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Addressing off-site consequence criteria using Level 3 PSA -

Phase 3 Status Report from the NKS-R L3PSA

(Contract: AFT/NKS-R(16)109/8)

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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 report summarizes the developments from four years of work, but focuses on the finalization of a Nordic Level 3 PSA Guidance Document which has been worked upon mainly during calendar years 2015 and 2016. Other activities that has been conducted, and provided valuable input to the Guidance Document, are an Industrial Survey, a study of potential Risk Metrics, a summary of Regulations & Standards, and two Pilot Studies (one Swedish and one Finnish). The main objective of the pilot studies was to gain practical experience that, together with insights from the other tasks included in the project, could be transferred to recommendations into a final guidance document.

During the project, targeted discussions between consultancies, utilities, regulators, and insurance companies on the subject of Level 3 PSA have taken place and at the end of each years working period a seminar has been arranged. The working group has also been engaged in international activities surrounding Level 3 PSA, i.e. the development of the IAEA Level 3 PSA TECDOC and the ANS/ASME Level 3 PSA Standard through the 2016 continuation of the project.

All project reports are provided as appendices to this final report.

Key words

PSA, PRA, Level 3 PSA, Probabilistic Consequence Analysis.

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Project reports / Appendices

Each of the task reports and pilot reports produced during the project are provided as attachments to this report:

Appendix A – Industrial Survey

Appendix B – Risk Metrics

Appendix C – Regulations and Standards

Appendix D – Finnish Pilot Study

- Appendix D.1 – Applying IDPSA in PSA Level 3
- Appendix D.2 – Improvements to Event Tree Model and Case Study

Appendix E – Swedish Pilot Study

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

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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 Daiichi 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 benefits of such an analysis. During the project there has been close interaction with utilities, regulators, and insurance companies 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, mainly as a consequence of the Fukushima Daiichi accident but also due to the interest in new reactors (primarily during a time up and until the Fukushima Daiichi accident).

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 the 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

As the primary goal of the project is to develop guidance on several significant topics related to Level 3 PSA, the aim/scope of the reports and seminars has been to develop guidance for 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

In order to achieve the above the project has included the following activities:

- The development of an industrial survey completed by Nordic utilities, Nordic Nuclear Safety Authorities, and Nuclear PSA experts.
- A study of Risk Metrics
- Involvement in IAEA and ASME/ANS Level 3 PSA activities.
- Two parallel Level 3 PSA pilot studies (conducted using Swedish and Finnish probabilistic consequence analysis tools).
- Three project seminars/workshops which have provided valuable conclusions and discussions.

1.3. Project organization

The project has included separate tasks that have been conducted in parallel. The separation of project tasks enabled the working group to efficiently divide responsibilities in the project while maintaining cohesion in the review process. The project tasks address the following topics, each in a separate task:

Task 0 – Industry and Literature Survey

Task 1 – Appropriate Risk Metrics,

Task 2 – Regulation, guides and standards,

Task 3 – Development of a Guidance document

Task 4 – Pilot Applications including tools for dispersion and consequence analysis

It shall be noted that even though development of the Guidance Document is listed as task three out of the total four tasks, the task number does not accurately reflect the chronological order of performance of the tasks. How the project tasks relates to each other in a more chronological order and how the tasks have influenced the Level 3 PSA Guidance Document is displayed in figure 1 below.

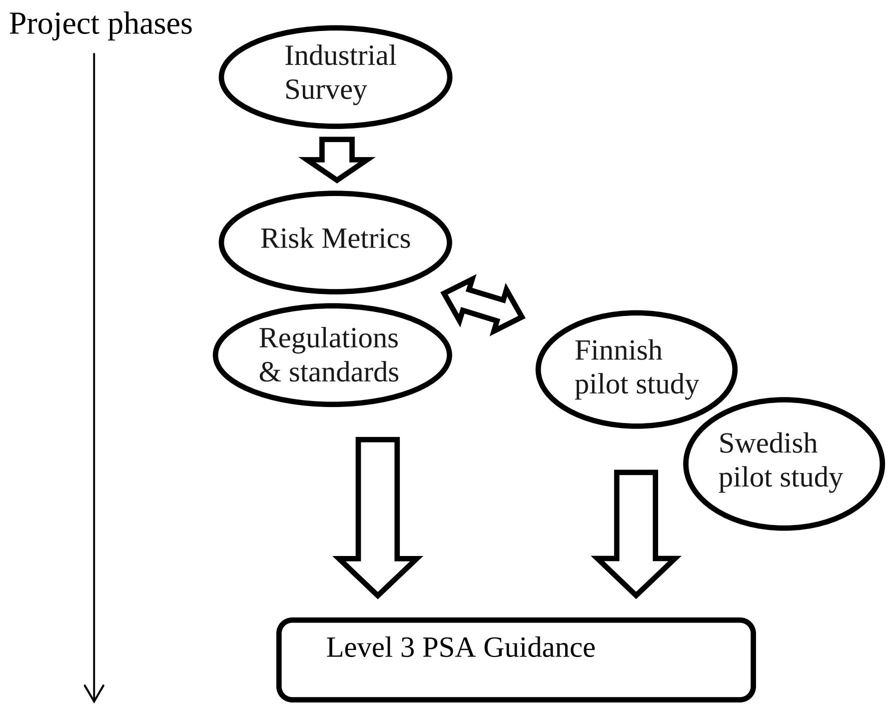

Figure 1. Level 3 PSA project overview.

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". The working group has also held four annual project seminars, in the beginning of each year (2014, 2015, 2016 and 2017). While the purpose of the first three seminars was both to summarize the progress during the previous year (2013, 2014 and 2015) and discuss a pathway forward for the coming year, the purpose of the 2017 seminar was focused on presenting the Guidance Document and to receive feedback on this.

The project has created interest in many international organizations and has fostered Nordic participation in several international Level 3 PSA activities. The project has also allowed the working group to contribute to IAEA Level 3 PSA efforts through member participation in IAEA Technical Meeting & Consultant Meetings as well as act as an expert lecturer for an IAEA Regional Workshop on Level 3 PSA. In addition, the project has interfaced with groups such as OECD/NEA Working Group RISK and the ANS/ASME Level 3 PSA standard writing committee.

1.5. Report contents

The project was originally planned to be a 3-year (2013, 2014 and 2015) exploratory study into Level 3 PSA but it was extended over a fourth year (2016) in order to get additional insights from international activities. During the final year significant efforts has also been to iterate the Guidance Document with the funding stakeholders of the Nordic PSA Group (NPSAG). Three previous annual reports have been written, detailing the first three years of the project [1], [2] and [3].

This report describes the developments from all four years of work with a specific focus on the finalization of Guidance Document (Appendix F) and the conclusions presented there (i.e. work conducted during 2016). The following sections briefly summarize the work performed under each of the separate Tasks as outlined in Section 1.3. Complete discussion on each of the tasks, are attached as appendices to this report where the appendices represent the actual documentation that has been produced for each 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). 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 or NPSAG. In particular, neither NKS, NPSAG, nor any other organisation or body supporting NKS activities can be held responsible for the material presented in this report.

2. Industrial Survey

The purpose of the Industrial Survey was to develop a baseline for the state of knowledge, the opinions, and interests with regards to Level 3 PSA. The Industrial survey opened the dialog between the working group, the project stakeholders, as well as other interested groups such as insurance companies. The complete findings of the Industrial survey are provided in the task report that is provided as Appendix A to this report.

3. Risk Metrics

The main goal of the Level 3 PSA risk metrics study was to discuss which could be the appropriate risk metrics for Level 3 PSA and to inform both the pilot studies and guidance documents. No safety goals, i.e., no numerical criteria, were explicitly connected to the risk metrics presented. However, safety goals were touched upon as a reference to which risk metrics that could be used.

A risk metric has two components: 1) probability (frequency) and 2) consequence (or impact) metric. Regarding the probability metric, it is a matter of choosing the normalization unit for risk comparison purposes (e.g. “per reactor year”, “per lifetime” or “per produced energy/electricity over the complete fuel cycle”). The consequence metric is associated with the impacts that are quantified in the consequence assessment part of Level 3 PSA. The following main groups of consequence metrics have been identified:

- Health effects
- Environmental impact
- Economic impact (can include every other risk metric).

The task report for the risk metrics study is provided as Appendix B to this report.

4. Regulations and standards

This project has also interfaced with several of the activities that are occurring internationally with regards to Level 3 PSA. The two primary activities, which are further outlined in task report in Appendix C, are the following: the development of the ANS/ASME Level 3 PSA Standard, and the drafting of an IAEA TECDOC. Both of these activities have seen active participation from the NKS/NPSAG Level 3 PSA working group. Unfortunately, both of these projects have seen stunted progress, but will hopefully see publication in the next several years.

5. Pilot studies

The pilot project is separated into two parallel activities, the "Swedish" and "Finnish" pilot studies. This section details the developments of these pilot studies.

5.1. Finnish Pilot Study

The task reports related to the Finnish pilot study is provided in Appendix D.1 and D.2.

5.1.1. Goal of Finnish Pilot Study

The main goal of the pilot was to study how to apply the IDPSA (Integrated Deterministic and Probabilistic Safety Assessment) methodology on level 3 PSA.

There were 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.

5.1.2. Description of pilot case

The Finnish Pilot project was an exercise in alternate history, and sought 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, 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.

In the pilot study it was assumed that the release would have been much more abrupt than it was (in reality there were multiple releases over several months). It was also assumed that the whole release would have happened in three hours. As the source term, the actual source term of the Fukushima Daiichi accident is used. Assuming such a short release time span is simplifying and in terms of health effects conservative.

5.1.3. Conclusions

A case of alternative history of the Fukushima Daiichi accident was modelled and analysed in order to assess what the radiological consequences would have been in terms of cancer deaths in the following scenario:

> *What would have happened if the source term of the Fukushima Daiichi NPP accident would have been released rapidly and the population of the large cities 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.*

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 the model used in the pilot study, of cancer deaths resulting from the release is 3.6, 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 (worst wind direction and speed, countermeasures failed) 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 amounts 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. 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 managing uncertainties through 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. Furthermore, using an event tree facilitates the conduct of uncertainty analyses, and e.g. enables the conduct of uncertainty analyses in the deterministic model (ARANO) and the probabilistic model (FinPSA) separately in a coordinated manner, thus allowing for relatively comprehensive uncertainty analyses with a reasonable amount of work and computational load.

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 modelling 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 the model in a rudimentary manner: evacuation is considered a success with a certain probability that depends on the time available for evacuation (the time it takes for the release to reach the city considered, given wind speed and the city's distance from the site). This evacuation model does however 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 such as the risks involved in evacuation etc. More refined evacuation models might shed light on the effects of these factors.

5.2. Swedish Pilot Study

The task reports related to the Swedish pilot study are provided in Appendix E.1 to E.5.

5.2.1. Swedish Pilot Plan

At the beginning of the Swedish pilot study a long list of 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 study's deliverables and promote cooperation between the many organizations participating in the project, a group of project reports were developed. An overview of the plan of the study is given in the Swedish Pilot Project Plan report (Appendix E.1).

5.2.2. Goals of study

The main project goals identified were the following:

1. To cover which types of insights can be attained from a Level 3 PSA
    a. Discrimination of consequences which exceed a regulatory risk threshold, e.g. released activity, marginally or substantially.
    b. Seek to establish the extent Level 2 PSA output may be relevant as a surrogate for Level 3 PSA insights.
2. To indicate resources (amount of and skills) required for performing a Level 3 PSA
3. To identify any key uncertainties in the analysis
4. To indicate how existing plant Level 2 PSA structure would interface with a Level 3 PSA analysis
5. To 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.

5.2.3. Pilot study reports

The pilot study subtasks were broken up into separate reports. The reasoning for producing several different reports for the major phases of the work was to allow the large group of stakeholders and working group members to collaborate throughout the work.

The reports that have been produced during the Swedish pilot are the following:

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

The entirety of the Swedish pilot study reports are provided as appendices to this report, see Appendix E.1 to E.5.

5.2.3.1. Methodology Specification

The Methodology Specification report outlines the methods that are employed in the pilot study. The report details the models and assumptions that are used by the software that was chosen for the study. Details from the Input Specification report are presented in Appendix E.4.

5.2.3.2. Application and result interpretation

The final report in the Swedish pilot study is the Application and Result Interpretation report (Appendix E.5). This report describes the result of the study, as well as the implications of these results. Details from the Input Specification report are presented in Appendix E.5

The Swedish Pilot study looks at a range of Level 3 PSA metrics, health effects, environmental effects, and even economic effects. Looking at different metrics highlights how different elements of the Level 2 PSA or weather input can be important for different metrics. Some of the notable findings are the following:

- A 100 TBq release criteria provides a reasonably good screening of which release categories are likely to cause health effects. Release categories below 100 TBq are unlikely to cause health effects, while those exceeding 100 TBq have a notable risk of causing health effects when applying very conservative assumptions.
- One of the goals of the study was to investigate what can be said of release categories that fall above or below the threshold. One clear finding is that for several of the risk metrics investigated the differences between a release exceeding 100 TBq and those greatly exceeding the 100 TBq (>10,000 TBq) threshold is significant. The contamination metrics were unlikely to cause significant effects unless the threshold was greatly exceeded.
- The study calculates acute health effects and latent health effects in a very simplified manner. Even with refined models the uncertainties for health effect quantification can be quite large as is shown in the SOARCA uncertainty analysis [4]. For this reason it may be recommended to focus Level 3 PSA studies on dose and contamination, especially in simple studies.
- A complete uncertainty analysis, including source term and modelling uncertainties, is not performed in the Swedish pilot study.

Through performing and discussing the Swedish pilot study some benefits of Level 3 PSA that are under-presented were found. First, simply by performing a Level 3 PSA study necessitates additional investigation and scrutiny of the Level 2 PSA study. By performing a Level 3 PSA one must take a structured view of the Level 2 PSA study and its results. Often the interest in Level 2 PSA studies lies in the frequency assessment of "Large Releases" or "Large Early Releases". In this study it was apparent that large releases could have limited or substantial off-site effects where elements such as release timing, release composition, and external conditions can have a substantial impact. Level 3 PSA also provides an interface for the radiological and PSA communities. These groups are addressing similar issues concurrently, both with separate skill-set and insights. Level 3 PSA can serve as a bridge between the radiological analysis and PSA communities which can likely provide other mutual benefits.

There are many places where this study can be expanded. Sensitivity analysis, and the impacts of shielding, and evacuation are essentially fundamental in a Level 3 PSA, but lacking here due to analysis and resource constraints. Ultimately, this is a "generic" study, and therefore it would be difficult to further develop it, and it is perhaps more useful to develop a Level 3 PSA for an actual application, i.e. a specific NPP located at a specific site. Many of the questions that still linger would be better answered by a site-specific, reactor specific study where actual Level 2 PSA data is available and must be applied to a Level 3 PSA. The true impact and benefits to the utility, emergency personnel, and the surrounding population are difficult to realize in further highly general/generic analyses.

The Swedish pilot study was limited compared to some of the expectations at the beginning of the project. Many of the input, modelling and methodology limitations have been expressed in the reports in Appendix E.1-E.5. Despite these limitations many new and interesting insights were made as a result of this work. Due to the fact that it was a "generic" study, and that quite simple and limited tools were used (which required a lot of manual arrangements) a lot of insight was made in the methods and logistics of performing a Level 3 PSA and the calculations that are required.

When developing the results it became apparent that a central limitation of the analysis was due to the generalized input data. These generalizations simplified the methods and in fact made the study manageable despite the limited resources; however, it was difficult to make real-world assertions that would have helped in assessing the utility of the analysis. The use of general source terms from the EPR report provided insight into the organization of the UK EPR Level 2 PSA [5]. It did not allow for the project to develop much needed experience in using or potentially developing release categories based on a Nordic Level 2 PSA.

