprehensive uncertainty analysis would yield valuable information about uncertainties.

3. Swedish Pilot Study

3.1. Swedish Pilot Project Plan

At the beginning of the Swedish Level 3 PSA Pilot project a long list of project goals were developed. These goals were used to develop the general plan for the project. The goals were also used to develop the scope of analysis of the project.

In order to organize the project deliverables and promote cooperation between the many organizations participating in the project a group of project reports were also developed. An overview of the project plan is provided in this section.

3.1.1. Project Goals

The main project goals identified are the following:

1. Cover which types of insights can be attained from a Level 3 PSA

    a. Discrimination of consequences which exceed a regulatory risk threshold, eg released activity, marginally or substantially.
    b. Seek to establish to which extent Level 2 PSA output may be relevant as a surrogate for Level 3 PSA insights.
2. Indicate resources required for performing a Level 3 PSA
3. Identify any key uncertainties in the analysis
4. Indicate how existing plant Level 2 PSA structure would interface with a Level 3 PSA analysis
5. Gain insights into the use of Level 3 PSA risk metrics:
    a. Health effects: Collective dose (Latent Cancers)
    b. Environmental effects: Contaminated area (Economic impact)
    c. Impact of Countermeasures/protective actions (Severe Accident Scenario Warning Time)

The features given under Level 3 PSA risk metrics in parenthesis indicate potentially useful derived metrics or important underlying characteristics. In particular, for the case with countermeasures it is essential that applicable severe accident sequences are allocated an appropriate warning time as only sequences with adequate time for countermeasures to be implemented will be affected by countermeasures.

3.1.2. Project reports

The project has been broken up into separate reports. The reasoning for producing several different reports for the major phases of the work is to allow the large group of stakeholders and working group members to collaborate throughout the work. All members will be able to review and provide comments for the subsequent reports. The reports will form the basis for the description of the study in the final project report and will also provide input in the development of a guidance document.

The five project reports that will be produced during the Swedish Pilot Project are the following:

1. Pilot Project Plan
2. Input Specification Report
3. Scope of Analysis Report
4. Methodology Report
5. Application and Result Interpretation

A brief description of the scope of each of these reports is discussed in the following sections.

3.1.3. Pilot Project Plan

The pilot project plan report represents the first in series of reports that will be released detailing the Swedish Level 3 PSA Pilot Project. This report outlines the purpose, the goals of the project, and the phases/reports that will be developed during the work.

3.1.4. Input Specification Report

The input specification report will specify the possible inputs for a Level 3 PSA study, provide additional discussion on those that are likely to be incorporated into the study (based upon available references/resources etc.), and to discuss the formats of inputs that may be used in the analysis. Details from the input specification report are presented in Section 3.2.

3.1.5. Scope of Analysis Report

The Scope of Analysis report will describe how the project intends to help satisfy the project goals provided in Section 3.1.1. The report will describe how the input data described in the Input Specification Report will be selected, the output data and corresponding risk metrics that will be assessed. Details from the scope pf analysis report are presented in Section 3.3.

3.1.6. Methodology Specification

The Methodology Specification report will outline the methods that are employed in the Pilot Project. The report will detail the models and assumptions that are used by the software that is used in the analysis.

3.1.7. Application and result interpretation

The final report in the Swedish Pilot study will be the Application and result interpretation report. This report will describe the result of the study, as well as the implications of these results. Potential uses for the results, uncertainties, and areas of improvement will also be identified in this report.

3.2. Input Specification

The second report developed during the Swedish Pilot Study is an Input Specification. The input specification portion of the project specifies the possible inputs for a Level 3 PSA study, provide additional discussion on those that are likely to be incorporated into the study (based upon available references/resources etc.), and to discuss the formats of inputs that may be used in the analysis.

3.2.1. Input data requirements for Level 3 PSA

The Swedish project is based on the LENA software. A description of the software, its capabilities, and its methods will be further discussed in two project reports that will be produced during the 2015 year: the Scope of Analysis Report, and the Methodology Specification Report.

The requirements of a Level 3 PSA input are not currently "standardized". Since the term "Level 3 PSA" is used rather broadly, for a wide spectrum of different analyses, the input requirements for Level 3 PSA analyses vary significantly. There is, however, a joint effort between the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) to develop a Level 3 PSA "standard." This standard is being developed in the manner of previous ANS/ASME PSA Standards, which define technical elements and qualitative definitions for levels of compliance. Some of the concepts provided by this standard have been used to inspire the discussion in this section.

There are three inputs of a Level 3 PSA, which are largely universal: the source term, the weather information, and land-usage / population information. It is possible that the population aspect may not be absolutely necessary if the offsite consequences being studied are not person-dose/ health effect related, e.g. contamination calculations. These inputs and their subsequent constituents are further described in this section. The focus of this discussion will also include a short discussion on the current level of detail of the state-of-practice for each of these inputs.

3.2.1.1. Source term

In the IAEA safety series document No. 53 titled, "Derivation of the Source Term and Analysis of the Radiological Consequences of Research Reactor Accidents" the following definition of a source term is provided:

> *"The source term is defined as the magnitude, composition, form (physical and chemical) and mode of release (puff, intermittent or continuous) of radioactive elements (fission and/or activation products) released during a reactor accident. The mechanism, time and location of the release must also be identified." [5]*

This definition highlights that source terms include the composition of the release as well as several key parameters that affect how the release will disperse in the environment. This definition does not provide the additional consideration in a probabilistic study, which is the source term frequency.

The source term is often seen as the connection between the Levels 1 & 2 PSA and the Level 3 PSA. In some cases Level 3 PSAs are performed without significant input to an upstream Levels 1 and 2 PSA, in which case the source terms may provide the sole link between the plant response and severe accident progression and the parameters. A somewhat common methodology was developed in the 1980s and 1990s based on what types of information in source terms had the most significant impact on probabilistic consequence analysis. These practices are still largely the basis for current Level 3 PSA analyses. There has been some expansion in the level of input which Level 3 PSA programs can accommodate, albeit modest. The major aspects that the source term should include are the following concepts, which will be further explored in this section:

- Radionuclide inventory / release fraction,
- Release frequency,
- Isotopic grouping,
- Heat of release / Release height,
- Delay [h],
- Duration of release [h],
- Release fractions [%],
- Release coordinates.

3.2.1.1.1. Radionuclide inventory, release categories, and release fractions

There are several aspects that are important in the development of the composition of the radionuclide release. A common distinction that is made in the draft Level 3 PSA standard with respect to modelling capability is the difference between generic data and actual site-specific data. Furthermore it is important for the Level 3 PSA practitioner to appropriately handle and organize the release composition information.

The isotopic composition of the release is often provided as a fraction of the total radionuclide inventory. The true values of radionuclide inventory depend on the fuel loading, the cycle burn-up, and if/how the plant was shutdown. These considerations may or may not have a calculable impact based on the specifics of the severe accident sequence. Ideally, these types of considerations will be incorporated in the source term calculation. In some cases, variables such as cycle burn-up are determined conservatively, based on worst-case conditions. Such assumptions can have un-intended impact on the consequence analysis especially when preventative actions are modelled as will be discussed in Section 3.2.1.4.1 and the Methodology Specification Report.

Release categories are a commonly misunderstood concept for many outside the Probabilistic Consequence / Level 3 PSA field. Release categories are the grouping of source terms in order to simplify and consolidate the number of source terms required to perform a Level 3 PSA. It is important that the release categories accurately represent the various accident sequences which could face a facility. Developing release categories is often done based on phenomenological similarities in the accident progression. It is also important to capture the spectrum and appropriate classification (release category binning) for source terms based on parameters that affect the consequence metrics, e.g. release energy influences plume rise/dispersion.