6. Guidance Document

The overall objective of the entire project has been to further develop understanding within the Nordic countries in the field of Level 3 PSA, the scope of its application, its limitations, appropriate risk metrics, and the overall need and requirements for performing a Level 3 PSA.

The way that the final conclusions and recommendations are presented is through the development of a Guidance Document, which aims to provide clear and applied guidance on Level 3 studies toward regulators, utilities, and Level 3 PSA practitioners based on the conclusions made over the course of the work.

6.1. Structure

The structure of the guidance document is as follows:

- **Chapter 1** provides an introductory discussion on the purpose and need for having a Guidance Document for Level 3 PSA.
- **Chapter 2** gives an outlook on the regulatory framework, guides and standards in the Nordic countries and internationally.
- **Chapter 3** discusses expected benefits, challenges and limitations with performing a Level 3 PSA.
- **Chapter 4** describes the main elements for a Level 3 PSA. In order to achieve a practical guide, the discussion in chapter 4 is based on three cases, made on possible consequences, aiming at covering the spectrum of Level 3 PSA consequences.
- **Chapter 5** provides a high level summary of the Guidance Document at the same time as some of the more important conclusions are highlighted.

The first two chapters of the guidance document provide introductory discussion and significant review of the regulatory structure with respect to Level 3 PSA in the Nordic countries. These are mostly objective discussion on the state of the analysis in the Nordic countries and abroad. The fourth and fifth chapters of the report guidance for performing and interpreting Level 3 PSA is provided, which is based on the experiences gained through the entire project including the pilot studies.

Since Level 3 PSA is not widely performed in the Nordic countries and specific risk criteria and methodologies are not specified, the guidance provided is quite general.

6.1.1. Elements of the analysis

In order to provide some specific guidance, three cases are postulated where the risk metrics of interest are identified:

Environmental risk – Case A:	Size of land area with significant Cs contamination
Individual risk – Case B:	Risk of (early) death to maximum exposed individual (individual risk)
Population Risk – Case C:	Number of lethal cancers (late effects)

Given these cases, considerations and recommendations for each of the elements of a Level 3 PSA are described.

6.2. Conclusions of the guidance document

6.2.1. Generic analyses

In the course of the project, stakeholders demonstrated an interest in the possibility to perform generic Nordic Level 3 PSA analysis, and whether such a study would be satisfactory.

It is concluded that there is ample potential for genericity, since there are only few relevant factors specific to each site (such as source terms, local weather conditions, local demographics, road network, etc.). The number of atmospheric dispersion runs, the weather conditions considered, statistical analyses applied etc. can be shared between sites in a generic way.

Unfortunately, the question about a generic Nordic PSA Level 3 cannot be completely addressed, since different stakeholders have different needs and interests. The non-trivial task of explicitly outlining the purpose of Level 3 PSA should be first decided.

6.2.2. Why a Level 3 PSA?

In spite of its impediments, it is found that Level 3 PSA is useful in many regards because it provides the unique ability to investigate the range of possible consequences and give an estimate of their relative likelihood which is of interest for the assessment of public safety.

The main potential benefits from performing a Level 3 PSA are the following:

A) Complement existing deterministic consequence analyses
Deterministic studies focusing on health effects are already performed with regularity in order to fulfil Nordic nuclear safety requirements. Level 3 PSA can provide further understanding of consequences for the spectrum of PSA derived source terms, and the related uncertainties.

B) Improvement of Level 1 and 2 PSA
In the same way as when a Level 1 PSA is extended to Level 2 it is common that some conservative assumptions, or simplifications, made in the Level 1 PSA need to be revised. The extension to Level 2 will therefore often result in an improved Level 1 PSA. A similar positive effect is likely to occur on the Level 1 and Level 2 PSA when extended to Level 3. This of course depends on the purpose of the Level 3 PSA (refer to the three cases described previously) and the risk metric chosen (health effects, environmental effects or economic impact).

C) Justification of Level 1 and 2 PSA acceptance criteria
As the criteria ("safety goal") often used in Level 1 and 2 PSAs in most cases are surrogates for implied off-site consequences the performance of a Level 3 PSA would provide important insights about the criteria used in the Level 1 and 2 PSA.

D) Communication of risk
Since the Level 3 PSA end-states are related to such risk metrics that non-PSA experts also can understand, the assessment of off-site consequences enables discussing risk with other organizations and groups.

E) Risk comparison
Since the Level 3 PSA risk metrics used are not industry dependent it would be possible to compare risk contributions from different industries and between sites within the same industry. It shall be noted though that care needs to be taken before such comparison is done since differences in scope, assumptions etc. may significantly impact the results which in turn can make one study un-comparable with another even though the same risk metric is used.

F) Link between different experts/communities
Performance of a Level 3 PSA should mean that an interface between the radiological and PSA communities needs to be created which in itself would be positive. Similarly, it is likely that a closer link would be created between utility personnel and emergency preparedness organizations.

6.2.3. Challenges and limitations

The primary impediment to Level 3 PSA is the question of uncertainty. This is also a major concern in Level 2 PSA and to an extent even in Level 1 PSA. In the relatively simple pilot studies, little could be directly addressed with respect to uncertainties. Fundamental questions, which remain to be addressed (such as whether uncertainties prohibit the usefulness of the results) require clear demonstration and presentation of uncertainties.

6.2.4. Analysis considerations

The guidance document covers the subject of Level 3 PSA quite broadly. As Level 3 PSA is not currently required by any regulatory authorities in the Nordic countries, and very few in the world, it was not possible to provide specific (“step-by-step”) guidance. Yet, by breaking down the analysis into the three cases described earlier it has been possible to outline some refined recommendations and considerations, which is not currently available in most existing Level 3 PSA guidelines, e.g. what type of release categories need to be considered for respective case and how to apply countermeasures.

6.2.5. Summary and need for future work

One important conclusion is that different types of stakeholders have different needs and interests in probabilistic off-site consequence analysis. A consequence of this is that the guidance given in this document is more of a brief overview of the methodology with description of a spectrum of possible analyses rather than detailed specific guidance on how to perform a Nordic PSA Level 3.

The following three areas are identified as especially important to investigate in more detail:

- The integration of Level 1, Level 2, and Level 3 PSA.
- Impact and logistics of implementation of countermeasures, e.g. as outlined in the Flag Book [6].
- Deeper understanding of the uncertainties related to a Level 3 PSA and their propagation from Level 1 and 2 PSA.

Even though the Swedish and Finnish pilot studies that have been performed within the project have given invaluable insights to the Guidance Document it is recommended to perform more site/plant specific pilot studies as a method to study the aspects listed above.

It shall also be noted that a Level 3 PSA is most likely even more beneficial when it comes to evaluating site wide consequences (i.e. a multi-unit PSA applications).

7. End-year seminar

On February 14, 2017, the last seminar on the topic of Level 3 PSA within this multiyear project was arranged. The seminar attracted a total of 19 people whereof including the project working group. The type of organizations represented during the seminar was utilities, one regulator (SSM), consultant companies and one insurance company (ELINI) from Sweden (13 participants), Finland (5 participants), and Belgium (1 participant). The main objectives of the seminar were to:

- present international status and progress during 2016,
- present the conclusions (recommendations) in the Level 3 PSA Guidance Document,
- get an overview of SSM's view on possible future requirements (in Sweden) related to level 3 PSA, and
- to have a workshop about the future research and needs for Level 3 PSA

A brief summary of the seminar is given in the following sections.

7.1. International status and progress during 2016

It was noted that there are some interesting development going on, mainly in the U.S. and the U.K.

In U.S work is in progress by USNRC with a full scope Level 3 PSA which is expected to be completed during spring 2018. It was also noted that the work with developing an ASME/ANS Level 3 PSA Standard (ASME/ANS RA-S-1.3) is still in progress even though the progress as such has been somewhat unsteady since the work started in 2004.

In the U.K. there is quite a lot of work performed, for instance related to the Generic Design Assessment (GDA) for the planned new-builds in U.K. It can be noted that the safety criteria in U.K. is of such nature (public risk) that Level 3 PSA is required and in case a nuclear site consists of multi-units this also needs to be taken into consideration. Another interesting development in U.K. is the development of the off-site consequence analysis code PACE which has been developed by Public Health England and is now commercially available (https://www.phe-protectionservices.org.uk/pace). This development is of interest for the Level 3 PSA community since it has been noted during the course of this project that not much development has been made in off-site consequence analysis codes since the 1990'ies.

7.2. Workshop on future needs

During the seminar a workshop was arranged with the seminar participants. The purpose with workshop was to have targeted discussion in what the future needs are in terms of Level 3 PSA. This was addressed by dividing the seminar into smaller groups where the following questions were discussed:

- What would be the most likely (or promising) application of Level 3 PSA
- What type of Level 3 PSA would be most useful and cost-effective to have?
  - Plant-specific, site-specific, or generic?
  - Risk metrics to be considered?
  - Degree of details in terms of:
    - source term categories, weather, emergency preparedness, countermeasures, surrounding population, surrounding environment
- What are the Level 2 PSA development needs in light of results from Level 3 PSA?
  - Release categories, phenomena modelling, probabilistic modelling?
  - Level 1 PSA development needs?
- How can we make use of a L3PSA despite of the uncertainties?
  - Which uncertainties are most important? Which can be addressed?
- Way forward?
  - Given the answers to above questions, outline a roadmap.

The workshop concluded that from the Nordic nuclear industry perspective, in order to progress in the topic of Level 3 PSA a reasonable and manageable scope needs to be defined. It was also suggested that some kind of generic PSA Level 3 could be developed where plant specifics could be added at a later stage. In terms of risk metrics it was suggested to start with those related to environmental impact as these are least complicated to address.

One concern that was raised was that in case more focus should be given to Level 3 PSA it is important that current focus, and resources, on Level 1 and 2 PSA are maintained. As the pilot studies already performed have shown that severe health impacts are unlikely if countermeasures are included it was questioned how much efforts that need to be spent if focus should be on environmental (long term) impacts. It might be more beneficial to focus on economic metrics due to environmental impacts and the countermeasures that need to be applied in order to relocate people and decontamination of the affected area.

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8. References

For the many references for each of the activities discussed in this report, please see the references provided in each of the attached appendices.

[1] Wallin Caldwell, A. Olsson, M. Nordqvist, G. Johanson, J.E. Holmberg, C. Sunde, I. Karanta, Addressing off-site consequence criteria using Level 3 PSA — Phase 1 Status Report, NKS Nordic Nuclear Safety Research, NKS-303, 2014.

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[5] AREVA, UK-EPR, Pre-Construction Safety Report (PCSR), Sub-chapter 15.4 Level 2 PSA, 2012.

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Bibliographic Data Sheet **NKS-386**

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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 report summarizes the developments from four years of work, but focuses on the finalization of a Nordic Level 3 PSA Guidance Document which has been worked upon mainly during calendar years 2015 and 2016. Other activities that has been conducted, and provided valuable input to the Guidance Document, are an Industrial Survey, a study of potential Risk Metrics, a summary of Regulations & Standards, and two Pilot Studies (one Swedish and one Finnish). The main objective of the pilot studies was to gain practical experience that, together with insights from the other tasks included in the project, could be transferred to recommendations into a final guidance document.

During the project, targeted discussions between consultancies, utilities, regulators, and insurance companies on the subject of Level 3 PSA have taken place and at the end of each years working period a seminar has been arranged. The working group has also been engaged in international activities surrounding Level 3 PSA, i.e. the development of the IAEA Level 3 PSA TECDOC and the ANS/ASME Level 3 PSA Standard through the 2016 continuation of the project.

All project reports are provided as appendices to this final report.

Key words: PSA, PRA, Level 3 PSA, Probabilistic Consequence Analysis.

REPORT

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ADDRESSING OFF-SITE CONSEQUENCE CRITERIA USING LEVEL 3 PSA - TASK 0 - FINAL REPORT

INDUSTRIAL SURVEY

Summary:

Task 0 is the first of total five tasks that will be presented in the project "Addressing off-site consequence criteria using Level 3 PSA". Task 0 includes an industrial questionnaire (survey) and a literature study.

The questionnaire was sent out to and responded by nuclear experts (authorities, nuclear industry and consultants) and nuclear insurance companies.

This report summarizes the results from task 0 and replaces earlier report: "Questionnaire 1.0" (2013009:002) and "Compilation report" (2013009:007).

Conclusions, recommendations and prioritizations, based on the results from the questionnaire, are presented in chapter 5.

In chapter 5 advantages and difficulties with risk comparison and the needs for Level 3 PSA are discussed. Also expected advantages with Level 3 PSA are defined together with expected challenges with Level 3 PSA. The challenges discussed are also debated as the reason for deciding whether or not to work with Level 3 PSA.

When discussing the challenges with Level 3 PSA it is stated that to be able to uniform the work with Level 3 PSA suitable risk metrics must be defined and the need for safety criteria's and guidelines must be determined. There is also the question on how to define an unacceptable release and how the results from a Level 3 PSA study should be used.

When discussing the results from the questionnaire in the report one of the responses were that: "The challenges are also the reasons for performing a Level 3 PSA".

R E V I S I O N S

Ver.	*Cause / Reviewed pages*	*Issued by*	*Date*	*Reviewed/Reg.no.*	*Approved*
1.0	New report	MN	2014-02-12	GJ, AOL, JH	LF

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1 BACKGROUND

This report summarizes responses to the questionnaire developed in the project "Addressing off-site consequence criteria using Level 3 PSA", in which the field of Probabilistic Consequence Analysis, often referred to as Level 3 PSA, are explored.

Level 3 PSA provides an assessment of off-site consequences from a radioactive release, which is not limited to nuclear reactor sites. Results from the identification and assessment of accident sequences leading to core damages (Level 1 PSA) and the severe accident and radioactive source term analysis (Level 2 PSA), in PSA level 3 are meteorological data, radionuclide release data, population and agricultural data incorporated to estimate the risks to the public.

The purpose of the questionnaire was to collect base information about current international practices and motivations of utilities and regulators for Level 3 PSA. Even though Level 3 PSA is required only in a few countries, the interest is broader. The increased interest and activities regarding Level 3 PSA is due to the interest in better understanding and characterization of off-site consequences following the findings from the Fukushima accidents, the obligations utilities have from insurance companies and shareholders, and the obligations regulators have to the public's health and safety.

The results from the questionnaire will contribute to the ultimate objective and outcome of the project in total, a guiding document to provide clear and applied guidance towards regulators, utilities and Level 3 practitioners.

1.1 TERMS

PSA	*Probabilistic safety analysis of risk due to operation of nuclear power plants*
PSA Level 1	*Using PSA to identify and assess accident sequences leading to core damages. Results normally given as core damage frequency*
PSA Level 2	*Using PSA to assess the amount, probability and timing of off-site releases due to the accident sequences identified in Level 1 PSA. Result normally given as release frequency*
PSA Level 3	*Identifying and quantifying off-site consequences from the accident sequences analyzed in Level 1 and Level 2 PSA*
RAMA	*Reactor Accident Mitigation Analysis*
Risk perspective	*There are different risk perspectives for a risk, e.g. the risk for nuclear energy is different from an individual or society point of view.*
Risk perception	*Ability to accept risk exposure, e.g. the pilot accepts a higher risk exposure than a person on the ground, when addressing the risk for a plane crash*

1.2 METHODOLOGY FOR TASK 0

Task 0 is the first of total five tasks that will be presented in the project "Addressing off-site consequence criteria using Level 3 PSA". Task 0 includes an industrial questionnaire (survey) and a literature study.

In the project plan for task 0 following four sub-tasks were presented:

0.1 Literature study and development of the questionnaire

0.2 Implementation of the questionnaire

0.3 Compilation of results

0.4 Final report

<u>Literature study and development of the questionnaire</u>

The first sub-task for task 0 included the formation of the questionnaire and for this a literature study was needed. The questionnaire was founded from earlier similar studies and from discussions between the project group and stakeholders.

<u>Implementation of the questionnaire</u>

The implementation of the questionnaire was done by sending out the questionnaire by mail.

The questionnaire was sent out to several organizations and the respondents consisted all from the category of identification, defined in the questionnaire as, "Experts" (authorities, nuclear industry and consultants).

When the responses first were discussed it was clear that it was important to receive answers from a broader public, "non-Experts". Insurance companies were then determined and contacted and given the questionnaire by mail.

A list of all of the responding organization can be found in Appendix 1.

The Nuclear Regulatory Commission (NRC) was not able to give a response to the questions in the questionnaire. Instead their response came in terms of a reference to the, at the time, latest revision of the PSA Use and Development report. Appendix 2 consists of an extract from the report that concerns Level 3 PSA.

<u>Compilation of results</u>

When all answers were received a compilation report were produced. Based on the compilation report a workshop was held.

At the workshop the project group and stakeholders were able to review the answers themselves and the interpretation of them. The participants were divided into working groups to allow for an active contribution from stakeholders and project working group members.

Based on the compilation report the questions and responses were discussed and assessed to generate a final conclusion for each question.

The workshop discussion also included discussions regarding possible and appropriate risk metrics.

<u>Final report</u>

This final report summarizes the responses from nuclear experts and insurance companies and also includes the discussion and conclusions made at the workshop.

The final report will be used in the following tasks of the project.

2 FORMAT OF THE REPORT

The format of the questionnaire is preserved in a similar order for this report. This is done to maintain the overview of the questions.

The questionnaire consisted of four main headlines under which the questions were divided.

1. Risk comparison and development of Level 3 PSA
2. Needs for Level 3 PSA
3. Advantages of using Level 3 PSA and risk communication
4. Challenges with Level 3 PSA

In this report all of the questions from the questionnaire can be found in chapter 4.

For each question the report expresses the compiled answers from each category of identification (experts and insurance companies). From the answers a final conclusions has been made for each question.

In chapter 5 recommendations and prioritization is given due to the answers to the questionnaire.

Chapter 3 presents the literature study, preformed simultaneously to the development of the questionnaire.

3 LITERATURE STUDY

While producing the questionnaire a literature study was done simultaneously. The literature study was performed prior to the development of the questionnaire as well as during the implementation and compilation of the answers to the questionnaire. Input to the literature study was also given by the respondents to the questionnaire and participates to the workshop.

The literature study is presented in appendix 2, with a short introducing text (summary) for each report/study.

4 RESULTS OF THE QUESTIONNAIRE

4.1 RISK COMPARISON AND DEVELOPMENT OF LEVEL 3 PSA

4.1.1 COMPARABLE RISKS

For the development of Level 3 PSA it is of main concern to be able to compare risks.

? Is it possible to compare risks for activities that are society-made? For example comparison between annual risks for automobile accidents, societal risks from nuclear power plants. If so, how?

Final conclusion

In theory yes, if comparable units can be found. However, this might not be able to do in practice since this means that the whole life cycle must be obtained.

Comparing risks for different activities are complex and should be done very carefully and if it is possible to do in a correct way with comparable units.

Risks are obtained from different point of view. Risk comparisons from a society point of view are easier to make than from an individual level.

Expert's opinion:

Comparing society-made risks is possible; the challenge is finding comparable units. Comparable units are for example number of deaths.

When making this kind of risk comparisons it is important to evaluate the risks from various perspectives. For example "voluntary" contra "involuntary" risks as well as risks at individual level contra from a societal point of view and effects in short- and long term.

Making risk comparisons from a societal point of view seem to be easier than at individual level. From a society perspective it is possible to use economic values or number of deaths as comparable terms. It is also possible to compare to any other technology causing health problems. At individual level other factors have to be taken in account, for example the fear for radiation, why risk comparisons from an individual level are more difficult to do.

Risk comparisons with similar outcome can be compared, for example limited to different energy sources, to make relevant comparisons. When doing this it can also be of interest to look at the costs (societal costs) contra the benefits (for society).

When comparing societal risks earlier studies can provide a base (e.g. http://www.psandman.com/articles/cma-appb.htm).

Opinion from insurance companies:

In some cases e.g. taking an airplane or train a probability of a fatality can be calculated. However in a lot of other cases there is too much uncertainty to calculate a probability and as people perceive risks in different ways it is hard to find good comparisons between different types of risk. Accidents with high frequency but with a low number of casualties are perceived as much less severe as accidents with low frequency but with a large number of casualties.

? How can the risk involving a release from a nuclear power plant accident be compared to risks from other types of energy sources?

Final conclusion

A simple PSA Level 3 might not be sufficient when comparing with other society-made risks. In order to make a complete comparison a life cycle analysis covering, e.g., both fuel production and waste management should be included in the assessment. This is possible theoretically but might be complex to perform in practice.

Expert's opinion:

When comparing the risk for an release from a nuclear power plant accident with risks from other types of energy sources the risks can be divided by their possible effects, in terms of effect on health, environment and economic. This was done in WASH-1400. A possible risk metric to use is effects for each group per produced TWh.

When making comparisons on effects from different energy sources a life cycle analysis (LCA) can be used. This has been done by Vattenfall as well as energy production declaration (EPD) for Forsmark (see www.klimatdeklaration.se)

Opinion from insurance companies:

Making a comparison is difficult, mainly because e.g. it is unclear too which extent human produced CO_2 contributes to global warming. Also the impact of other emissions on human health is basically impossible to calculate.

The respondents, on the other hand, questioned if these types of comparison are important. For an insurance company, insuring only nuclear energy production, there is no real interest to compare the result with other forms of energy production.

4.1.2 RISK COMPARISON

You are constantly exposed to a wide range of risk and as earlier stated it is important to make relevant comparisons when comparing risks.

? What are reasonable risk comparisons to the risk of a release from a nuclear accident (such as Fukushima)?

Final conclusion

The answers from the respondents to the questionnaire varied significantly, and the opinions at the workshop varied with respect to the Risk Comparison.

The respondents from insurance companies felt that the risk from nuclear should not be compared to those of non-nuclear industries. This was because the insurance provider worked exclusively with the nuclear industry, but also because of the perception that nuclear risks are "different" than other types of risks. Respondents from utilities had mixed thoughts on this issue. Some felt that it is difficult to compare even within the nuclear industry, and that accidents must be assessed on a case-by-case basis, while others felt that in order to understand the risks related to nuclear one must have a point of comparison.

The discussions during the workshop pointed out mixed thoughts on the issue. It was felt that one must baseline the results to something, and this is done in other industries. It was also noted that one must be very careful because it is easy to make ill-advised or unfair comparisons across different industries.

A conclusion from the workshop was that perhaps the primary issue with risk comparisons is actually a problem of risk communication, but any sort of comparison must be made carefully.

Expert's opinion:

In general:

Society-made risks, at an individual level, exist in many activities in the society. The risk for a nuclear accident, such as Fukushima, can be compared to the risks for other disasters such as:

- Natural hazards as tsunamis, earthquakes etc.
- Large oil spills
- Chemical pollution
- Transportation accidents
- General risks for developing cancer
- Risk of other types of power plants

If we want to compare, for example, the risks in terms of number of deaths per year we can calculate the number of deaths from each disaster. By comparing the outcome there is a possibility to grade each risk.

An example can be to take the number of deaths caused by the radiation levels from the Fukushima accident, which will result in a lower number than the number of deaths caused by the actual tsunami. The risk caused from exposure of radiation in hospitals is on the other hand not a good example since this is a “voluntary” risk and the risk for a nuclear accident is an “involuntary” risk. It might also be misleading to compare one single accident out of its context. As pointed out earlier it is important to look at the whole life cycle (LCA).

Specific:

When comparing effects there is a difference in the effects in short term contra long term and therefore the terms that are comparable also differ. Short term (health) effects from a nuclear accident results in small acute costs in terms of human life’s (e.g. the number of direct casualties in Fukushima, Chernobyl etc. are low) in comparison to the number of lives lost from e.g. a tsunami or earthquake. For long term (health) effects the cancer frequency, due to a nuclear accident, can be compared with e.g. health effects due to the burning of fossil fuel, loss of natural resources or to other large scale environmental damages like global warming, acidification etc.

Economic impacts of a nuclear accident are more difficult to assess and to compare to other risks. This is because the economic impacts depend on political decisions: radiation limits on food and housing, potential phase out of all nuclear power in the whole country or even in other countries, etc. The cost of a nuclear accident is thus very difficult to assess.

When looking at the risk for exposure of radiation it is common today to compare it to the “background” radiation, caused by nature as well as man.

Opinion from insurance companies:

As an insurance company we only insure nuclear risks, and therefore we would only compare the PSA level 3 results of one nuclear facility with other nuclear facilities worldwide and not with other (conventional/non-nuclear) risks.

Neither of the areas from the list above is considered as comparable to the risk from a release from a nuclear accident since the effect for people’s health will differ. A release from a nuclear accident can have long term effect for people’s health and lives while other risk that we are exposed to have a direct effect on people’s lives.

4.2 NEEDS FOR LEVEL 3 PSA

4.2.1 PURPOSE

Level 3 PSA provides an assessment of off-site consequences from a radioactive release, not limited to nuclear reactor sites. Outputs from Level 3 PSA can vary; often the outputs include collective radioactive doses, different health effects, economic impacts and agricultural effects.

? What is the main purpose of Level 3 PSA in your opinion?

Final conclusion

At this stage it is difficult to foresee the benefits before the scope for Level 3 PSA has been finalized. Main expected purposes is, however, to use Level 3 PSA as a tool for decision making, e.g. regarding costs for rebuilds and emergency preparedness work. What kind of plant modification is possible and how to act in case of an accident.

Another important purpose is to establish requirements from insurance companies.

Other expectations with developing Level 3 PSA are to:

- Use Level 3 PSA to establish the economical responsibility between different parties and enable a better communication between the nuclear industry, authorities and insurance companies.
- Establish if there should be any requirements for preforming Level 3 PSA from the authorities
- Enable risk communication to third part by creating a tool to be able to communicate between different groups regarding possible risks. Although this should not be seen as a first prioritization.
- Use Level 3 PSA as a tool to gain insights and improve emergency preparedness work

We should not start to perform a Level 3 analysis before the use of the result is defined; to do this the scope needs to be finalized. Finalizing the scope is a purpose in itself.

Expert's opinion:

In general:

Level 3 PSA could be an important tool for decision making, objective guidance. Energy companies need to have an understanding of the societal impact of a potential accident. With PSA level 3 it may also be possible to compare safety benefits versus costs for plant modifications in a more refined way than today.

Working with Level 3 PSA demonstrates that the risk for a nuclear accident is taken seriously and the severity shows responsibility for possible consequences. This could help create a higher acceptance for nuclear power in society, showing that nuclear power is a reasonable way of producing electricity. By identify the impact on individual's life (e.g., in numerical value) the result can then be compared against other risks (identified in a similar manner). Level 3 PSA is required for several activities at the same time in order to do a complete risk assessment.

Fundamental principles:

- To study number of deaths to be able to communicate a correct risk picture to the general public.
- To study economic impacts and agricultural effects (e.g. per TWh). This could be used to prove that it would be possible to create an independent insurance between all power plants without support from the individual governments.
- Principles like where to build nuclear power plants, appropriate number of units at a site, suitable reactor sizes, etc. might also be evaluated with Level 3 PSA.
- Authorities as well as nuclear power plants could use insights from Level 3 PSA for emergency planning.
- Level 3 PSA may give perspectives on consequences of potential antagonistic threats.
- Insurance companies are interested in Level 3 PSA results.

Specific:

In the nuclear context, Level 3 PSA complements Level 1 and 2 PSAs by extending the analysis to the consequences of a nuclear accident.

Specific issues are:

- Optimization and selection of suitable protective measurements for early and late phase of accident
- Contribution to an appropriate and optimized level of protection for people and the environment
- Provide insights in accident management and emergency planning. Current PSA treatment of accident management is limited to prevention of severe accidents
- Contribution to an integrated risk perception and a holistic risk approach (include defense in depth level 5)
- Interpretation of risks on health and loss of property due to radioactive releases.
- To better mitigate risk defined by Level 3 PSA (causalities, health damages) through identifying and understanding them.
- Cost (monetary costs) of the fallout (compared to other types of energy sources) to be able to quantify the cost for a severe accident.
- A level 3 PSA could be of use in a cost-benefit model to motivate plant modifications.

<u>Opinion from insurance companies:</u>

From the responding insurance company point of view the main purpose with Level 3 PSA is to identify the risk exposure for the society, different effects for the general public as a result of a radiological release, e.g.:

- how many people will be affected
- health effects and associated costs
- impact on biodiversity and damage to nature reserves
- how large parts of the surroundings will be affected
- economic effect in the long term
- commercial loss for farmers
- effects on tourism, various industries
- decontamination costs
- financial loss due to real estate depreciation

When the risk exposure has been identified it is of interest to see how well prepared the energy company are in case of an accident.

While doing the analysis hopefully measures will be identified to reduce the risk potential and to mitigate the consequences of a release.

? Have you or your organization used Level 3 PSA for this purpose? If so, how?

<u>Expert's opinion:</u>

In general:

No. The organization responding had, at the time, not used Level 3 PSA in their organization.

Specific:

Some of the respondents had used a type of consequence analysis. In these cases the effects that had been analyzed were mainly health effects apart from one respondent that had performed analysis to measure cost effects.

<u>Opinion from insurance companies:</u>

No. For the responding insurance companies Level 3 PSA has not been applied.

However data for a Level 3 PSA study for a nuclear power plant would be valuable for an insurance company to be able to calculate the insurance premiums.

4.2.2 HEALTH EFFECTS

Health effects after a release of radioactive substances differ. Some effects are noticeable right away, as short terms effects, for example acute radiation sickness. Other effects will be noticed after months, or even years as long terms effects, for example developing cancer. Another problem after the Fukushima accident was the effects that came afterwards regarding, for example, psychical health problems such as depression after losing near ones in the tsunami or not being allowed to farm their land.

? How would you define unacceptable health effects in short- and long term?

Final conclusion

Unacceptable short term health effects are any casualties due to radiation. Acute radiation sickness can sometimes be acceptable, e.g. due to the cause of trying to save human lives. An example of unacceptable health effects in short terms from Fukushima was set to 250 mSv for “too high” in case of emergency actions.

Long term unacceptable health effects are not as easily to determine but one example could be a significant increase in the number of radiation-induced cancers.

The risks for health effects in long term should be compared to other health risks, for example background radiation, air pollution in large cities, radiation dose from air flight etc. An example of unacceptable health effects in long terms could be to compare 10 mSv/y with the equivalent air pollution, e.g. the health effect in terms of ppm of small particles (http://www.euro.who.int/en/health-topics/environment-and-health/air-quality/publications/pre2009/air-quality-guidelines.-global-update-2005.-particulate-matter,-ozone,-nitrogen-dioxide-and-sulfur-dioxide, page 275, table 5).

Note that the discussion regarding health effects does not involve any discussions regarding frequencies. There is a combination of effects and frequency and there is a difficulty with deciding on an unacceptable frequency for an effect, see question 4.4.4. The closest to frequency when discussion health effects are dose limits.