For each of the severe accident sequences analysed in the Level 2 PSA, the nuclides in the release are defined, as mentioned previously; this is typically represented in terms of release fractions of the entire core's radionuclide inventory. These ratios are expressed for each of the isotopic groups used in the dispersion, deposition, exposure, dose calculations.

3.2.1.1.2. Isotopic grouping

For computational economy, Level 3 PSA analyses have historically been performed by grouping released radionuclides into isotopic groups. Groups are still widely used even though the computational limitations are no longer of much concern. These groups are based on isotopes that can be combined into representative groups in terms of their physical, chemical, and radioactive properties.

In LENA, these groups are designated consistently with those that were developed in the WASH-1400 Study.

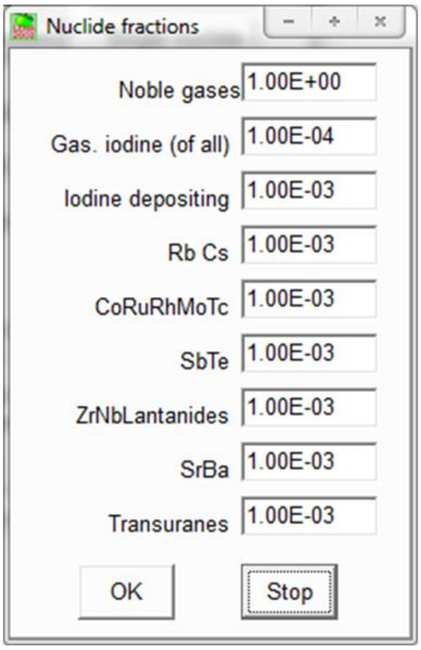

Figure 2. Isotopic fractions as shown in the LENA Graphical User Interface.

3.2.1.1.3. Release frequency

The release frequency is often developed from fault and event tree analyses for each of the release categories. These values are applied to the fractional results following the dispersion calculations, exposure, and dose evaluations.

3.2.1.1.4. Release location, height, and Release energy

The release location, height, and release energy are all very important aspects of the release that can potentially have a large effect on the dispersion calculations. The specific location of the release, may have significant implications for the atmospheric dispersion. The impact of localized effects, such as building wake effects, can have a large impact on the cross-section of the plume very near the plant. Accounting for these effects is difficult to implement in the rather simple methods employed for most probabilistic off-site consequence studies because they are very sensitive to the particular scenario and are difficult to generalize for the many unique situations calculated in probabilistic studies. For this reason, Level 3 PSA is usually not recommended for making assertions very near the release location, e.g. actions for onsite personnel, since these local effects would dominate results.

The release height and the release energy are integral for determining the plume rise. These parameters are used for calculating the effective plume height, which is the height from which the horizontal component of the dispersion calculation is based. This level is very important when determining where the plume comes into contact with the ground, which eventually influences the deposition of radionuclides as the plume diffuses.

3.2.1.1.5. Particle size

The particle size can have a significant impact on the dispersion and deposition calculations. In Level 3 PSA codes, particle size is often varied based on release category but uniform amongst each of the isotopic groups. For more recent updates to codes such as MACCs a variety of particle sizes can be separately accounted for in a single plume calculation.

3.2.1.1.6. Release timing & warning time

How time is incorporated in Level 3 PSA calculations can vary significantly depending on the capabilities of a Probabilistic Consequence Analysis (PCA) code. The release timing + warning time is important for determining the fission product levels in the nuclide inventory, and the duration of the release will also influence how plume shall be modelled. In some cases, a single continuous release over a long period of time is divided into several separate Gaussian plume or puff calculations, for short duration releases it may be appropriate to model with a single plume calculation.

The state of practice is to have data from a Severe Accident analysis code such as MAAP or MELCORE, and use the tabular results as input for the dispersion calculation(s).

3.2.1.2. Weather

In dispersion calculations the weather / environmental data requirement can vary greatly. Some advanced particle-tracking models require enormous sums of data to drive the calculation models. For Gaussian plume calculations the data requirement is quite modest. More discussion on the impact of models and methods will be placed in the methods and applications reports in this Pilot Study. This section will just describe the general requirements of weather data, and those expected to be implemented in the pilot study analysis

3.2.1.2.1. Local meteorological data

- For the simplest plume models the following data are required for the release location.
- Wind speed [m/s],
- Wind direction [degrees],
- Mixing height [m],
- Pasquill Stability class [A-F (1-6 in batch input file)]
- Precipitation [mm/h].
- The current state-of-practice in the Level 3 PSA community is to have each of these meteorological data hourly over the course of several years. These data can be sampled or used in their entirety.

3.2.1.2.2. Mixing height

The mixing height is represented by the distance between the earth's surface and the bottom of inversion aloft. Effluents released below this point tend to disperse below this level. An obvious exception is for very energetic releases where the plume heat will cause the plume rise to exceed this level.

3.2.1.2.3. Stability classification

Atmospheric stability and atmospheric turbulence is an extremely important parameter which effects dispersion. A common system for classifying atmospheric turbulence based on the meteorological conditions is using the Pasquill atmospheric stability classes.

More advanced methods for defining the atmospheric turbulence are applied in emergency preparedness activities, however it is quite common to use Gaussian plume and Gaussian plume dispersion calculations using stability classes in probabilistic analyses.

3.2.1.2.4. Precipitation

Precipitation has a major effect on the deposition of released plumes. These data should be included for each of the meteorological data time points. Depending on the complexity of the models used to perform the analysis the precipitation data can be highly detailed or less detailed. For the analysis that is planned during this work, the precipitation will be assumed for the duration of the plume calculation. Different types of precipitation, e.g. rain vs. snow, may have an impact on the deposition rates. In general this has not been incorporated in Level 3 PSAs, but may have a notable impact.

3.2.1.3. Population

Population data is usually defined in radial sectors for distances that are applicable for the methods being employed. For Gaussian plume dispersion calculations, this is usually 8-64 radial sectors from 1 km to a few 100s km.

3.2.1.3.1. Population cohorts

Populations are not monolithic and the effects of radiation exposure to children, adults, and the elderly have clearly differing societal consequences. By including models to incorporate the difference in how radioactive releases affect different population demographics valuable consequence and protective action decisions may be applied. The modelling of population cohorts is been reserved for very large scope Level 3 PSAs and will probably not be possible in this limited pilot study.

3.2.1.4. Other inputs

The complexity and focus of probabilistic consequence analyses have varied. In many studies the impacts of protective actions, shielding, and other countermeasures have been a major focal point. Furthermore, for assessing economic consequences and land and water contamination additional input information is required to determine these results. It is expected that some, limited, additional consequences shall be assessed in this pilot study work.

3.2.1.4.1. Protective action modelling

A variety of possible countermeasures or protective actions may be taken following an accidental release to reduce exposure of human populations to the radioactivity released in the accident. Protective action modelling can be quite extensive and also have some very surprising consequences when incorporated into an analysis. Some things that are commonly incorporated in probabilistic off-site consequence studies are distribution of iodine tablets, shielding/sheltering considerations and evacuations. These can be employed using various methods of varying complexity.

Protective actions are often separated into two categories depending upon the time at which they are implemented and the effects which they are designed to mitigate. Short term protective actions (emergency response) are implemented either before or shortly after a release to the environment. The objective of such measures is to limit deterministic effects and minimizing risks of stochastic effects. Long term countermeasures are designed to reduce chronic exposure to radiation, both externally from deposited material and internally from ingestion of contaminated food, with the intention of reducing the incidence of late health effects.