Expert’s opinion:

In general:

When defining health effects, in both short- and long term, this is today defined in national (in Sweden made by SSM) and international (made by IAEA) safety standards. Another possibility to define unacceptable health effects is risks that would significantly affect the overall health risk. This could be compared to, e.g., consequences of nature background or normal people activities. Following effects are examples for terms expressing health effect:

- Casualties (Short term)
- Acute radiation sickness (Short term)
- Risk for developing cancer (Long term)
- Other radiation-related diseases (Long term)

Safety standard, defined by SSM today, for unacceptable health effects in short term are no immediate deaths caused by radiation. This safety standard has been interpreted by Swedish nuclear industry to a dose of less than 1 mSv to any person. This type of dose criteria can be easier to relate to than occurrence of diseases.

Specific:

Limitations in terms of dose levels can be used to estimate unacceptable health effects in both short term and long term, since they include acute damage due to radiation as well as cancer risks.

Safety standards do not only apply protection of humans but also protection of populations of biota. i.e., health effects on biota should also be considered according to international safety standards.

Consequences and health effects can be hard to measure and isolate to cause of consequence, e.g. criteria for unacceptable psychosocial health effects will not be possible to formulate, since death by near ones or other difficulties like unemployment etc. will occur for everyone during a life time due to various factors. Good risk communication with public and other stakeholders, before during normal operation, during and after potential accident can on the other hand decrease potential psychosocial health effects caused by “radioactivity”.

Other possible long term health effects to take into consideration are for example psychological consequences. After the Fukushima accident there were statements in media that the fear for radiation is widespread in Japan and is considered to be a larger problem than the actual risk for developing cancer. For example the children in school were only allowed to play outside a maximum of 2-3 hours per day. The number of suicides for entire Japan increased with 20 % due to homelessness, relatives who died in the tsunami, farmers who lost their income etc.

<u>Opinion from insurance companies:</u>

A health effect that is not acceptable could be defined as all kinds of health effects that require a visit to hospital and would not exist if the accident would not have happened.

The general public should not have any adverse effect from the operation of a nuclear power plant.

Examples for unacceptable effects in short terms are effects that are immediately visible after the release (within e.g. 1-2 weeks) e.g.:

- Early fatalities
- Acute radiation sickness

- Burns

Examples for unacceptable effects in long terms are effects that would not be directly visible but would occur after a long period (several months) e.g.:

- All types of physical or psychological consequences that can be connected to the accident, e.g. induced stress because of uncertainty whether or not having been exposed to increased radiation levels and its impact
- Cancer and other diseases

A difficult aspect in this regard is to evaluate the impact of small doses on human health. The effect of small doses is unclear at this stage.

4.2.3 ENVIRONMENTAL IMPACT

The impacts on the environment after a release of radioactive substances can last for a long period of time. Land contamination can result in restrictions for land use over many years. For example there were elevated levels of cesium measured many years after the Chernobyl accident in meat from reindeers, mushrooms and berries.

? How would you define unacceptable environmental impact?

Final conclusion

Short term environmental effects, e.g. evacuation, could be defined as unacceptable.

Contamination of land areas should be used as the parameter for defining unacceptable environmental impact in long terms. Different areas of contamination can have different restrictions depending on the level of radiation, e.g. areas were one cannot live or areas were one can live but not farm or harvest, and so on.

Expert's opinion:

In general:

Criteria and dose limits have been set for example various foods, based on risk estimates for humans, which reflect unacceptable environmental impact.

When defining unacceptable environmental impact various factors can be estimated, see list below. To be unacceptable the effects needs to be in long terms (more than a couple of months) and affect a large area (larger than the plant site and close surroundings).

Examples for environmental effects:

- Evacuation
- Contamination of land/sea
- Restrictions in land use (for example loss of agricultural and forestry land)
- Wide change of food consumption behavior
- Damage to biosphere(for example impacts on animals and plants)
- Irreversible loss of protected (red listed) species, biotope or natural resource

Specific:

Some environmental effects could be acceptable. For example long term effects in a small area surrounding the plant.

Environmental impact often leads to economic impact why other studies have converted environmental effects to monetary value, for example business interruption in activities like fishing, tourism, food production, why these two impacts not always have to be separated for one another.

Opinion from insurance companies:

An environmental effect can be defined as unacceptable if the land is contaminated or if there are restrictions in land use and that would not exist if the accident would not have happened.

Any damage to land, nature reserves etc. should be compensated for (by restoring the land to the same condition as it was prior to the accident. If such is not feasible by other means of compensation e.g. money or other land being given to the people who lost their land.

4.2.4 ECONOMICAL IMPACT

After the Fukushima accident the financial strain became large and the economic compensation uncertain. Uncertainty concerning for example what off-site consequence is to be compensated from what part of the society (e.g. community or industry).

? How would you define unacceptable economic impact?

Final conclusion

From the answers to the questionnaire it can be noted that insurance companies are interested in the economic considerations for Level 3 PSA, while the Nuclear Safety Authorities were somewhat less interested in the question of economic impact.

Defining unacceptable economic impacts is difficult to define in general and would also imply political elements.

One way of separating economical risk could be to define the economic effects on the plant organizations to be acceptable, while economic effects on third parties outside the plant organization are unacceptable.

Expert's opinion:

In general:

Economic impacts are hard to foresee because they depend on political decisions, e.g. radiation limits on food and housing, as well as health and environmental effects also results into economic effects. Economic impact should be based on the sum of the effects.

One way to define unacceptable economic impact is when the "bills" are higher than the economic preparedness; the organization is not able to pay the "bills" either as part of normal operation or through an insurance solution. If Level 3 PSA could help by assessing and quantifying potential economic impact there could be a better way to prepare for such an event.

Specific:

Since the economic impacts depends on political decisions a reduced dose criteria would lead to economic costs increasing rapidly.

The economical strain from a nuclear accident could be helped by better insurance solutions (that should be able to cover the losses outside the licensee) or by a collaboration among a large number of reactors. Extreme solutions like the need for reconstruction of the societal budget could then be reduced in case of the event for a nuclear accident.

Opinion from insurance companies:

The financial loss for the company that operates the nuclear power plant and the loss of income due to that they cannot supply power during a long time period is acceptable. Since they operate the plant, they "own" the risk and this should have been taken into account. Any financial loss or incurred expenses (such as due to evacuation) suffered by companies or individuals as a result of the nuclear accident should be compensated for by the nuclear operator (when his facility is at the origin of the loss).

An unacceptable economic impact can be defined as costs related to third parties such as compensation to people that have to evacuate and to move from their homes. This also includes costs for decontamination and salvage operations. The taxpayer should not be called upon to pay for the damages.

Performing a Level 3 PSA study can provide a more accurate estimation of the costs, as a result of a severe accident. More liable insurance limits can be reviewed for adequacy and be increased if necessary. Compensation will however be difficult to calculate in some cases. Financial loss suffered by the utility can be insured through a property insurance policy. It is up to the utility to decide whether or not to purchase such insurance coverage. If not purchased, the utility should not be able to claim for compensation (from the taxpayer e.g.). Shareholders are aware that stock prices can fall heavily after an accident. Such did also occur after the financial crisis. Shareholders cannot be compensated for this loss (inherent risk that shareholders should accept if they buy shares).

4.3 ADVANTAGES OF USING LEVEL 3 PSA AND RISK COMMUNICATION

4.3.1 BENEFIT IN THE COMMUNICATION WITH DIFFERENT STAKEHOLDERS

Using Level 3 PSA might enable different stakeholders (see figure 1) to communicate about risks.

? Are there any advantages of using Level 3 PSA for communication with or between different stakeholders regarding the societal risks of commercial nuclear power?

Final conclusion

Main expected advantages with Level 3 PSA are:

- Defining risks (in comparable terms) and calculating the risks (e.g. in monetary values) to make them communicable
- Better understanding for consequences of a nuclear accident and thereby improve emergency preparedness work

Defining risks in comparable terms and making them communicable between the stakeholders is difficult, see question 4.2.1, purpose.

Expert's opinion:

In general:

Yes, there are advantages with using Level 3 PSA for communication between stakeholders regarding risks to society with nuclear power.

From a Level 3 PSA study economic impacts can be defined, needed to be able to communicate with insurance companies and making proper risk assessments. Other effects, effects on health and environment, are needed to be able to communicate with the public. This could also have an effect to create an opinion regarding new nuclear power plants.

Better understanding of the consequences with a nuclear accident also benefits the stakeholders, the utilities.

Defining (in comparable terms) and calculating (e.g. in monetary values) risks makes them communicable.

Specific:

The support from authorities, e.g. SSM, is important when working with Level 3 PSA and the results from the analysis needs to be comparable between utilities, within the nation and at an international level.

One of the respondents saw no advantages with using Level 3 PSA for communication between stakeholders. The respondent felt that evaluating the risks with nuclear power can be done with simpler calculations than the ones needed in a Level 3 PSA study.

Opinion from insurance companies:

Yes, there are advantages with using Level 3 PSA for communication between stakeholders regarding risks to society with nuclear power.

Advantages with results from a Level 3 PSA study, regarding better communication that the respondents saw were:

- Proof to the general public that nuclear power companies are able to carry the cost of a radiological release themselves and will not shift the cost to the taxpayer.
- Proof to the general public that nuclear power production is safe and thereby increase public acceptance of nuclear energy
- An increased acceptance could lead to greater public support to build new nuclear power plants
- A clear compensation schemes could lead to the people feeling less unsure about economic consequences
- Showing the risk exposure of the operation should help the insurance company to be able to understand and put a value on the risk
- Communication with authorities to verify that the company that operates the power plant are aware of the effects in case of a release scenario and that they have a relevant plan for how to handle their responsibilities.
- Clarify the roles and responsibilities of all parties involved. The authority should use it to have a plan for handle their part in case of an accident.

4.3.2 ASSESSING THE IMPORTANCE OF DIFFERENT COMMUNICATIONS PATHS

The way of how to grade important communication paths may differ between different groups and/or persons. Different communications paths may vary in importance for different parts of the society. For example, media may rely on information coming from the government while information between two private persons can be of equal importance for an individual person.

? Scale the most important communications paths (from your point of view) in the matrix below on a scale 1-5 (1 being least important and 5 most important). Mark at least five different communication paths.

In the questionnaire five groups of identification were defined, see figure 1.

<u>Expert's opinion:</u>

The figure below represents the expert's opinion in general. The number in each box represents the mean of the answers.

From → To↓	Experts	Authorities and Government	Media	Health and Envorinment	Private person
Private person	3	4	3	3	2
Health and Envorinment	3	4	1	2	
Media	3	4	2		
Authorities and Government	4	3			
Experts	3				

Red=Important, Yellow=Medium, Green=Not so important

Opinion from insurance companies:

The figure below represents the responding insurance company's opinion in general. The number in each box represents the mean of the answers.

From → To↓	Experts	Authorities and Government	Media	Health and Envorinment	Private person
Private person		5	3		
Health and Envorinment	5				
Media	3	3	1		
Authorities and Government	2	3			
Experts	3				

Red=Important, Yellow=Medium, Green=Not so important

? Which is the most useful communications path?

Final conclusion

One benefit of Level 3 PSA is to create a tool that can help to provide communication between different stakeholders.

The most important communication path consists of two parts:

1. From experts to authorities
2. From authorities to everybody else (private persons, non-governmental organizations, media).

However, the authorities (STUK and SSM) are in a double role because they are both experts and authorities. Communication by authorities is more important than communication by experts.

Expert's opinion:

In general:

The communication from experts to authorities and government are important. The communications from authorities and government to the public, either directly or via organizations, or media, are another important communication path.

Specific:

In general, all paths from experts to the other subjects (non-experts) are important. The most important communication path can differ regarding on the purpose of the communication and different stages, for example the most important communication looks different after an accident.

Bi-directional communication is needed.

<u>Opinion from insurance companies:</u>

The most important communication path, for a private person, to provide trust and a feeling of responsibility of the given information is from authorities and Government to other parties. Media can play a similar role here but there is a much lesser degree of trustworthiness.

Also the exchange of information between experts is useful to make a good risk analysis.

One of the respondents personally found the stress tests after Fukushima very useful in order to show that an extensive risk analysis has been done and hence to reassure the general public of the safety of nuclear power plant.

1. Private person (student or other)
2. Health and Environment (e.g. environmental organizations, medical team, non-profit organizations)
3. Media
4. Authorities and Government (e.g. Swedish Transport Administration, MSB (Swedish Civil Contingencies Agency), parliament, politicians, insurance company)
5. Experts (SSM, STUK, IAEA, consultant, plant experts, shareholders and board members)

Figure 1 Defined categories of identification from questionnaire

4.3.3 GENERAL BENEFIT TO THE NUCLEAR COMMUNITY

One of the aims with Level 3 PSA is to better understand societal risks.

? How can nuclear community in general benefit from a Level 3 PSA?

Final conclusion

Level 3 PSA could help the nuclear community to communicate with insurance companies and the analysis could possible even help reduce the costs for the nuclear industry.

Level 3 PSA could also help the nuclear industry to communicate with the society in large, if the nuclear industry so wishes. It is not yet stated out whether or not all the stakeholders want to communicate the results?

The results from a Level 3 PSA study could also be compared between different existing sites.

Comparing the results between existing sites should however been done carefully since there already are differences in PSA Level 1 and 2. Site specific and not unit specific Level 3 PSA should be performed when considering external events.

Expert's opinion:

In general:

The benefits for the nuclear community by performing PSA level 3 can be:

- Better understanding of societal risks of commercial nuclear power
- Improve preparedness work
- Improved risk communication in terms of understandable and clear communication
- Better insurance possibilities
- Create acceptance for nuclear power in society
- Better design and siting considerations for new construction projects
- Cost benefit metric for plant retrofits

Specific:

A common methodology or even a common PSA Level 3 report for a large group of units (e.g. all Nordic utilities) would be preferred.

Using Level 3 PSA as a tool to compare effects between different plant locations, and thereby making better judgment according to site considerations, comparison can be extremely difficult. Comparison between levels 1 and 2 cannot be done due to differences in approaches and methodology for individual plants.

Opinion from insurance companies:

The benefits for the nuclear community by performing PSA level 3 can be:

- Better design and siting considerations for new construction projects.
- Risk informed severe accident response procedures
- Identifying measures to reduce the risk potential as well as to mitigate the consequences of a release

4.3.4 BENEFIT FOR THE USE OF PSA LEVEL 1 AND 2

The main aim with PSA is to identify and prioritize safety issues.

? How can Level 3 PSA improve the benefit of PSA Level 1 and 2?

Final conclusion

Level 3 PSA has a significant value in that it is a tool that can provide lines of communication to different groups that are not currently possible with the Level 1 and Level 2 PSA.

However the responses to the questionnaire varied for this question, especially between utilities and safety authorities. The responses that were "favorable" to Level 3 PSA provided elements that can be beneficial, the organizations that were more skeptical toward Level 3 PSA thought that the benefit to Level 1 and 2 would be very minimal.

From an authorities point of view the benefit to the earlier PSA analyses is in having a "holistic" view of power plant risk, while the other authority felt that it was not necessary to perform Level 3 PSA and therefore has little positive impact on Levels 1 and 2.

Insurance companies did also see the possibility for improvement to the earlier PSA with Level 3 PSA.

Expert's opinion:

In general:

Possible improvements for PSA Level 1 and 2 with performing a PSA Level 3 study:

- Risk informed severe accident response procedures
- With a holistic point of view it is possible to improve risk management and the interpretation of PSA level 1 and 2 safety goals
- Reducing conservatisms as well as optimistic assumptions in PSA level 1 and PSA level 2 by harmonization
- Enhance the validity or conclusions regarding the most important risk contributions reached in Level-1 and Level-2 PSA

Specific:

Whether or not Level 3 PSA can improve the benefit of PSA Level 1 and 2 is not uniformly clear from the respondents' answers. At an existing plant the improvement can be seen as pretty low, thinking that Level 1 and 2 can be enough. Some of the respondents thought that the effects of a release already are well known, some see Level 1 as a complete analysis but that improved definition of release categories and defining risk metric and risk criteria can help improve Level 2 analysis.

Level 3 can tighten the requirements for level 1 and 2, but in the same way also reduce the requirements, depending on the outcome from Level 3. To be able to use Level 3 it is required that the results in Levels 1 and 2 are accurate and precise (although these analyses are built on probabilities and therefore will include uncertainties).

Opinion from insurance companies:

Some actions taken to reduce the Level 3 PSA may also have a benefit for the Level 1 and 2 PSA. It may also justify (to shareholders) the cost to realize certain upgrades.

4.3.5 OTHER COMMENTS ON RISK COMMUNICATIONS

Expert's opinion:

If the results from a Level 3 PSA study can be presented in an understandable way it would help risk communication in a positive direction. In the same way risk communication can give the possibility to find optimally protective measurements.

Risk communication to the general public can, with the help of Level 3 PSA, help to make comparison of "complete" NPP operation risk and perhaps prove that, objectively, this risk is generally much lower than some other risks.

On the other hand making risk comparisons between factors that are not comparable can give the wrong impression and seem like nonsense when expressing them to the public. If Level 3 PSA is used incorrectly there can be risk for large uncertainties, especially if it is performed individually at each utility. Also if the analysis is difficult to understand or explain risks expressed in probabilities can be difficult to communicate.

Opinion from insurance companies:

Respondents from insurance companies had no comment here.

4.4 CHALLENGES WITH LEVEL 3 PSA

4.4.1 RISK PERCEPTION

Using Level 3 PSA might be misleading depending on how we look at risk and how the risk is perceived.

? What are the obstacles in using Level 3 PSA in your opinion?

Final conclusion

There are many kinds of uncertainties involved in Level 3 PSA: those coming from Levels 1 and 2 PSA, and also uncertainties from Level 3 PSA itself. Presenting this uncertainty in consequences to media, decision makers and the general public is challenging and requires responsibility.

Performing Level 3 PSA requires a lot of work and there is also a risk for a large gap in time between preforming Level 3 PSA studies which leads to problems with knowledge transfer.

To be able to express something meaningful from the results from a Level 3 PSA study to the public the result should express the risk of exceeding an impact assessment threshold, e.g. the max acceptable consequence limit 100 TBq, with all uncertainties integrated into a best estimate of the mean.

Notice, in the discussions on the obstacles when preforming Level 3 PSA it was also stated that: “The challenges are also the reasons for preforming a Level 3 PSA”.

Expert’s opinion:

In general:

Possible obstacles when using Level 3 PSA are:

- uncertainties in the analyses, there is a possibility that uncertainties increases from Level 1 to Level 2 to Level 3 PSA
- uncertainties when working with probabilities, there are for example no “standard weather” and reality often differ from models, accident can happen in unforeseen ways
- uncertainties that comes with a lot of ingoing parameters, many of the parameters are also subjective
- screening criteria for Level 1 and 2 PSA might not be suitable for Level 3 PSA
- difficult to make comparisons between different reactors

Aside from this there is also, as earlier discussed, difficult to communicate about risks and there are different risk perceptions to take into consideration. There can be an obstacle to value emotions and then to be consistent between different interest groups, communities etc. It is important to try and define acceptable risks for the society.

The result from Level 3 PSA needs to be presented in an objective and suitable manner because the same risk management approaches is not used in all parts of society and could lead to misunderstandings and misinterpretations concerning nuclear power.

Specific:

Some of the respondents expressed a concern for the obstacles with Level 3 PSA to be extensive, possible more extensive than the advantages. One fear is that the analysis method will be too expensive to perform and that it therefore will be hard to perform a study that is large enough to truly be used for risk decision making.

On the other hand the advantages with Level 3 PSA can also be large. One of the (unique) advantages that Level 3 PSA can provide is the possibility to compare negative impacts from different technologies. There is also a possibility to see the uncertainties with Level 3 PSA to be, in fact, one of the reasons that we need the analysis method. Level 3 PSA is needed due to the uncertainties.

This different point of view is important to take into consideration when deciding whether or not to work with Level 3 PSA.

Opinion from insurance companies:

Possible obstacles when using Level 3 PSA are:

- The risk with comparing the risk for a nuclear accident to other types of risks, since the effect of a nuclear release differs from other type of accidents. The respondent in question did not see the need for the risk to be compared with other scenarios. PSA Level 3 should be done to increase the knowledge about different scenarios to face risks that we are exposed to. In that way the nuclear industry and the society can take proper actions and make plans to be used in case of an accident.
- The amount of uncertainties. Uncertainties can make the result feel less meaningful and therefore the level of assumptions should be kept to a minimum.
- That the results might not be very meaningful for a member of the general public. The only message such person wants to receive is that nuclear energy will not have any effect on his/hers health, life etc.

4.4.2 RISK METRIC

Appropriate risk metrics is one of the main questions when developing Level 3 PSA. This project will have the possibility to contribute to the development of Level 3 PSA and highlight important aspects like: What are the proper risk metrics? How should we look at risks and how should they be graded?

? What kind of risk metric is suitable in a Level 3 PSA?

Final conclusion

The complete risk metric is the economic risk metric, since it will cover all the aspect of the risks, but it is the hardest one to use. Too much work to get it realistic due to difficulties to put economic value for the consequences.

Doses and contamination of land are also a possible appropriate risk metric. It is relatively easy to calculate fatalities from these metrics.

The points of interest depends very much on the purpose of the analysis, and it can be envisioned that for an insurance provider the economic analysis would be important, and relevant economic metrics would be of interest, while for authorities some other risk metric could be of greater interest.

At this stage of the project it can be difficult to decide which risk metrics is the most suitable. All risk metrics have to be discussed regarding pros and cons, limitations, uncertainties and purpose. During the pilot study more can be elaborated in connection to risk metrics. The criteria to be applied, the risk metrics, should be application specific, and therefore the scope of Level 3 PSA needs to be defined.

The purpose of performing Level 3 PSA should lead to increase the safety for a nuclear power plant and the risk metrics should reflect this work, decreasing risks at the plant. Monetary value is easily understood, hence communicable; the problem is to decide a monetary value for each risk.

Expert's opinion:

In general:

Suitable risk metrics can be divided based on the possible risk effects; health effects, environmental effects and economic effects, in both short- and long-term. Examples for each type of effects can be found below. To present risk assessment results in terms of:

- short term effects (radiation levels, dose levels)
- long term effects (health effects, social effects)
- economic impact (for example ground contamination, cost of evacuation, cost of lost production)
- environmental impact

Risk metrics regarding environmental effects can be presented in terms of land area considered to be lost for a long time. Presenting risk assessment in long term effects (and not just short term effects) would include cumulative effects as well. Health effects (in terms of number of deaths/cancers) are easier to determine than environmental and economic. The effects in terms of environmental and economic will also have a higher level of uncertainty.

Specific:

Different risk metrics is suitable for different parts of the society depending on the target group. For example, health effects in terms of frequency of deaths or number of cancer can be suitable if the target group is the public but the economic effect on the other hand is of greater interest when measuring the risk for a power plant organization.

A possible risk metric is to present the (health-) effects in terms of frequency per produced TWh. This makes the risk metrics possible to compare to other types of energy sources.

When considering health effects (short- and long term) the reference values used in safety standard should be used. For short term effects this means no immediate deaths caused by radiation and for long term health effects SSM has developed limits for non-acceptable land contamination and radiation doses. Dose criteria's are easier to relate to than occurrence of diseases.

Opinion from insurance companies:

A level 3 PSA should both give an indication of the effective dose and dose rates people will be exposed to after a release. Health effects from high doses are well known but this is much unsure for low doses. Financial losses, decontamination costs etc. should be estimated as accurate as possible in order to have a (rough) estimation of the total loss amount caused by a nuclear accident.

Suitable risk metrics are listed in bullets above.

4.4.3 SAFETY CRITERIA

Analogous to Level 1 PSA and Level 2 PSA there is a discussion whether safety criteria are also needed for Level 3 PSA. A possible way is to attach a numerical value to other risk of society. For example, general accidental death risk of an individual is on the level of about 1E-4 per year. From this numerical value it may be possible to decide how much less the risk from a radioactive release should be. Often a factor of 100 is used, resulting in the number 1E-6 per year. This leads to a safety criteria for individual risk from radioactive release to 1E-6 per year.

? Are there any needs for Level 3 PSA safety criteria?

<u>Final conclusion</u>

The respondents to the questionnaire were divided in whether they thought safety criteria were required or not. Those that felt they should not be required felt that Level 3 PSA has not been performed or applied enough to define such criteria. Those that felt criteria should exist were interested in using them as a means of defining the scope of the analysis.

The outcome from the discussions during the workshop was that safety criteria must be defined in order to understand the results of a PSA. One needs such criteria to understand if the results are good/bad or acceptable or unacceptable. This provides focus to an analysis.

The safety criteria’s should be the same for old and new plants.

<u>Expert’s opinion:</u>

In general:

Yes. Most of the respondents (60 %) saw needs for defining Level 3 PSA safety criteria. With the help of safety criteria a general (national and hopefully international) agreement on acceptable risks could be defined. This can then be used by politicians etc. By being able to combine individual and societal risks the analysis thereby makes the results communicable.

Specific:
In some cases that the respondents answered "No" regarding whether or no safety criteria’s are needed for Level 3 PSA. Some of the reasons for this was that it is too soon in the development of Level 3 PSA to decide on suitable safety criteria’s. Other comments were the concern that possible criteria’s for Level 3 PSA would be misleading or useless due to uncertainties, differences in models and very low probabilities.

<u>Opinion from insurance companies:</u>

Yes. The respondents saw needs for defining Level 3 PSA safety criteria.

? **Examples for possible safety criteria's for Level 3 PSA**

Expert's opinion:

Multiple safety criteria could be used, for individual and societal risks with an ALARA approach.

There is also a possibility to use the safety criteria regarding RAMA.

Criteria based on health effects are preferably based on values of comparable parameters taken from analyses of other technologies operated on high level of risk. This approach may be better than to use some general accidental risk value, which is connected with very high level of uncertainty. It may be reasonable to define the maximum quantity of released radioactive substances applied in the criteria regardless of size and power of the reactor.

Opinion from insurance companies:

The respondents did not have a direct suggestion other than that the criteria should be expressed in terms to give guidance. The total loss amount should not exceed the purchased amount for Third Party Nuclear Liability (TPL nuclear liability).

As indicated previously, the number by itself may not have a lot of value. Any clear improvements which can be taken to reduce the impact of a radiological release should not be deemed unnecessary just because the Level 3 PSA result meets the criteria.

? **RAMA addresses the same requirement regarding the maximum quantity of released radioactive substances, applied to all reactors regardless of size and power. When defining safety criteria's in Level 3 PSA are there any needs for separate targets for old vs. new plants?**

Expert's opinion:

In general:

Most of the respondents saw no need for separate targets for old vs. new plants. Level 3 PSA results depend significantly on the attributes of the locality and quality of accident management so the design of the plant, in terms of differences between "old" and "new" plants, is (relatively) not that big.

Specific:

Even though the targets would be same for old vs. new plants, the way targets would be used could be certainly different between old vs. new plants, see "Safety Goal" project's reports.

Opinion from insurance companies:

The respondents had a different opinion. One of the respondents saw needs for separate targets. But the other did not see a reason why an old plant should have a different risk from a new plant and as a nuclear insurer do not make a distinction between an old and a new plant.

4.4.4 UNACCEPTABLE RELEASE

An unacceptable release can be defined in many ways in Level 2 PSA. It can be based on the emission size and content, time or place of the release, number of acute deaths, economic impact and so on.

? How would you define an unacceptable release?

Final conclusion

Defining unacceptable release for Level 3 PSA should be related to how we define unacceptable release in Level 2 PSA. Then there are some factors that need to be defined.

First acceptance criteria needs to be defined. Example of acceptance criteria's are ALARA (not zero but "as low as reasonably achievable") were we accept some "acceptable" risks.

Regulations from the authorities in term of dose criteria's are also one kind of acceptance criteria. Dose criteria´s are a combination of dose and frequencies.

Reference values also need to be defined. One example on how to define a reference value is to define a limitation for the background radiation from normal operation of a NPP to be 1/10 of the "natural" background radiation, and them limiting all accidents to 1/10 for the background radiation from normal operation (1/100 of "natural" background radiation).

Note that there is a risk that the work with minimizing the risks could be held up if we define what risks that is acceptable.

In parenthesis:

One plant might define their unacceptable releases different than another plant regarding on where the plant is located.

Can we really accept that, for example, a plant placed in the north is less safe than a plant located in a site closer to a larger population?

The site for a power plant cannot be integrating when looking at a serve accident because their impacts will affect larger areas then the site and its nearest surroundings. The effects are not site dependent since the effects will be so large.

Site considerations might however be taking in to account when constructing a plant. But in case of severe accident site considerations are not applicable.

Expert's opinion:

In general:

In Sweden a government decision says that no more than 0.1 % of the Cesium (Cs-137) inventory of a core of the size of those at Barsebäck NPP (1880 MWt core) may be released. Several of the respondents saw this as a good enough definition of an unacceptable release.

Other examples on how to define unacceptable release, that corresponds to the decision above:

- 100 TBq Cs-137
- Dose larger to than 100 mSv to the persons living nearby.
- Collective dose (average dose) larger than 10 mSv/year.

See also answers to questions 4.2.2 and 4.2.3.

Specific:

Unacceptable release could also be defined so that it would be comparable to risks from other types of energy source.

Definition on unacceptable release should be dependent on meteorological conditions during release and the site and demographic situation.

A comment on the decision from Swedish government (0.1 % of the Cesium inventory of a core) is that it might be more correct to allow a higher release for a plant with a higher power level since it might correspond to several smaller plants (and allow a lower release for smaller plants).

<u>Opinion from insurance companies:</u>

Any release which will have an adverse effect should be deemed unacceptable. As the impact of low dose rates, low activity releases etc. are unsure it is difficult to define which level is acceptable and which is not. This has to be defined by experts and should be adjusted whenever new (reliable) data is available.

4.4.5 USE OF RESULTS

Level 3 PSA can allow for better communication between stakeholders and give a general benefit to the nuclear community.

? How can the results be used and to what purposes?

<u>Final conclusion</u>

Two main uses were discussed:

- To the public, the results can be used in communication, if used carefully (e.g. the acceptable consequences).
- To experts, the results can be used in planning (e.g. emergency planning, accident management).

The starting point for developing Level 3 PSA should be the intended use of the results.

<u>Expert's opinion:</u>

In general:

Level 3 PSA can be used by communicating the results to a broader public. The analyses can also give a better view on the effect of a specific release. When the possible effects are determined the results can be used to:

- Contribute to an appropriate and optimized level of protection for people and the environment
- Provide insights in accident management and emergency planning
- Contribute to an integrated risk perception and a holistic risk approach
- Choose suitable protective measurements, for early and late phase of an accident

Specific:

However some of the respondents did not see the use of Level 3 PSA results for analyzing the effects. According to the respondents the effects could be defined from predefined releases, and thus Level 3 PSA would not be required.

Possibility of misinterpreted results from a PSA Level 3 study is similar to the way as for Level 1 and Level 2 PSA results.