One element of protective action modelling is how the notion of "conservative" assumptions may influence how protective actions are implemented. If releases are over-estimated more severe protective actions may be implemented, which will affect results, possibly inaccurately. The effectiveness of countermeasures, such as stable iodine tablets is another common point of interest in off-site consequence studies. Events like Fukushima have highlighted how difficult it is to accurately model such phenomena as they function in practice. Usually, the implementation and input requirement for such countermeasure modelling depends largely on the methods, which may be quite coarse.

3.2.1.4.2. Economic

Economic models can vary significantly in complexity and widely in the input requirements. The methods of performing economic analysis in Level 3 PSA are currently evolving due to the enormous expenditures seen in the wake of the Fukushima disaster. The current state-of-practice for the economic impacts, as often applied in Level 3 PSA studies, are discussed in the OECD-NEA report published in 2000 [14].

3.2.2. Input sources

The utilities participating in the Nordic PSA Group unanimously decided that generic source terms derived from literature sources would be sufficient to draw conclusions. This can be done practically, but has some unfortunate consequences, some of which are listed below:

- Researching available source terms in literature has added an additional burden on the projects resources of finding and comparing somewhat complete source terms in literature.
- Access to "raw data" may not exist, and source terms from literature may be incomplete
- The practitioners may gain little or no experience "pairing down" information from more "complete" Level 2 PSA and severe accident sequence data and models.
- Making the methodology and coupling to upstream Level 1 and Level 2 analyses performed in this work potentially less relevant to possible future analyses (may require future rework).
- The working group may make assumptions / guesses which could be erroneous.
- Potentially less experience will be gained in what is currently lacking and should be expanded in Level 1 and Level 2 PSA

There are, of course, potential up-sides to exploring source terms from literature. For example, results using publicly available source terms should be freely publishable, additional information may be available for source terms from literature that may not be readily available from NPSAG member's current Level 1/2 PSAs, Source terms defined in literature will also provide some commentary as to why certain elements are included/excluded, which may serve as a literature survey of sorts.

This section will discuss some of the potential sources of information that could be used as a basis for this study.

3.2.2.1. Potential source term input sources / references

Two potential candidates for source terms from literature are regulatory submissions from new reactor designs or generalized source terms from large Level 3 PSA studies. The former, source terms for new reactors, are of interest because of the relevance and the novelty of performing a Level 3 PSA with new reactors, which are being built (Finnish EPR) or could possibly be built in Nordic countries in the future. The obvious drawback to using new reactor designs is that information in literature is often incomplete or wholly omitted because it is proprietary.

The use of well-known Level 3 PSA studies provide the positive that they include input descriptions as well as Level 3 PSA results and methods descriptions, which may be useful in terms of benchmarking this pilot project.

The available resources for each alternative are discussed in the following sections. The final decision of how the source term input will be developed is discussed in Section 3.2.2.2 .

3.2.2.1.1. New reactor designs

An attractive choice for providing insight on the current operating reactors may be the publicly available information on the new reactor designs that are being constructed or have been. The plants that were researched were all "western" style nuclear reactors with large containment structures which are representative of plants that currently exist in Finland in Sweden, are being built, or may be built. After an investigation of publicly available literature on the subject, it became quite apparent that even though many of these designs are being subject to the same regulatory investigations, the information that is provided in these design submittals are quite different. Subsequently, some plant types have significantly more information on Level 2 / severe accident analysis and even Level 3 PSA analysis than other plant designs. Some of the positive and negative aspects of using the currently marketed designs are shown in Table 7.

One negative aspect toward using a new reactor type is the possible negative attention that publishing potential illnesses and death which could arise from a particular plant design when these designs may have no relevance to Nordic plants nor plants that may be constructed in the Nordic countries in the future.

Table 7. Pros and Cons for using new reactor designs as basis for new reactor study input, and available references.

Plant	Pros	Cons	References
ABWR	- Similar to BWR-75 plants currently operating in Sweden - Offered by Toshiba/Westinghouse & Hitachi-GE	- Difficult to find source term information in literature despite the broad number of organizations and current operating reactors	
ESBWR	- ESBWR Design Control Document provides some discussion on offsite Consequence Analysis	- Not as relevant for currently operating Nordic plants	[9]
EPR	- EPR Level 2 PSA largely available (UK-EPR submittals) - Relevant due to current Finnish- construction	- Not especially relevant to current operating Swedish reactors.	[10], [11]
APWR		- Comparatively little publicly available information on source terms / existing Level 3 PSA results	
AP1000		- Comparatively little information on source terms / existing Level 3 PSA results	

3.2.2.1.2. Large-scope Level 3 PSA studies

Another possibility for resource material for the pilot study would be existing large-scope Level 3 PSA studies. Using such material has several advantages. These inputs will probably provide much more rigorously defined source terms than those ascertained from the regulatory submissions of new reactor designs. These studies will also have extensive discussion on the Level 3 PSA methodologies employed as well as their results.

Using such old studies will mean that the pilot project may lack in terms of novelty, as the study would be revisiting well-trodden territory. Also, the studies are notably "conservative" in their applications and results, which is especially true for the oldest studies. Such conservatism is not ideal for drawing accurate assessments of the usefulness and applicability of Level 3 PSA results in the contemporary sense.

Some of the existing studies that have been identified are the WASH-1400 (1976) report, NUREG 1150 (1990), and the Probabilistic Accident Consequence Assessment Codes (1994).

WASH 1400

WASH 1400 is one of the oldest studies performed using PSA methods. The study is often described as one of the early "land mark" studies. In general, it is very thoroughly documented. [12]

The WASH 1400 study is often critiqued for being very conservative in terms of methodology and results. The study's results are markedly higher than those of subsequent studies performed more recently. The input material is fairly well developed, but some of the modelling considerations in the Level 1 and Level 2 portions of the analysis are dated.

U.S. NRC 5-plant study (NUREG 1150)

A study titled, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," was performed in the late 1980s and published in the early 1990s. As its title suggests, the report summarizes the analyses and results of a probabilistic study of plant risks, severe accident progression, and off site consequences for five commercial nuclear reactors in the U.S.

The study is one of the largest and well known Level 1/2/3 PSAs. The study's documentation and support documentation includes a description of the inputs used for the study and exhaustive descriptions of the results. These input data are, however, somewhat less complete than those provided in the 1994 code comparison and the information provided on the new reactor designs shown in Section 3.2.2.1.1. Like WASH 1400, the results have been argued as being conservative, (however less so than those in WASH 1400). [13]

1994 Code Comparison

Following the significant amount of work that was performed in the late 1980s in the field of Level 3 PSA / Probabilistic Consequence analysis (PCA) a very large-scope study of the various probabilistic off-site consequence analysis tools was performed. An earlier version of the analysis tool that is being presented for this pilot project, LENA, was included in the consequence assessment tools in the study.

The "pros" for using the code comparison study is that much of the input information is provided in the reports. So, with the exception of some of the information that was distributed via floppy-disks, the scenarios are quite fully represented in the reports themselves. The use of this report also comes with relevant data for several probabilistic accident consequence codes, including an earlier version of LENA.

On the negative side, the input is still somewhat incomplete since the raw data was delivered in a digital format. The event is not necessarily representative of Nordic plant configurations and typical Nordic weather/and population conditions. [6]

3.2.2.2. Source term input

The previous section, Section 3.2.2.1, developed the possible inputs from literature that could be used for performing the Level 3 PSA pilot study. This section will go into detail about the source terms selection, and why it was chosen, and the values that will be used for the analysis.

From the survey of literature that could provide possible input to the Level 3 PSA study and in particular the development of release categories and source terms, three clear alternatives surfaced. The three alternatives were the 1994 code comparison study, the US-ESBWR licensing topical reports on PSA, and the UK-EPR licensing documentation. Ultimately, it was decided that the new reactor units that had reasonably well defined Level 1, Level 2, and Level 3 PSA studies in literature would provide the best basis for the Level 3 PSA pilot study.