<u>Opinion from insurance companies:</u>

PSA Level 3 can be used to increase the knowledge about different scenarios to face the risks that we are exposed to. In that way the company and the society can take proper actions and make plans to be used in case of an accident.

The pure numerical value should not be used to implement easy and relatively cheap upgrades. A requirement from regulators to perform a PSA

Level 3 study, the respondent feel, is needed. At the same time the respondent acknowledges that a PSA Level 3 analysis might lead to extensive costs, which leads to the need for nuclear liability coverage.

4.4.6 GUIDELINES

There is an ongoing work regarding the peer review standards (ANS/ASME 58.24 (PSA Level 2) and ANS/ASME 58.25 (Level 3 PSA) that are currently being developed. This project will have the possibility to influence of the progress of these standards.

? Are there any needs for Level 3 PSA guidelines?

Final conclusion

It may be too early to define if a guideline is needed or not. First we must decide if we need Level 3 PSA or not. If we need it we need guidelines.

The guidelines should be written as suggestions rather than a strict guide line. The guideline should give some input on different ways of performing Level 3 PSA depending on the objectives. Use of international guideline and specify the order of detail used in Nordic countries.

Expert's opinion:

In general:

Yes. The responding organizations thought there were needs for Level 3 PSA guidelines. It is important since the analysis area is relatively immature that the overall purpose, methodology, risk metrics and use of results should be alike.

In fact some of the respondents saw the need for guidelines for Level 3 PSA to be greater than for Level 1 and Level 2 PSA. One reason for this is that result from a Level 3 PSA study needs to be comparable.

Guidelines for Level 3 PSA can give help to:

- Reduce the work for each user (analysis job and experience feedback)
- Harmonize Level 3 PSA methodology
- Communicate between authorities and the public
- Increase the trust in the analyses
- Give balanced understanding on risks of different nuclear power plants

Whether or not there are any needs for separate guideline for old vs. new plants the respondents did not see this as necessary. The results will mainly be based on the site than the design for a plant.

Specific:
Some of the respondents (20%) did not see any needs for Level 3 PSA guidelines. A reason for this was that it is too soon to discuss this need. More work with Level 3 PSA must be done before this can be determined. First of all this pilot project needs to be finalized before it is possible to decide if we (the project) need to develop a specific guidance ourselves.

<u>Opinion from insurance companies:</u>

Yes. The respondents thought there were needs for Level 3 PSA guidelines.

Since there are so many ways to perform analysis and evaluate results it would be good with guidelines to ensure that scenarios from one plant can be compared with scenarios from another plant. Without guidelines there are risk for nuclear power plants to, for example, make a lot of assumptions or exclude some parts of the study in order to simplify things (to reduce cost).

Guidelines can give a good overview of how the total loss after a nuclear accident can be calculated.

4.4.7 OTHER CHALLENGES WITH LEVEL 3 PSA

Final conclusion

The other challenges suggested by the respondents to the questionnaire varied significantly, and some were quite specific. Under the discussions the challenges were divided into following themes:

- Scope & Scope Definition
- Results, result formatting (and communication)
- Uncertainties
- Methods and application challenges (perhaps related to uncertainty), e.g. Long term releases, external events, etc.

Expert's opinion:

In general, Level-3 PSA is very comprehensive task with big possible variations in the scope so that the scope of analysis and character of the results should be carefully defined at the beginning of the project on the base of site specifics. The following focus areas have been mentioned:

- Assumptions and uncertainties in the Level 1 and Level 2 PSA can be hidden in the Level 3 PSA and incorrect comparisons can be made. How should we deal with uncertainties in PSA Level 1 and especially PSA Level 2 (e.g. regarding results from MAAP/Melcor)?
- Finding the appropriate scope of a PSA Level 3. How long we will look at the consequences, when do we stop? How do we ensure that it is not to extensive or too simplified?
- How should protective measures be included in PSA Level 3? (E.g. evacuation, distribution of Iodine pills, order to keep the public indoors, etc.)
- How should PSA Level 3 results be communicated? E.g. qualitative information or frequencies?
- Prove the benefits with Level 3 PSA and make it interesting enough to widen the scope of PSA to full Level 3 evaluation
- Level 3 PSA results are typical with very high level of uncertainty so that uncertainty analysis has to be performed in detail.
- External events risk became very important part of spectrum of risks derived on PSA Level 1 and 2. Since, for external events, the consequences studied and quantified in Level 3 PSA will be strongly influenced by the character of (natural) external event, it has to be addressed in the analysis, which may be difficult.
- Level 3 PSA results are rarely needed. Therefore there can be a challenge for information transfer.
- If harmonized the power plants are forced to reflect on differences in core damage definitions, analysis programs (MAAP, Melcor, etc.). This is necessary in order to perform a meaningful PSA Level 3.

- What time window should be considered in PSA Level 3: days, weeks, months, years?
- Presentation of results. It is important to focus on qualitative information rather than frequencies. It is important that SSM is involved in the communication of PSA level 3 results and PSA Level 3 methodologies should be developed together with SSM.

Opinion from insurance companies:

Respondents from insurance companies had no comment here.

5 CONCLUSIONS, RECOMMENDATIONS AND PRIORITIZATION

5.1 RISK COMPARISON AND DEVELOPMENT OF LEVEL 3 PSA

Risk comparisons for society made risks are possible to do in theory; however, this might not be possible in practice. One reason is the difficulty in finding comparable units, based on risk. If risk comparisons are to be done this must be done carefully.

There is a difference in voluntary contra involuntary risks as well as making risk comparisons at different perspectives, e.g. from an individual or society point of view.

When comparing the risk with a nuclear power plan to other types of society made risks the whole life cycle must be taken in to account (by making a life cycle analysis, LCA).

One possible comparable unit (risk metrics) for comparing the risk with a nuclear power plant to the risks from other types of energy sources is number of deaths (e.g. per produced TWh or per operating year).

Today the issue remains, whether or not comparisons of the risks with a nuclear power plant to other energy sources are needed.

5.2 NEEDS FOR LEVEL 3 PSA

The scope for Level 3 PSA and the use of results from this type of analysis needs to be established before the need for Level 3 PSA can be defined. Main expected purposes is, however, to use Level 3 PSA as an objective guidance tool for decision making, e.g. regarding costs for rebuilds and emergency preparedness work.

By performing Level 3 PSA hopefully potential risks measures can be defined to help reduce the risk potential for a radioactive release by improving the preparedness work.

In the attempt to define unacceptable effects from a nuclear accident this is looked upon differently between the two responding categories of identification (nuclear experts and insurance company’s). This indicates the needs for defining the scope for Level 3 PSA and the use of results.

Unacceptable health effects, from a nuclear expert’s point of view, could be defined from national and international safety standards, e.g. no immediate deaths caused by radiation. Possible, unacceptable, health effects in long term could be compared to other health risks, for example background radiation. There is also the possibility of defining unacceptable health effects by setting dose criteria.

An example of an unacceptable health effects, from an insurance company’s point of view, could be: all kinds of health effects that require a visit to hospital and would not exist if the accident would not have happened, the general public should not have any adverse effect from the operation of a nuclear power plant.

Environmental effects could be defined in terms of, e.g. evacuation in short terms and contamination of land areas in long term. For the effects to be unacceptable, from a nuclear expert's point of view, the effects need to be in long terms (more than a couple of months) and affect a large area (larger than the plant site and close surroundings). From an insurance company's point of view, an environmental effect can be defined as unacceptable if the land is contaminated or if there are restrictions in land use and that would not exist if the accident would not have happened.

Environmental impact often leads to economic impact when other studies have converted environmental effects to monetary value, for example business interruption in activities like fishing, tourism, food production, this is why these to impacts not always have to be separated for one another.

Defining unacceptable economic impacts is difficult to define in general. One way to define unacceptable economic impact, from a nuclear expert's point of view, could be; when the "bills" are higher than the economic preparedness. From an insurance company's point of view, however, it could be defined as costs related to third parties such as compensation to people that have to evacuate and to move from their homes. The taxpayers should not be called upon to pay for the damages. From the answers to the questionnaire it can be noted that insurance companies are interested in the economic considerations for Level 3 PSA, while the Nuclear Safety Authorities were somewhat less interested in the question of economic impact.

One way of separating economical risk could be to define the effects in terms of risks owned by plant organizations to be acceptable while effects outside the plant site are unacceptable.

5.3 ADVANTAGES OF USING LEVEL 3 PSA AND RISK COMMUNICATION

If the use of Level 3 PSA could lead to defining the risk with nuclear power off-site and expressing the risks in terms that are possible to compare, discuss and calculate (e.g. in monetary values) with other societal risks then the results would be communicable.

Making the risks communicable could help to improve the communication between the nuclear industry, authorities, insurance companies and the community.

The most important communication path consists of two parts. One consists of the communication from experts to authorities and the other one is from authorities to the community (e.g. private persons, non-governmental organizations, and media). However, the authorities (e.g. STUK and SSM) are in a double role because they are both experts and authorities. Communication by authorities is more important than communication by experts.

For the nuclear industry Level 3 PSA could help to:

- Communicate with insurance companies and the analysis could lead to better insurance possibilities
- Communicate with the society in large and thereby create higher acceptance for nuclear power in society
- Better understand societal risks of commercial nuclear power and thereby improve preparedness work
- Provide better design and siting considerations for new construction projects
- Cost benefit metric for plant retrofits
- Improve and extend earlier levels of PSA, Level 1 and 2, in creating a more holistic point of view (this is not a unified opinion).

5.4 CHALLENGES WITH LEVEL 3 PSA

There are several possible uncertainties involved in Level 3 PSA, e.g. uncertainties in the analyses, uncertainties when working with probabilities, uncertainties from ingoing parameters, difficulty to make comparisons between different reactors. The method might also be expensive and require a lot of work and there is also a risk for a large gap in time between preforming Level 3 PSA studies which leads to problems with knowledge transfer.

Aside from this there is also, as earlier discussed, difficult to communicate risks and there are different risk perceptions to take into consideration.

On the other hand, as earlier discussed, the possible advantages with Level 3 PSA are many. One of the (unique) advantages that Level 3 PSA can provide is the possibility to compare negative impacts from different technologies. There is also a possibility to see the uncertainties with Level 3 PSA to be, in fact, one of the reasons that we need the analysis method. Level 3 PSA is needed due to the uncertainties.

This different point of view is important to take into consideration when deciding whether or not to work with Level 3 PSA.

“The challenges are also the reasons for performing a Level 3 PSA”.

To be able to uniform the work with Level 3 PSA suitable risk metrics must be defined and the need for safety criteria’s and guidelines must be determined. There is also the question on how to define an unacceptable release and how the results from a Level 3 PSA study should be used.

<u>Risk metrics</u>

Suitable risk metrics can be divided based on the possible risk effects; health effects, environmental effects and economic effects, in both short- and long-term.

The complete risk metric would be economic risk metric, since it will cover all the aspect of the risks, but it is the hardest one to use. It can be too much work to get it realistic due to difficulties to determine the economic value for the consequences.

Other possible risk metrics are doses and contamination of land. It is relatively easy to calculate fatalities from these metrics.

Different risk metrics are suitable for different parts of the society depending on the target group. For an insurance provider the economic analysis would be important, and relevant economic metrics would be of interest, while for authorities some other risk metric could be of greater interest.

At this stage of the project it can be difficult to decide which risk metrics is the most suitable; the scope of Level 3 PSA needs to be defined. A separate task within this project, task 1, is focused on finding the appropriate risk metrics.

<u>Safety criteria</u>

Whether or not safety criteria are required for Level PSA have been debated during the work with task 0.

Some of the respondents felt that Level 3 PSA has not been performed or applied enough to define such criteria, or even to see the needs for such criteria. Those that felt criteria should exist were interested in using them as a means of defining the scope of the analysis.

The outcome from the discussions during the workshop was that safety criteria must be defined in order to understand the results of a PSA. We need such criteria to understand if the results are good/bad or acceptable or unacceptable, this provides focus to an analysis.

Possible safety criteria's should be the same for old and new plants.

The need for and definition of Level 3 PSA safety criteria's need to be further studied.

<u>Unacceptable release</u>

Defining an unacceptable release for Level 3 PSA needs to be based on acceptance criteria's. Examples for this today are the ALARA principle and dose criteria's, determined from regulations by the authorities. There is also a need for defining reference values

From a nuclear experts point of view a definition of an unacceptable release for Level 3 PSA should be related to how we define unacceptable release in Level 2 PSA. Based on reference values, e.g. to define a limitation for the background radiation from normal operation of a NPP to be 1/10 of the "natural" background radiation, and them limiting all accidents to 1/10 for the background radiation from normal operation (1/100 of "natural" background radiation).

The responding insurance company's definition of an unacceptable release is; any release which will have an adverse effect should be deemed unacceptable.

Note that there is a risk that the work with minimizing the risks could be held up if we define what risks that are acceptable.

Use of results

Use of results from a Level 3 PSA study has been discussed in several different contexts related to the intended use.

The discussions have been regarding the use for communication to the public (if this is done carefully), the use in planning (e.g. emergency planning, accident management) and the use by increase the knowledge for possible effects from a nuclear accident and thereby to face the risks that we are exposed to and make use of this for new built (e.g. site location) or rebuilt.

The starting point for developing Level 3 PSA should be the intended use of the results.

Guidelines

If we are to use Level 3 PSA we are going to need guidelines.

Since there are so many ways to perform the analyses and evaluate the results guidelines are needed to ensure that scenarios from one plant can be compared with scenarios from another plant.

The guidelines could, though, be written as suggestions rather than a strict guideline. The guideline should give some input on different ways of performing Level 3 PSA depending on the objectives. Use of international guideline and specify the order of detail used in Nordic countries.

Overall challenges

The challenges with the further work of Level 3 PSA are defining the scope for the analysis method. Finding the appropriate risk metrics and comparable units is another challenge.

When preforming Level 3 PSA the challenges are related to necessary assumptions and uncertainties. The analysis method might also be expensive and complex to perform. There are also difficulties to make right comparisons between different reactors.

Communication of the results from a Level 3 PSA study is a challenge in itself. However, the challenges of Level 3 PSA might also be the reasons for preforming Level 3 PSA.

APPENDIX 1 RESPONDING ORGANIZATIONS

Name of organization	Category of identification
FKA	Experts
RAB	Experts
ES-konsult	Experts
SSM	Experts
RiskPilot	Experts
STUK	Experts
UJV Rez	Experts
VUJE	Experts
Fortum	Experts
OKG	Experts
AON	Insurance company
Elini	Insurance company

APPENDIX 2 LITERATURE STUDY

The literature study is presented below consisting of a short introducing text (summary) for each report/study.

Probabilistic Safety Goals for Nuclear Power Plants

The outcome of a probabilistic safety assessment (PSA) for a nuclear power plant is a combination of qualitative and quantitative results. Quantitative results are typically presented as the Core Damage Frequency (CDF) and as the frequency of an unacceptable radioactive release. In order to judge the acceptability of PSA results, criteria for the interpretation of results and the assessment of their acceptability need to be defined.

Safety goals are defined in different ways in different countries and also used differently. Many countries are presently developing them in connection to the transfer to risk-informed regulation of both operating nuclear power plants (NPP) and new designs. However, it is far from self-evident how probabilistic safety criteria should be defined and used. On one hand, experience indicates that safety goals are valuable tools for the interpretation of results from a probabilistic safety assessment (PSA), and they tend to enhance the realism of a risk assessment. On the other hand, strict use of probabilistic criteria is usually avoided. A major problem is the large number of different uncertainties in a PSA model, which makes it difficult to demonstrate the compliance with a probabilistic criterion. Further, it has been seen that PSA results can change a lot over time due to scope extensions, revised operating experience data, method development, changes in system requirements, or increases of level of detail, mostly leading to an increase of the frequency of the calculated risk. This can cause a problem of consistency in the judgments.