It was decided that the most representative exercise for performing an actual Level 3 PSA from Literature sources would be to perform an analysis from a Level 2 PSA. The 1994 code comparison study provided very detailed Level 3 PSA input, Level 3 PSA output / analysis, which included an earlier version of the LENA program. However, the code comparison study was not a Level 3 PSA study of an actual plant. The ultimate purpose of the study was to compare and contrast the values and capabilities of probabilistic consequence analysis tools. Since the scope was focused on distinguishing and comparing the tools themselves it was decided that basing a study on the analysis may miss some of the major questions and concerns of Level 3 PSA. Using such a study would not provide an exercise in developing release information from a Level 2 PSA. Nor, would such a study develop further insight in the sensitivity to choosing release categories.

The Level 1, Level 2, and Level 3 PSA studies that were provided by the US-ESBWR licensing topical reports and the licensing documentation for the UK-EPR represent fully developed PSA studies of prospective nuclear power plant designs. The level of detail of information provided is nearly complete, and is probably the most detailed source of input one could hope for of literature sources.

A more detailed breakdown of the information that is available in each the ESBWR and UK-EPR references are provided in Table 8. The primary missing element is the time dependent information from the severe accident analysis program (MAAP) in both cases. Another goal of the analysis that will be difficult to incorporate is the impact of a filtered containment system, which is not discussed in either report.

Based on the information provided for the UK-EPR and the ESBWR it was decided that the available input for the UK-EPR provided the most complete source of input information. Therefore the pilot project is based on the UK-EPR values. If time and resources permit USESWR inputs may be used for comparison to the UK-EPR information.

Table 8. Availability of source term input elements.

Alternative	ESBWR - LTR report	UK-EPR
Reactor inventory	Yes, the reactor inventory is based on bounding @ 102% power. It is not specified where in the cycle this would be representative.	Yes, The reactor inventory was provided in Chapter 15 of the US EPR DCD, or the release inventory was provided for each of the Release categories in the UKEPR (15.4.4.3).The Spent fool pool is also included in the UK-EP.
Release fractions	Yes, Release fractions are specified 24 hours, and 72 hours after the onset of core damage.	Yes, Release fractions for the Analysis performed in the reference were calculated with MAAP, these time histories are not provided, however a summary table of the releases are provided, and can be used as a rough approximation of the releases
Release categories	Yes. A range of sources and frequencies are provided. However, it is not described in detail why certain sequences were quantified. (15)	Yes. the release categories are quite well described in the Level 2 PSA documentation in the UK-EPR submission. (29)
Release Frequency	Yes. Included for several different sequences	Yes. Release frequencies are defined for each release category
Release Location	Unclear	Release location is quite well described in the UK-EPR documentation: 15.4.4.3. Even provides "junction"?
Release Height	Unclear (assume top of reactor building?)	Yes, Defined in [m]
Release Energy	Yes. Release energy is defined	Yes, Defined as a release energy rate [W], and as an integrated [J]
Particle size	Not specified… default values to be used	Not specified… default values to be used
Release time history	No. Raw data for time history is not provided. However quite detailed plots of results are provided out to	No time history is not provided
Release Delay	The time to start of the release is specified for each release	Yes. the time to start of the release is specified for each release
Release Duration	Not really. The release duration is poorly defined	Yes. Release duration is specified

3.2.2.3. Weather input

Collecting a significant and complete set of meteorological information to perform a Level 3 PSA is a difficult task. The weather information used in this analysis was borrowed from the data used in the previous Level 3 PSA thesis work [4]. These data were available through cooperation with SSM and representative of southern Sweden.

The data required for a single LENA calculation are wind speed, incoming wind direction, the mixing height, the amount of precipitation, and the stability class. The wind speed, and stability class are crucial to the determination of the shape of the plume. The precipitation parameter is important for deriving the deposition and the reduction in airborne radiological concentrations. The direction does not truly effect the LENA calculation because LENA only provides results based on the plumes center line in the downwind direction. However, this parameter is passed along, via the LENA output file, for the post processing program to determine the impact of the direction on consequences.

The weather data used for this analysis is best summarized graphically. First, Figure 3 shows the wind rose, which is a distribution of the wind velocities and angular direction of the wind used in this analysis. In the wind rose, a histogram of wind directions is shown with respect to the cardinal directions, while, the wind velocities are represented with separate color segments for each direction. Figure 4 shows a histogram of the stability conditions, and the probability of different precipitation rates.

It should be noted that a significant amount of Swedish weather information are made freely by the Swedish Meteorlogical and Hydrological Institute (SMHI). These data include time history of wind direction, temperature, and precipitation levels. At the time of writing this report, these data are not available specifically for nuclear plants sites. For more information see the following link:

http://opendata-catalog.smhi.se/explore/

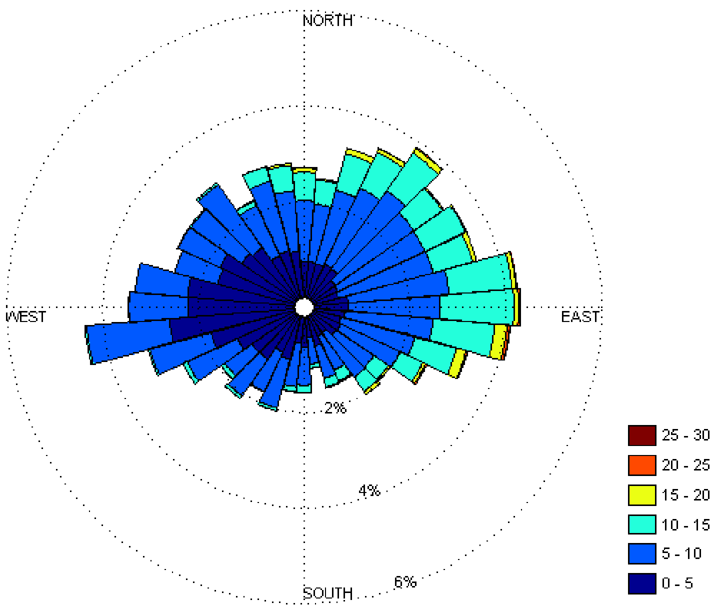

Figure 3. Wind rose showing distribution of information used in this analysis. The weather information was provided by SSM and characteristic of weather conditions in southern Sweden. Velocities scaled in meters per second [m/s]

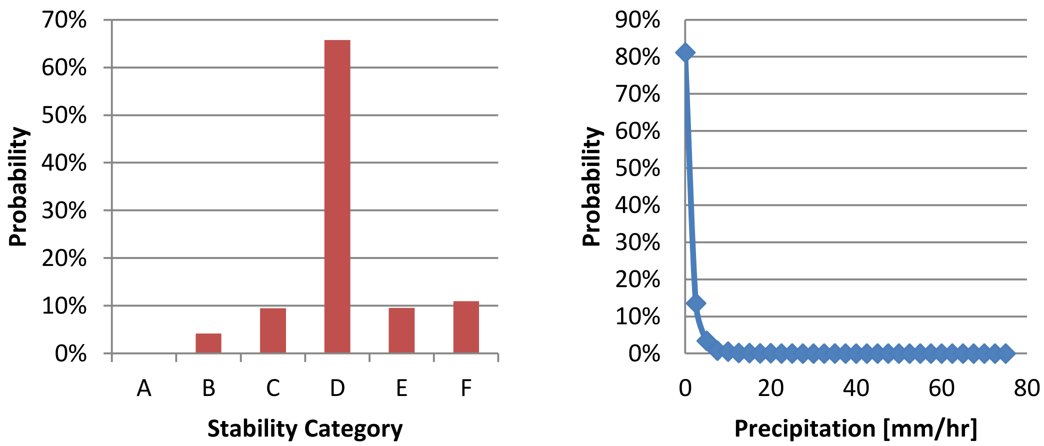

Figure 4. (left) The distribution of stability categories used in this analysis. A representative characteristic of the Swedish climate is the complete lack of the highly unstable atmospheric class A. (right) This figure shows the distribution of precipitation by probability.

3.2.2.4. Population input

The population is an integral part in determining the magnitude of the consequences following a nuclear accident. LENA provided the dose and deposition parameters for the dispersion calculation, but does not incorporate the population. In order to deduce collective doses and the effects to the populous, the LENA results will be combined with the population information with a post processing program, which will be further described in the Methodology Specification.

The population data used in this analysis was representative of southern Sweden, which also coincides with the weather information. Like the weather data this population data is borrowed from the previous Level 3 PSA thesis work [4].

The population distribution was also representative of a coastal location, where nearly 50 percent of the area surrounding the reactor site is unpopulated. These population data are also available within the SSM version of LENA's libraries for all populations within Sweden. These data, however, were relegated to the small population centers throughout Sweden, and do not capture populations residing outside of incorporated areas as well as those populations not included in census data. The population data used in this analysis was provided for 36 evenly spaced angles, providing a separate angular sector every 10 degrees, and 18 radial distances from 3-200 km, these data are shown in Figure 5. The populations per sector ranged from 524168 people north of the plant site to 0 people in the sectors located in the sea.

The population distribution does not further specify separate ages which can vary significantly and could affect long-term health effect calculations. This provides a limitation in the analysis. Future studies may choose to include such considerations to better describe the situation at hand.

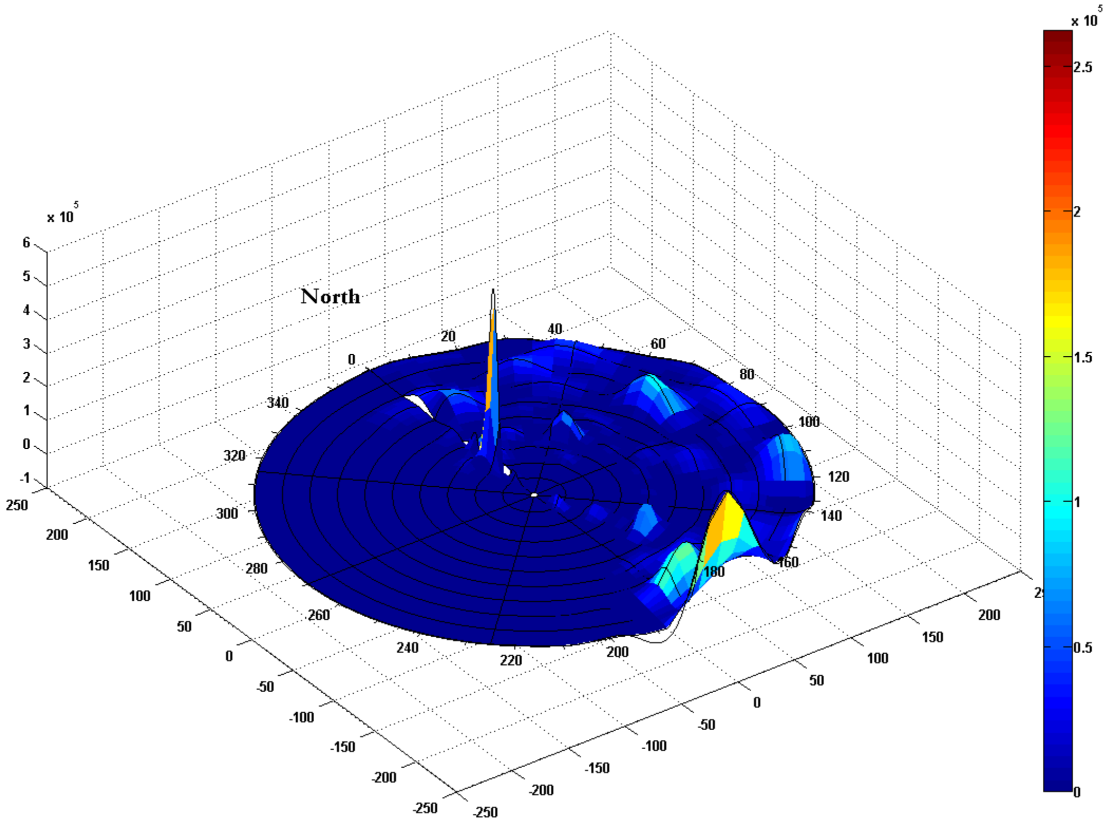

Figure 5. Surface plot representation of population distribution used in this analysis, representative of southern Sweden with maximum distance stretching 200 km and a coastal location. The colour bar to the right of the graphic shows the correspondence between the colours of the surface plot and the population per sector.

3.2.2.5. Protective actions input

Countermeasures can be applied in several different ways using LENA:

- Shielding factors
- Filter factor
- Deposition velocities

Shielding factors and filter factors that are applied for reactor accidents can often be divided by exposure pathway (e.g. ground shine, cloud shine, etc.). This functionality is not strictly possible in LENA, although shielding factors can be modified in the program globally. There can be substantial regional differences in these factors, which can make it difficult to use "general" shielding factors very. An example of how the US NRC State-of-the-art reactor consequence study implemented shielding factors is summarised in Table 9. More discussion on how countermeasures will be calculated will be provided in the Methodology Specification Report.

Table 9. SOARCA Surrey Shielding Factors [15]

	Ground Shine			Cloud Shine			Inhalation/Skin		
	Normal	Evac.	Shelter	Normal	Evac.	Shelter	Normal	Evac.	Shelter
Cohorts	0.26	0.5	0.2	0.68	1	0.6	0.46	0.98	0.33
Special Facilities	0.05	0.5	0.05	0.31	1	0.31	0.33	0.98	0.33

3.2.2.6. Economics input

The economics of the accident is planned to be modelled in a very simplified manner. It was suggested during the development of the project to make a simplified analysis based upon the number of displaced households following an accident. Therefore, no specific economics data were collected at this stage of the project.

3.3. Scope of Analysis

This section presents the scope of analysis to be performed for the Swedish Pilot Project within the NKS/NPSAG Level 3 PSA project.

The scope of analysis needs to be closely tied to the overall Level 3 PSA project goals. These goals are detailed in Section 3.1.1. The analysis scope proposed in this report is aligned to the resources available to a pilot-type project, hence it is important to recognize that the analysis scope is not trying to illuminate the aspects of all the project goals in detail. Rather, the analysis scope purposely focuses on goals which can be feasibly considered given the project constraints in terms of both resources and generic publicly available New Build nuclear power plant input data.

Two project constraints which particularly affect the analysis scope are discussed in more detail in separate sections below, as these constraints serve as starting points for defining the analysis scope. These constrains are:

1) Output available from the used probabilistic consequence analysis code LENA.
2) Nuclear power plant severe analysis progression and Level 2 PSA data available to serve as input to the Probabilistic Consequence Analysis (PCA) code.

The next sections present an overview of the LENA output and the available severe accident progression input. Using the overview of available input and output for the pilot project, the scope of analysis is presented.

3.3.1. LENA output and its application in a probabilistic framework

To establish what is achievable with the scope of a pilot project it is instructive to describe the output which is available from the LENA probabilistic consequence analysis code utilized in this Level 3 PSA project.

The output available from LENA can be grouped as listed below:

a) Cloud induced dose uptake both with respect to internal and external exposure paths. Including organ doses for thyroid and lungs.
b) External dose rate post-cloud passage.
c) Ground contamination, in particular due to ^{131}I and long-lived Cs (^{137}Cs is calculated in LENA).