The first phase of the project (2006) provided a general description of the issue of probabilistic safety goals for nuclear power plants, of important concepts related to the definition and application of safety goals, and of experiences in Finland and Sweden. The second, third and fourth phases (2007–2009) have been concerned with providing guidance related to the resolution of some of the problems identified, such as the problem of consistency in judgment, comparability of safety goals used in different industries, the relationship between criteria on different levels, and relations between criteria for level 2 and 3 PSA. In parallel, additional context information has been provided. This was achieved by extending the international overview by contributing to and benefiting from a survey on PSA safety criteria which was initiated in 2006 within the OECD/NEA Working Group Risk. Finally, a separate report has been issued providing general guidance concerning the formulation, application and interpretation of probabilistic criteria.

The results from the project can be used as a platform for discussions at the utilities on how to define and use quantitative safety goals. The results can also be used by safety authorities as a reference for risk-informed regulation.

The outcome can have an impact on the requirements on PSA, e.g., regarding quality, scope, level of detail, and documentation. Finally, the results can be expected to support on-going activities concerning risk-informed applications.

Bengtsson, L., Holmberg, J.-E., Rossi, J., & Knochenhauer, M. (2010). Research 2010:35 Probabilistic Saftey Goals for Nuclear Power Plants. Swedish Radiation Saftey Authority.

Analysis of the impact on society by radioactive emissions in Japan in 2011

Only two major releases of radioactive substances from nuclear accidents that have occurred over the world. One of them are the nuclear accident that occurred in Japan in 2011 and it is therefore of interest to study the social impacts from this large accident.

The analysis, made by MSB, shows that the largest and most serious consequences from the accident I Japan are:

• The concern over the future at an individual level, about the health risks of ionizing radiation, residents in the long term and questions about economic benefits.

• Decontaminate from a social organizational perspective. It is expensive, requires collaboration and takes time to resolve. No reconstruction can begin in contaminated areas until it is resolved.

• Analyses for possibilities to replace nuclear energy from a technology and resource perspective. Sampling of food and control of radiation doses in humans in the affected area is extensive.

• Management of costs from the economic perspective. Expenses are expected to be very large and the Government of Japan has begun to make changes in the state budget for managing and allocating the costs of the community.

The analysis preformed in the report is also meant to be used as a basis for the further development of the Civil Contingencies with respect to large radioactive release.

MSB. (2012). Analys av samhällskonsekvenser efter radioaktiva utsläpp i Japan 2011.

RAMA II, RAMA III

RAMA II and RAMA III was both included in the Swedish program for consequence mitigation measures for severe reactor accidents, along with the projects FILTRA and RAMA. The program ended in 1988.

The aim for the program was to:

- Build a knowledge base for understanding of the important processes during a severe reactor accident

- Further develop and validate a tool for calculating failure analysis with site-specific adaptation
- Document the knowledge that formed the basis for the development and implementation of the mitigating measures at the Swedish NPPs

RAMA were to act as a complement to the utilities plant specific analyses and find appropriate means for protecting the environment in case of severe reactor accident.

The purpose of the project RAMA II was to develop the analytical tool for the analysis of severe accidents, to be employed by the utilities in their plant specific studies and to validate the analytical tool and to consolidate the scientific basis for the conclusions of the RAMA project.

RAMA II. (1987). Final report. Nyköping: Studsvik Library.

RAMA III. (1989). Handbok över haveriförlopp i svenska reaktorer. Nyköping: Studsvik Library.

Air quality guidelines

Health effects from particles were discussed during the workshop. One good overview is the air quality guidelines published by the World Health Organization. The report is large, but the summary table on the best risk estimates for PM exposure is found on page 275 (table 5).

World Health Organization (2005). Air quality guidelines. Global update 2005. Particulate matter, ozone, nitrogen dioxide and sulfur dioxide. (http://www.euro.who.int/en/health-topics/environment-and-health/air-quality/publications/pre2009/air-quality-guidelines.-global-update-2005.-particulate-matter,-ozone,-nitrogen-dioxide-and-sulfur-dioxide)

WENRA-documents

One of the objectives for WENRA (Western European Nuclear Regulator´s Association) is to develop a harmonized approach to nuclear safety and radiation protection issues and their regulation.

A significant contribution to this objective was the publication, in 2006, of a report on harmonization of reactor safety in WENRA countries. This report addresses the nuclear power plants that were in operation at that time in those countries.

Since then, the construction of new nuclear power plants has begun or is being envisaged in the short term in several European countries. Furthermore, some plants whose construction had been halted several years ago are now under completion. Despite all these plants were not addressed in the study published in 2006, it is expected that, as a minimum, they should meet the corresponding “Safety Reference Levels”.

These “Safety Reference Levels” were designed to be demanding for existing reactors. However, in line with the continuous improvement of nuclear safety that WENRA members aim for, new reactors are expected to achieve higher levels of safety than existing ones, meaning that in some safety areas, fulfillment of the “Safety Reference Levels” defined for existing reactors may not be sufficient.

Hence, it has been considered timely for WENRA to define and express a common view on the safety of new reactors, so that:

- new reactors to be licensed across Europe in the next years offer improved levels of protection compared to existing ones;
- regulators press for safety improvements in the same direction and ensure that these new reactors will have high and comparable levels of safety;
- applicants take into account this common view when formulating their regulatory submissions.

One example on proposed safety objectives for new builts from the report *Safety Objectives for new Power Reactors:*

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

WENRA. (2007). Reactor Safety Reference Levels

WENRA. (2009). Safety Objectives for new Power Reactors. Reactor Harmonisation Working Group, WENRA

www.wenra.org

OECD/NEA-documents

The main objective of the Working Group on Risk Assessment (WGRISK) of the Nuclear Energy Agency (NEA)/Committee on the Safety of Nuclear Installations (CSNI) is to advance the Probabilistic Safety Assessment (PSA) understanding and to enhance its utilization for improving the safety of nuclear installations. Due to its disciplined, integrated and systematic approach, PSA is now considered as a necessary complement to traditional deterministic safety analysis.

To accomplish this mission, WGRISK performs a number of activities to exchange PSA-related information among member countries.

The results of exchanges have been compiled in the report “Use and Development of Probabilistic Safety Assessment”. The report provides a description of the PSA activities in the member countries at the time of the report writing. Since there have been significant new developments in PSA since the last version, and considering the interest and usefulness of the previous versions, WGRISK initiated the development of an updated version of the report in early 2011.

NRC response on questionnaire: An extract from the OECD-NEA report; “Use and development of PSA in Member and non-member countries” that concerns PSA Level 3: page 405

Although Level 3 PSAs are required to directly estimate the risk to the public from nuclear power plant accidents, the NRC does not routinely use them in risk-informed regulation. In fact, NRC-sponsored Level 3 PSAs have not been conducted since the late 1980s. These Level 3 PSAs were documented in a collection of NUREG/CR reports and a single corresponding summary document, NUREG-1150. The NUREG-1150 study provides a set of PSA models and a snapshot-in-time (circa 1988) assessment of the severe accident risks associated with five commercial nuclear power plants of different reactor and containment designs. The NRC has used the landmark NUREG-1150 results and perspectives in a variety of regulatory applications, including development of PSA policy statements, support of risk-informed rulemaking, prioritization of generic issues and research, and establishment of numerical risk acceptance guidelines for the use of CDF and large early-release frequency (LERF) as surrogate risk metrics for early and latent cancer fatality risks.

Since then, the NRC has ensured safety primarily by using results obtained from Level 1 and limited Level 2 PSAs—both less expensive than Level 3 PSAs—and how they relate to lower level subsidiary safety goals based on CDF and LERF to risk-inform regulatory decision making.

There are several compelling reasons for conducting a new comprehensive site Level 3 PSA. First, in the two decades since the publication of NUREG-1150, there have been substantial developments that may affect the results and risk perspectives that have influenced many regulatory applications. In addition to risk-informed regulations implemented to improve safety (e.g., the Station Blackout and Maintenance Rules), there have been plant modifications that may affect risk (e.g., the addition or improvement of plant safety systems, changes to technical specifications, power uprates, and the development of improved accident management strategies). Along with NRC and industry acquisition of over 20 years of operating experience, there have also been significant advances in PSA methods, models, tools, and data— collectively referred to as “PSA technology”—and in information technology. Finally, the NRC is conducting a State-of-the-Art Reactor Consequence Analysis (SOARCA) study, which leverages many of the same safety improvements and technological advances, integrates and analysis two of the essential technical elements of a Level 3 PSA for some of the more likely reactor accident sequences–the severe accident progression and offsite consequence analyses. A new level 3 PSA could therefore seek to leverage the methods, models, and tools used in the SOARCA analysis and capitalize on the insights gained from the application of state-of-the-art practices.

In addition to these developments, the Level 3 PSAs documented in NUREG-1150 are incomplete in scope. Figure 9.US-2 illustrates the scope of a complete site accident risk analysis, with the approximate scope of the NUREG-1150 PSAs shown by the gray-shaded region. These PSAs were limited to the assessment of single-unit reactor accidents initiated primarily by internal events occurring during full power operations. The partial coverage of external events indicates that a limited set of external events (fires and earthquakes) were considered for only two of the five analyzed nuclear power plants.

To update and improve its understanding of reactor accident risks, the NRC is considering evaluating accidents that might occur during any plant operating state, that are initiated by all possible internal events and external events, and that may simultaneously affect multiple units per site. Moreover, for a comprehensive site accident risk analysis, the NRC is also considering analyzing the risk from other site radiological hazards, such as spent fuel and radioactive waste streams. Because corresponding surrogate risk metrics that can be meaningfully integrated with and compared to CDF and LERF do not exist for these other radiological hazards, this analysis can only be accomplished in Level 3 space.

For these reasons, the NRC staff has identified three specific objectives for a potential new comprehensive site Level 3 PSA project. The first objective is to update and improve staff understanding of site accident risk by (1) incorporating plant safety improvements, insights from SOARCA, and advances in PSA technology that have occurred in the two decades since NUREG-1150, and (2) integrating the risk from additional radiological hazards using consistent assumptions, methods, and tools to enable a meaningful comparison and ranking of risk contributors to focus the NRC's safety mission. Second is to upgrade and disseminate information about the NRC's PSA technology, using 21st-century information technology in a comprehensive risk analysis toolbox that will enhance the NRC's ability to risk-inform current and future regulatory decision making. Third is to develop PSA expertise by training a new generation of risk analysts who will gain state-of-the-art knowledge and experience.

RES has initiated a scoping study to identify various options for the following elements of a pilot study: (1) site selection, (2) project scope, (3) PRA technology to be used, (4) new research needed to accomplish the project's objectives, and (5) resource estimates and information needs to better understand and address potential challenges. The results of this scoping study, along with a specific recommendation for a Level 3 PSA pilot project, to the Commission for consideration in 2011.

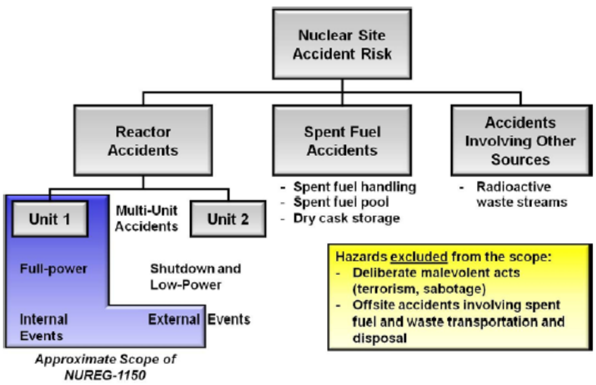

Figure 9.US-2. Site Accident Risk and Approximate Scope of NUREG-1150 (source: NUREG-1925, Rev. 1, Figure 5.3)

NEA/CSNI/R(2007)12. Use and development of PSA in Member and non-member countries, OECD/NEA WGRISK, Paris, Update 2011, limited distribution

NEA/CSNI/R(2012)11.Use and Development of Probabilistic Safety Assessment, An Overview of the Situation at the end of 2010Nuclear Safety

www.oecd-nea.org

WASH-1400

Comparisons of different societal risk were made in WASH-1400, the final report of the Reactor Safety Study "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants”.

WASH-1400. NUREG. (1975). Reactor Safety Study, An assessment of Accident risks in U.S. commercial nuclear power plants.

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Addressing Off-site Consequence Criteria Using Level 3 PSA **Task 1 – Appropriate Risk Metrics** **Final report**	

Summary

This report is the final report of *Task 1 – Appropriate Risk Metrics* within the NKS/NPSAG-project *Addressing Off-site Consequence Criteria Using Level 3 PSA – Enhanced Scoping Study*. Different risk metrics for Level 3 PSA is discussed in the report. A risk metric has two components: 1) probability metric and 2) consequence (or impact) metric. The main probability unit used in the nuclear regulatory decision making is "probability per year per reactor". Probability units "per lifetime" and "per produced energy over the complete fuel life cycle" can be considered in risk comparisons. Three categories of consequence metrics are defined and discussed in the report: 1) health effects, 2) environmental impact and 3) economic impact.

Health effects and environmental impact are rather similar metrics from the estimation and purpose point of view. The assessment of these metrics should be of interest to all stakeholders. There are still a number of open issues to be further explored, e.g., how far in time and place the estimations need to be done, i.e., what is the time frame for the risk metrics and how far away from the plant should the impact be accounted for? The pilot study should elaborate more on these risk metrics when the scope of the study is determined. The pilot study should also elaborate how level 2 release category related risk metrics could be used as surrogates for level 3 criteria.

Economic impact is an ideal metric from decision making point of view and it would allow cost-benefit studies. In practice, it can be difficult to agree on what should be included in the quantification of economic impact and how to convert different impacts in a monetary scale. Despite the difficulties of evaluating economic impact, one possibility could be to apply some simplified categorisation of economic impacts in terms of orders of magnitude. It is suggested that the pilot study should include at least a discussion of economic impacts within the framework of a licensee's risk analysis. This discussion should also explore ways of assigning a monetary value on non-monetary impacts, e.g., doses and environmental impacts to euros. Main use of economic impact may be in cost-benefit assessments instead of being compared to absolute numerical risk criteria.

Table of Contents

1 Introduction

The project *Addressing Off-site Consequence Criteria Using Level 3 PSA – Enhanced Scoping Study* is defined in the project plan [1]. The project is performed jointly by Lloyd's Register Consulting, Risk Pilot, ES-Konsult and VTT. This report is the Final report of *Task 1 - Appropriate Risk Metrics*.

1.1 Background

Level 3 PSA provides an assessment of off-site consequences from a radioactive release. Results from the identification and assessment of accident sequences leading to core damages are assessed in the Level 1 PSA, the severe accident and radioactive source term analysis are assessed in Level 2 PSA, whereas Level 3 PSA uses meteorological data, radionuclide release data, population and agricultural data to estimate the risks to the public.

Even though Level 3 PSA is required only in a few countries, the interest is broader. The increased interest and activities regarding Level 3 PSA is due to the interest in better understanding and characterizing the off-site consequences from the Fukushima accidents, the obligations utilities have from insurance companies and shareholders, and the obligations regulators have to the public's health and safety.

The purpose of this report is to discuss which kind of risk metrics is suitable to use for Level 3 PSA. The results from the report will contribute to the ultimate objective and outcome of the project as a whole, and serve as a guidance document providing clear and applied guidance towards regulators, utilities and Level 3 PSA practitioners.