An illustration of the available LENA output is given in Figure 6.

The output data is valid for a particular weather configuration (ie wind direction, wind speed, precipitation etc), time after the beginning of the accident and specified release characteristics which are dependent on the severe accident progression of the chosen (representative) sequence.

To use the raw output from LENA, which is in deterministic form, in a probabilistic fashion, sampling must be performed on weather data and for each weather data point the dose contours need to be calculated for the area around the plant and at select times post-initiation of the radioactive release.

At a high level the LENA output will provide:

a) Dose uptakes by individuals at different locations away from the plant.
b) Ground contamination levels at different locations away from the plant.

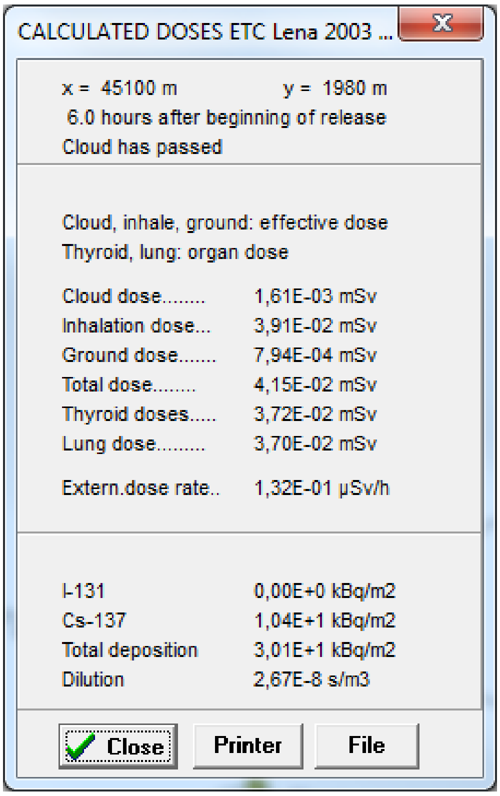

Figure 6. Core output produced by LENA 2003 (note that the specified release in this case only consisted of Cs).

3.3.2. Available Level 2 PSA output data selected for the Swedish pilot study

Another key ingredient, which significantly impacts the analysis scope, is the available nuclear plant data that serves as input data to the LENA code.

As reported in the Input Specification (See Section 3.2), the data from the publicly available UK-EPR Pre-Construction Safety Report (PCSR) was chosen as a generic source of input data for the Level 3 PSA study. One of the key challenges with defining the analysis scope is to select an appropriate set of severe accident scenarios for studying the application of Level 3 PSA which on one hand provides sufficient coverage to provide insights to help fulfill the project goals while at the same time is sufficiently limited in scope to fit within the constraints of a pilot study.

The publicly available Hinkley Point C (HPC) EPR PCSR [4] presents a reasonably complete set of Level 2 PSA output suitable for further analysis. It should be emphasized that given the selection procedure of the HPC data and the limited nature of the study, any insights which can be derived from this study only try to cover the limited set of project goals, and, hence, any conclusions or suggestions mentioned in this report only apply to the application of Level 3 PSA from the Nordic point of view. Consequently, any information in this and other Level 3 PSA project reports does not provide any meaningful insights with regard to the particular design of the HPC EPR.

Sequences represented by corresponding Release Categories were selected firstly such that

1) They cover a reasonable range of Cs releases as represented by the MAAP second isotopic group (the CsI/RbI group) release fraction (corresponding to the maximum between the 2nd and the 6th (CsOH/RbOH) MAAP isotopic group).
2) The timing of the release is considered in the selection process as releases with adequate warning time will need due consideration in a Level 3 PSA and may augment implied consequences as interpreted using Level 2 PSA output.

As a starting point for finding a suitable range of Cs release fractions the threshold between an acceptable and non-acceptable release utilized in the Swedish regulatory framework. The Swedish regulatory Cs release fraction threshold between an acceptable and a non-acceptable release (see [18] for more information) for a reactor of the size of the EPR (see [21]) is about $0.1\% \cdot 0.4 = 0.04\% = 4 \cdot 10^{-4}$ (the ratio 0.4 originates from the ratio between the thermal powers of the Swedish Barsebäck nucleaer power plant unit versus the EPR which is given by the factor 1800 MWth / 4500 MWth). Selecting scenarios around the threshold seeks to establish the level of added insight Level 3 PSA based risk metrics could provide compared to risk metrics based on Level 2 PSA output as currently used.
In the context of the Swedish regulatory framework, it should be noted that one of the overarching requirements is that deaths due to the early effects of radiation must not occur. There are no additional specific quantitative requirements associated with the risk of early deaths apart from the acceptable/non-acceptable Cs release threshold. For this reason, the analysis scope enables the consideration of early effects within the Level 3 PSA study.

In terms of timing of the releases which can be important from the perspective of a Level 3 PSA, a divider of 10 hours post-core damage is often used in Sweden as the distinction between early and late releases. Therefore, in the scope of analysis, scenarios are selected such that early and late releases have adequate coverage.

The high-level matrix depicted in Table 10 below provides a high-level overview of the selected scenarios as represented by the available Release Categories (RC) utilized in the HPC EPR. The table indicates that, as expected, the threshold of non-acceptable release is not a good measure for the radiological magnitude of early releases. The reason for this is that since early releases tend to be large the early releases all end up being unacceptable, making them indistinguishable when solely utilizing the single threshold for Cs release currently in use.

Table 10. Selection matrix for SA scenarios for further Level 3 PSA analysis. RC numbers and Cs release fraction in parenthesis are from [17].

Cs Release Fraction/ Release Timing	<0.04%	≈0.04%	>0.04%	>>0.04%
Early (release starts < 10 hr post-CD)	No relevant case found	No relevant case found	RC 802b (Small, 9.17E-4)††	RC 202 (3.99E-3) RC 205 (1.16E-1)
Late (release starts > 10 hr post-CD)	RC 501 (5.72E-5) RC 503 (1.08E-4)	RC 504 (4.08E-4)	RC 502 (7.72E-4)	RC 404† (2.47E-2)

†: Release starts at 7.8 hr, however, since the release is of long duration it is judged adequately represented as a late release.
††: The maximum of the CsI and CsOH MAAP isotopic group release fractions is listed.

Given the scenarios listed in Table 10, the deterministic LENA code will be run using sampled weather information in order to produce dose and contamination contours around the plant out to a distance of up to 100 km from the plant for each of the eight RCs listed in Table 10.

Given the coarseness of input data, a single phase release will be assumed in the Level 3 PSA analysis. Assuming a single phase release also significantly simplifies the analysis making the analysis tenable for a pilot type study.

3.3.3. Level 3 PSA analysis scope for the Swedish Pilot Project

Performing probabilistic consequence analysis calculations for the specified scenarios listed in Table 10, it is possible to evaluate Level 3 PSA specific risk measures. A matrix which seeks to provide an overview of the key probabilistic consequence analysis combinations is given in Table 11.

The analysis scope envisions looking at three key analysis characteristics together with a set of five Level 3 PSA risk measures. The importance of each of the attributes from the point of view of a Level 3 PSA analysis is summarized below.

From the point of view of risk metrics for a Level 3 PSA study, use of health related risk metrics is standard and would be required in even a very limited scope study. This would also be the case for environmental risk metrics, although to a lesser degree. Although complex to accurately evaluate, an economic type risk metric can be considered the ultimate output of a Level 3 PSA and is, therefore, discussed below from the view of the Swedish pilot study. As the Swedish regulatory framework focusses on prompt fatalities due to radiation and land contamination (see, for instance, [18] for further information), the proposed analysis scope specifically includes risk metrics which measure these aspects of the impact of a release of radioactivity.

From the perspective of health impacts from a release, the proposed analysis scope includes the risk metric given by the maximum dose uptake for a hypothetical individual situated 1 km away from the plant. This risk metric seeks to measure the risk of death due to early effects in line with Swedish regulatory focus. In addition to the maximum individual dose, two other health related risk metrics are utilized, namely, the collective dose burden within the specified analysis area and the predicted number of latent cancer fatalities. It is anticipated that, depending on the convolution of the population density and the release profiles that the extent of the analysis area may impact these metrics. The analysis area is defined as the maximum radius around the plant for which the probabilistic consequence analysis is carried out. As a starting point for this pilot study it is proposed to use a simple evaluation of the number of latent cancer fatalities based on the collective dose.

The environmental impacts from a release is covered in the proposed analysis scope through the risk metric defined as the land area with significant Cs surface contamination in line with Swedish regulatory interest of avoiding long-lived land contamination. Implied in this risk metric is an assumed threshold of acceptable Cs surface contamination above which the contamination is deemed significant. The value of this risk metric is simply the land surface area with a level of Cs surface contamination above the specified threshold value. The two currently suggested threshold values were chosen based on operating experience from the Fukushima accident.

It is generally accepted that a measure of the economic impact of an accident would be the ultimate goal of a Level 3 PSA assessment. This is understandable since in practice it would be useful to be able to compare some form of monetary value associated with nuclear risk impacts, for instance, to evaluate changes or impairments to safety significant plant features. Unfortunately, a complete treatment of the economic risk metrics is complex and is out of the reach of a limited scope pilot project. However, in order to try to illustrate the benefit of evaluating some form of economic impact it is suggested that a simple economic measure consisting of the economic loss associated with the total land area assumed to be lost, ie land which, for the purposes of this simple study, can be assumed to inaccessible both for residential and agricultural purposes. The main idea behind the risk measure is to compute a value of the land area which is considered lost based on an assumed maximum allowed Cs surface contamination level and a simple average price of the land (taking into consideration the current use of the land, eg population/agriculture).

The analysis area indicates how far from the plant the Level 3 PSA analysis is to be conducted. This attribute could be important from the point of view of capturing the impact of the release to an adequate degree, and analysis for the two alternatives would seek to answer if a smaller analysis area could suffice. The attribute will have the greatest impact on all the risk measures except the two risk measures which utilize the maximum individual dose at 1 km from plant, as this particular dose is captured by both analysis area alternatives. The analysis area parameter values suggested for the analysis scope were selected using the spread of radioactivity in terms of Cs ground contamination observed in the Fukushima event (see [19] for contour map of ^{137}Cs ground contamination).

With respect to countermeasures only a single measure, namely the implementation of a small evacuation zone around the plant is considered; in the proposed analysis scope a 5 km radius around the plant was suggested as a representative evacuation zone. It should be noted that the countermeasure is only expected to have any significant effect on the maximum individual dose received by a hypothetical individual closest to the plant. Evacuation of people around the a relatively small zone around the plant is assumed to be fully implemented within the 10 hours warning time characteristic of the late releases such that the countermeasure is able to provide a degree of mitigation. Note that when people are assumed to be evacuated within 5 km of the plant, the 1 km maximum individual dose utilized for the early effect health risk metrics is instead evaluated at 5 km from plant for the scenarios with adequate warning time.

A number of elements in the analysis case matrix given in Table 11 are not considered, as indicated with a '-'. These combinations are judged not worthwhile pursuing as part of the pilot project, as explained below.

For the maximum individual dose risk metric only the smallest analysis area suffices as larger analysis areas would not yield additional information as the risk metric is driven by information within a distance of 1 km from the plant.

The countermeasure consisting of the implementation of a 5 km evacuation zone around the plant is judged only to have any significant impact on the maximum individual dose risk metric with only a minor impact on the collective dose since in most realistic scenarios the population beyond the 5 km boundary typically far exceeds the population close to the plant. A minor impact on the collective dose implies, at least to first order, a minor impact on latent cancers, when the detailed population dose uptake profile is neglected. Since an evacuation of the population does not affect the level of land contamination, the evacuation countermeasure has no impact on the environmental risk metric used in this study. Similarly, the economic risk metrics used in the analysis scope is based on land contamination and, hence, the evacuation countermeasure also has no impact on this risk metric.

Finally, since the threshold for Cs ground contamination only servers to determine whether the contamination level can be considered significant from the perspective of calculating the environment and economic risk metrics there is no impact of the Cs ground contamination threshold on the health related risk metrics.

At the moment there are a number of chosen analysis parameter combinations to be analyzed for each chosen severe accident scenario. It should be noted that the particular definition of analysis cases ensures that the risk metric results can be extracted using appropriate post-processing of a single set of LENA results for a given accident scenario without the need of re-running LENA multiple times for a single accident scenario.

Table 11. Analysis cases for the evaluation of Level 3 PSA specific risk measures

	Metrics	Health				Environment	Economic
Analysis Characteristics	Risk Measure/ Assumption	Maximum individual dose at 1 km (early effects)	Risk of (early) death to max-imum exposed individual	Collective Dose (late effects)	Number of Latent Cancers (late effects)	Size of land area with significant Cs contamination	Estimate of value of lost land due to Cs contamination
Analysis Area	Up to 50 km	X	X	X	X	X	X
	Up to 100 km	-	-	X	X	X	X
Countermeasures	5 km evacuation zone	X	X	-	-	-	-
Cs‡ ground contamination thres-hold	1000† kBq/m^2	-	-	-	-	X	X
	100† kBq/m^2	-	-	-	-	X	X

†: Cs ground contamination thresholds may need some iteration once radioactivity contour maps have been produced.

‡: Combined activity of ^{134}Cs and ^{137}Cs.

4. International activities during 2014

The project working group has also been engaged in several of the ongoing international Level 3 PSA efforts during 2014. The two primary ongoing activities internationally are the development of an IAEA TECDOC on Level 3 PSA and the development of a Level 3 PSA Standard by the American Nuclear Society.

4.1. IAEA

The IAEA issued a procedure guide on Level 3 PSA in 1996, IAEA Safety Series No. 50-P-12, "Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 3)," following significant work performed in the US, Europe, and Japan in the field of Level 3 PSA methods.

The IAEA has recently reopened the issue of Level 3 PSA with an IAEA Technical Meeting on Level 3 PSA, which took place in July of 2012. The meeting was the first activity specifically discussing Level 3 PSA since the publication of the IAEA Safety Series No. 50-P-12. The purpose of the meeting was to articulate the work performed during this meeting, monitor any further IAEA developments and also follow and discuss similar developments in international and national organizations.

Following the IAEA Technical Meeting, two further IAEA activities have taken place. The first was an Eastern European Regional Workshop on Level 3 PSA, and the second was a Consultant Meeting on Level 3 PSA. The funding provided by the project allowed the working group to participate in both activities.

4.1.1. The objectives of the TECDOC

The objectives of the TECDOC are the following:

- Outline the methodology and indicate the techniques most widely used to date
- Provide general guidance for conducting a Level 3 PSA with description of major technical elements (e.g. interface between Level 2 and Level 3 PSA, atmospheric dispersion, countermeasures, consequence results interpretation)
- Survey of current practices and computer codes available for consequence assessment (real difficulties learned by Level-3 PSA analysts)
- Provide information on the use of Level 3 PSA and applications, and effective presentation of the results
- Identify areas of further research
- Update previous (now outdated) IAEA of the previous IAEA Level 3 PSA publication.

4.1.1.1. Scope:

- Level 3 PSA for nuclear power plants considering all facilities at the NPP site is in focus
- However, the general methodology may be also applicable for other parts of the nuclear fuel cycle, such as reprocessing plants and spent fuel storage installations, and also for research reactors, although specific aspects of Level 2 and Level 3 analysis

may be quite different for such installations and appropriate models would need to be used.

- Not prescriptive document

The general scope of the TECDOC should not be completely different from the scope outlined in the IAEA Safety Series No. 50-P-12, publication:

The main emphasis in this Safety Practices document is on the procedural steps of a PSA, rather than on the details of corresponding methods. This document is primarily intended to assist technical personnel with responsibilities in managing or performing PSAs. A particular aim is to promote a standardized framework, terminology and form of documentation for PSAs so as to facilitate external review of the results of such studies. The report outlines the methodology and indicates the techniques most widely used to date.

In general, this document seeks to provide sufficient detail to define unambiguously the methods to be used, while avoiding prescriptive detail at a level that would inhibit the flexibility of the user in applying available resources, recognizing that the resources available to various studies will vary widely. The publication of this report is therefore not intended to pre-empt the use of new or alternative methods; on the contrary, the advancement of all methods of achieving the objectives of PSA is encouraged. (IAEA, 1996).

4.1.2. IAEA TECDOC Status

Two Consultant meetings were held in 2014 in order to update the text of the “superseded” Safety Series guide IAEA Safety Series No. 50-P-12 (5 days in May 2014 and 5 days in December of 2014).

A complete draft has been completed by the end of the December 2014 meeting. A significant amount of editing and revising needs to be performed by the working group. The next meeting will take place in July of 2015.

Additional guidance, mostly by way of references, will be provided for multi-unit accidents and aqueous pathways as they pertain to Level 3 PSA.

4.2. American Nuclear Society Level 3 PSA Standard

The ANS Level 3 PSA standard working group did not meet during 2014. Further work is underway during 2015.

5. Conclusions and future work

The project is planned to continue through 2015. A significant amount of work was completed during the first two years of the project (2013-2014).

Task 3, which is the task allocated for developing the final guidance document, and the Swedish part of the Task 4 pilot project began during 2014. The bulk of the work on the guidance document will be performed during the last 6 months of 2015. The Pilot studies will continue through the first part of the year of 2015.

5.1. Finnish Pilot Project

This Finnish Pilot study demonstrates the application of deterministic and probabilistic methods in Level 3 PSA. On the deterministic side, VTT's consequence analysis code ARANO is used in calculating the atmospheric dispersion of a release of radioactive substances, and in estimating the total dose of ionizing radiation. On the probabilistic side, VTT's level 2 PSA code SPSA is used to assess the probabilities of different consequences. The main model is an event tree, where each branch concerns either the value of a weather variable (wind direction, wind speed, precipitation) or a countermeasure variable (evacuation success, sheltering success).

The case considered is an alternative take on the Fukushima Daiichi nuclear power plant accident: what radiological consequences (in terms of population dose and cancer deaths) would the radioactive release from the site have had, if the population of the major cities close to the site had been in place (and not killed by or evacuated after the earthquake and tsunami), and the impact of weather is analysed on the basis of what it statistically is in March in that part of Japan?

The population doses are analysed in the event tree, and uncertainty analyses are conducted on the weather variables, evacuation and sheltering success probabilities, and the effectiveness of sheltering. We find that, even under rather conservative assumptions, the radiological consequences are small. However, the results should be seen as only indicative due to simplifications made in modelling.

The pilot study demonstrates that the approach used is a viable way of conducting Level 3 PSA.

Additional work on the study has been outlined and are being examined for 2015.

5.2. Swedish Pilot Project

During 2014 a few major activities were completed and several others started for the Swedish Pilot study: The Pilot Study Plan, and The Input Specification. The Scope of Analysis was started during the year with a first draft completed and presented to stakeholders. The Scope of Analysis will be completed during 2015 along with the Methodology Specification, and Application and Result interpretation.

5.2.1. Input Specification

This input specification is split into two major sections:

- A brief overview of Level 3 PSA standard practice.
    - This information can be supplemented by more detailed guidance provided by the IAEA, ANS, and ASME.
- A discussion on the input data that will be used in this study

The discussion in of input data provided some justification for why the EPR reactor would be the focus of the source term development for the Swedish pilot project. Table 8 provided a description of the input data available in the several publicly available publications for the EPR and ESBWR. Further input data, such as weather and population data, will be extracted from the Thesis work outlined in Reference [4].

Further details on the metrics to be used and how the input data will be applied will be provided in the Scope of Analysis and Methodology Specification reports. A final report will outline the application and results upon completion of the study.

5.2.2. Scope of Analysis

The analysis scope proposed for the Swedish Pilot Project of the NKS/NPSAG Level 3 PSA project is given by the severe accident scenarios listed in Table 10 combined with the Level 3 PSA analysis cases proposed in Table 11.

Reverting to the overall project goals listed in the introductory section, the analysis scope proposed in this report will help the project provide:

- Additional insights provided by Level 3 PSA output within a regulatory framework based on thresholds related to activity release
- Indications to which extent current Level 2 PSA output may serve as potential surrogates for full Level 3 PSA output
- Indicative resourcing required for performing Level 3 PSA
- Insights into calculation and usage of a broad range of Level 3 PSA risk metrics, including health, environmental and simple economic risk measures.

The proposed analysis scope will help the project to ultimately provide important insights related to the main project goals. It should, however, be noted that the current analysis scope will provide little insight into how a Level 3 PSA could be integrated into the Level 2 PSA structures currently used at the Swedish plants. This is mainly due to the source of plant input data (UK EPR) utilized for the project.

5.3. Continuation of work

During 2015, the final year of the project, the pilot projects will be completed and a guidance document will be formulated along with the project stake holders. The working group will remain engaged in international activities surrounding Level 3 PSA, the development of the IAEA Level 3 PSA TECDOC and the ANS/ASME Level 3 PSA Standard.

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Abstract
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The goal of this project is to further Nordic understanding of the potential for Level 3 PSA to determine the influences and impacts of off-site consequences, the effectiveness of off-site emergency response, and the potential contributions of improved upstream Level 1 and Level 2 PSAs.

This status report summarizes the work performed during 2014, which mainly belongs to Task 4, the Pilot Application, which is separated into two parallel activities, the "Swedish" and "Finnish" Pilot projects.

The Finnish Pilot study demonstrates the application of deterministic and probabilistic methods in Level 3 PSA. The case considered applies the source term of the actual Fukushima Daiichi nuclear power plant accident, but without taking into consideration that the population was evacuated before the accident due to the tsunami

The population doses are analysed in the event trees, and uncertainty analyses are conducted on the weather variables, evacuation and sheltering success probabilities, and the effectiveness of sheltering. We find that, even under rather conservative assumptions, the radiological consequences are small. However, the results should be seen as only indicative due to simplifications made in modelling.

For the Swedish Pilot study, two major activities were completed during 2014: The Input Specification and the Scope of Analysis.

The input specification includes a discussion on the input data that will be used in this study. Here some justification for why the EPR reactor would be the focus of the source term development for the Swedish pilot project. Further input data, such as weather and population data, will be extracted from a Thesis work.

The proposed analysis scope will help the project to ultimately provide important insights related to the main project goals.

During 2015, the final year of the project, the pilot projects will be completed and the guidance document will be formulated along with the project stake holders. The working group will also remain engaged in international activities surrounding Level 3 PSA, the development of the IAEA Level 3 PSA TECDOC and the ANS/ASME Level 3 PSA Standard.

Key words PSA, PRA, Level 3 PSA, Probabilistic Consequence Analysis, Nuclear Power Plant Safety