In the previous performed work in the NKS/NPSAG Safety Goals project [3], information can be found on what safety goals are being used in different countries and industries, together with arguments and historical background on why different criteria are being used in these countries. Some of the safety goals are related to Level 3 PSA.

During the past two years a Masters project investigating Level 3 PSA has been performed, [4]. The Masters project included participation in the ANS/ASME Level 3 PSA standard writing committee and in the IAEA Level 3 PSA technical meeting, which will produce guidance for the IAEA's future actions in the field of Level 3 PSA.

1.2 Scope of work and limitations

The main goal of this report is to present appropriate risk metrics for Level 3 PSA. No safety goals, i.e., no numerical criteria, related to the risk metrics are presented. However, safety goals will be touched upon as a reference to which risk metrics that could be used.

2 Off-site consequence criterion – Safety Goal project

There are a number of countries worldwide which have more or less clear safety goals or off-site consequence criterion connected to Level 3 PSA or risks with hazardous industries. Examples can be found in [3], [4], [14] and [19] and are presented in Table 1 without the numerical criterion itself presented.

Table 1. Definition of different off-site consequence criteria (safety goals) used in different countries.

Country	Individual risk	Societal Risk	Other
UK	The individual risk of death to a person off-site, from on-site accidents that result in exposure to ionising radiation.	The total risk of 100 or more fatalities, either immediate or eventual, from on-site accidents that result in exposure to ionising radiation	Frequency dose The total predicted frequencies of accidents on an individual facility, which could give doses to a person off the site.
The Netherlands	The individual risk of death as a consequence of the operation of a certain installation. The individual risk shall be calculated for one-year-old children, since this is, in general, the most vulnerable group of the population.	The risk of 10 or more casualties, which are directly attributable to the accident. F/N-curve	
US	Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.	The risk to society from generating electricity using nuclear power is compared with that from generating electricity by other techniques. It is also compared with other societal risks (sum of cancer fatalities from other sources)	
Sweden*	There shall be no short-term fatalities in acute radiation syndrome (sickness)	Long-term ground contamination of large areas shall be avoided.	
Japan	Average risk of acute fatality for individuals in the vicinity if the site boundary. Average risk of latent fatality for individuals living within a certain distance from the facility.		
Canada	Average risk of latent effects (per site)		

* In Sweden, no level 3 PSA is required. These safety goals are related to acceptance criteria for the mitigating systems for a severe accident [7]

Most of the off-site consequence criterions used in different countries is related to health effects both to individuals and to the society at large. For numerical criteria, see e.g. [19].

3 Input from Task 0 questionnaire

A questionnaire has been sent out to different stakeholder as one part of this project *"Addressing Off-site Consequence Criteria Using PSA Level 3 – Enhanced Scoping Study"*. The following stakeholders answered the questionnaire:

- Nuclear power plant and plant owners (RAB, FKA, OKG, Fortum)
- Authorities (SSM, STUK)
- Nuclear related companies (Lloyd's Register Consulting, Risk Pilot, ES-konsult, VUJE, UJV Rez)
- Insurance related companies (AON, Elini)

The stakeholders in the first three groups are seen as experts and last group is obvious seen as representatives from the insurance companies.

Many of the questions in the questionnaire are more or less related to risk metrics. However, there was one specific question related to risk metric that was phrased as [8]:

> *"4.4.2 Risk metric*
>
> *Appropriate risk metrics is one of the main questions when developing Level 3 PSA. This project will have the possibility to contribute to the development of Level 3 PSA and highlight important aspects like: What are the proper risk metrics? How should we look at risks and how should they be graded?*
>
> ***What kind of risk metric is suitable in a Level 3 PSA?***
>
> *Examples for discussion:*
>
> - *To present risk assessment results in terms of short term effects (radiation levels, dose levels)*
> - *To present risk assessment results in terms of long term effects (health effects, social effects)*
> - *To present risk assessment results in terms of economic impact (for example ground contamination, cost of evacuation, cost of lost production)*
> - *To present risk assessment results in terms of environmental impact*
> - *Other"*

The following is a compilation of the answers:

> ***<u>"Expert's opinion:</u>***
>
> *In general:*
>
> *Suitable risk metrics can be divided in health effects (in both short- and long-term), environmental and economic effects. Examples for each type of effects can be found below. To present risk assessment results in terms of:*
>
> - *short term effects (radiation levels, dose levels)*
> - *long term effects (health effects, social effects)*
> - *economic impact (for example ground contamination, cost of evacuation, cost of lost production)*
> - *environmental impact*
>
> *Risk metrics regarding environmental effects can be presented in terms of land area considered to be lost for a long time. Presenting risk assessment in long term effects (and not just short term effects) would include cumulative effects as well. Health effects (in terms of number of deaths/cancers) are easier to determine than*

environmental and economic. The effects in terms of environmental and economic will also have a higher level of uncertainty.

Specific:

Different risk metrics is suitable for different parts of the society depending on the target group. For example, health effects in terms of frequency of deaths or number of cancer can be suitable if the target group is the public but the economic effects on the other hand is of greater interest when measuring the risk for a power plant organization.

A possible risk metric is to present the (health-) effects in terms of frequency per produced TWh. This makes the risk metrics possible to compare to other types of energy sources.

When considering health effects (short- and long term) the reference values used in safety standard should be used. For short term effects this means no immediate deaths caused by radiation and for long term health effects SSM has developed limits for non-acceptable land contamination and radiation doses. Dose criteria's are easier to relate to than occurrence of diseases.

Opinion from insurance companies:

A level 3 PSA should both give an indication of the effective dose and dose rates people will be exposed to after a release. Health effects from high doses are well known but this is much unsure for low doses. Financial losses, decontamination costs etc. should be estimated as accurate as possible in order to have a (rough) estimation of the total loss amount caused by a nuclear accident.

Suitable risk metrics are listed in bullets above."

In connection with the questionnaire a workshop among the Swedish and Finish expert stakeholders was held. The conclusion from the workshop was as follows:

"The complete risk metric is the economic risk metric but it is the hardest one to use. Too much work to get it realistic.

Dose and contamination of land should be used. It is relatively easy to calculate fatalities from these metrics.

At this stage of the project it can be difficult to decide which risk metrics is the most suitable. All risk metrics have to be discussed regarding pros and cons, limitations, uncertainties and purpose. During the pilot study more can be elaborated in connection to risk metrics."

The general conclusion from the questionnaire and a succeeding workshop in December 2013 is that multiple risk metrics should be used and that different risk metrics is to be used when presenting the Level 3 PSA results for different kinds of stakeholders. However, it is not possible based on today's knowledge to decide on which risk metric to use or the priorities amongst the different risk metrics. Rather a comparison between the different risk metrics should be done.

4 Risk metrics for Level 3 PSA

4.1 Introduction

Risk metrics at Level 3 PSA have two components: 1) probability metric and 2) consequence (or impact) metric. Regarding the probability metric, it is matter of choosing the normalization unit for risk comparison purposes. These are discussed in Section 4.2. The consequence metric is associated with the impacts which are quantified in the consequence assessment part of a Level 3 PSA.

Based on the questionnaire [8] and the project workshop connected to the questionnaire, Sections 4.3 to 4.5 discuss the different consequence metrics that can be used for Level 3 PSA. The discussion includes definitions of different consequence metrics and advantages/disadvantages in using them. The following main groups of consequence metrics are identified:

- Health effects
- Environmental impact (strongly related to the possible health effect due to contamination)
- Economic impact (includes other consequence metrics).

4.2 Probability units

The results of a PSA, at any level (1, 2 and 3), are typically presented as probabilities of the unwanted events (core damage, large release, offsite impact) *per year*, and, hence, it can be interpreted as a frequency. The interpretation of the probability per year is that it represents the average risk for a certain nuclear plant that has been analysed by PSA methods and, if it is a full-scope PSA, the numbers should have been integrated over different plant operating states taking into account the fraction of operating time spent in these different operating states. "Probability per year" is the unit which is used in the regulatory framework and it is almost always associated with a single reactor, since operating licenses are reactor specific. However, in some countries a "probability per year per site" is used (see [14]).

In living PSA applications, other probability units may be applied, like probability per an event, probability per a specific time period or probability per expected (remaining) lifetime. The probability per expected lifetime should be relevant from the investment decision making point of view.

From the risk comparison point of view, probabilities could also normalized by the produced amount of energy, e.g., per TWh (or TWh_e). An example comparing the full fuel life cycle risks of different energy options can be found in [9].

Since "probability per year per reactor" is the probability unit applied in the regulatory context, the probability metric is mainly considered in this report. Probability units "per lifetime" and "per produced energy over the complete fuel life cycle" can be considered for risk comparison purposes.

4.3 Health effects

In the following section risk metrics related to health effects are discussed. Health effects are mostly considered as the radiological impact on the population.

4.3.1 Input

INES

The International Nuclear and Radiological Event Scale (INES) is used for communicating to the public the safety significance of events associated with sources of radiation [10]. The scale was developed by international experts convened by the IAEA. Events are classified on a scale of seven levels: Levels 4–7 are termed "accidents" and Levels 1–3 "incidents".

The rating of events is based on both qualitative (e.g. barriers broken in defence-in-depth) and quantitative criteria (e.g. dose estimation). The dose criteria given in INES are listed in Table 2. Release criteria are given for INES-classes 4–7 which involve radiological releases. Doses to individuals are defined for INES-classes 1–4. It should be noted that these are not the only criteria to be used in the classification of events and that in many cases conversion factors need to be used to find the equivalent class, see [10] for guidance. For instance, a multiplication factor 40 should be used for Cs-137 release to obtain the radiological equivalence to I-131 release.

In [11], a safety goal framework is proposed in the framework of INES. A probabilistic scale associated with source terms (noble gas, iodine, and caesium) are defined. Importantly, the safety goals of this approach are deployed to an individual plant and require site-specific assessments.

Table 2. Dose criteria related INES-classes. For technical details see [10].

INES-class		Equivalent I-131 release	Doses to individuals
7	Major accident	More than several tens of thousands of terabecquerels	
6	Serious accident	The order of thousands to tens of thousands of terabecquerels	
5	Accident with off-site risks	The order of hundreds to thousands of terabecquerels	
4	Accident mainly in installation	The order of tens to hundreds of terabecquerels	(1)The occurrence of a lethal deterministic effect; or (2) The likely occurrence of a lethal deterministic effect as a result of whole body exposure, leading to an absorbed dose of the order of a few Gy.
3	Serious incident		(1) The occurrence or likely occurrence of a non-lethal deterministic effect; or (2) Exposure leading to an effective dose greater than ten times the statutory annual whole body dose limit for workers.
2	Incident		(1) Exposure of a member of the public leading to an effective dose in excess of 10 mSv; or (2) Exposure of a worker in excess of statutory annual dose limit
1	Anomaly		(1) Exposure of a member of the public in excess of statutory annual dose limits; or (2) Exposure of a worker in excess of dose constraints

ASAMPSA2

The ASAMPSA2 (volume 2) states that in an extended Level 2 PSA one can use the off-site dose calculated using simplified deterministic methods as risk metrics [12]. It is mentioned that for the French 900 MWe NPP ISRN uses the total effective dose equivalent, integrated over a 15 day period to a one year old child 2 km from the damaged plant as risk metric.

Realistic radiological consequences in Swedish NPPs

At the Swedish NPPs a project related to evaluating realistic radiological consequences have been performed during 2010–11. The project calculated realistic radiological consequences for all anticipated operational transients and design basis accidents events [6]. Two different dose metrics related to dose were used in the project.

- Effective dose (sum of effective dose from external radiation from radionuclides in the air, internal radiation during 50 years from inhaled radionuclides and external radiation over 30 days from radionuclides on the ground).
- Equivalent dose to the thyroid of a one-year old child due to inhaled radioactive iodine.

The values for the dose metrics were calculated for different distances off-site from the plant.

WENRA

The Reactor Harmonization Working Group (RHWG) of Western European Nuclear Regulator's Association (WENRA) has released a report on safety of new NPP designs [5]. WENRA has issued safety objectives for new reactors including objective for accidents with core melt. The following criteria are stated:

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

To meet the criteria Level 3 PSA can be used as one tool to show that an accident is practically eliminated. In connection with the second criteria some consequence metrics are mentioned, i.e., dose and ground contamination.

Safety goal project

In section 2 different safety goals are presented for some countries worldwide. Some of them are related to health effects and are mostly related to individual or collective dose or fatalities [14].

4.3.2 Identified metrics

Based on the above references a metric connected to health effects and dose is relevant. Both individual dose and collective dose are of interest for both short-term and long-term effects. From the individual short-term and collective long-term dose both prompt fatalities and cancer fatalities can be calculated, se section 4.3.3.

The following metrics related to health effects are identified:

- Collective dose/individual dose (short- and long-term) [mSv]
- Prompt fatalities (short term)
- Cancer fatalities (long term).

4.3.3 Fatal dose level

The connection between dose and fatalities are described below.

Prompt fatal dose level

In order to estimate the prompt fatalities from dose exposure one needs to define the dose level at which acute radiation syndrome occurs or where the risk for it increases (deterministic effects). The Swedish industry has set 1 Sv as the short-term dose limit for acute radiation sickness causing death to occur. This is in line with the threshold value given in a basic radiation physics textbook [17] and in education material from KSU [18]. The risk of death is about 50 % (LD_{50} , median Lethal Dose) if a short-term whole-body dose of approximately 4 Sv is received and 100 % (LD_{100}) if a short-term whole-body dose of approximately 6 Sv is received, and if no treatment is given. In order for acute radiation syndrome to occur, the dose rate has to be in the order of Sv/min.

In Figure 1 the relationship between risk of death and received whole body dose exposure is shown. Using the information above it is relatively easy to connect the individual dose to prompt fatalities from acute radiation sickness. It should be noted that the threshold value for foetus is much lower, approximately 100 mSv (0.1 Gy as stated in [17]).

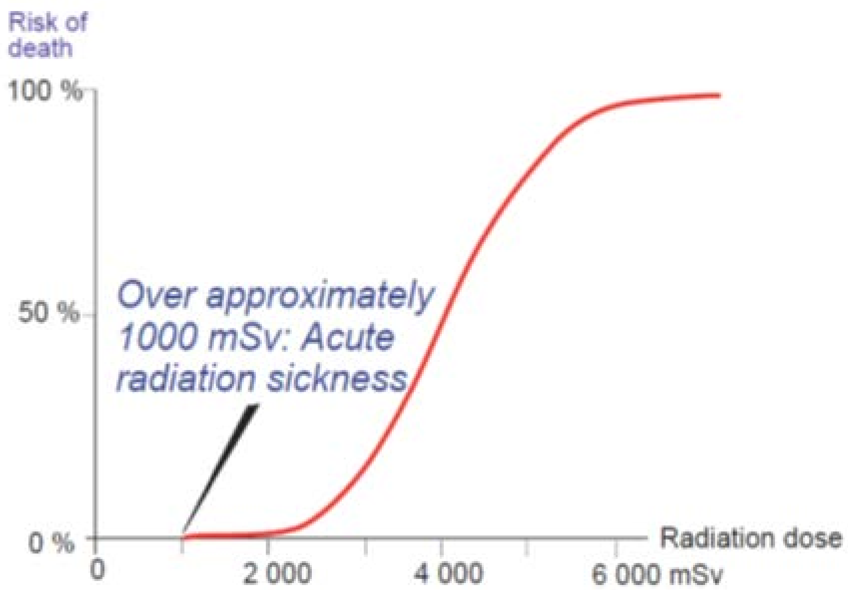

Figure 1. Risk of death in acute radiation sickness due to different radiation dose exposure.

Long-term fatal (cancer) dose level

In order to estimate the long-term fatalities from dose exposure one needs to define the dose level at which the risk for cancer increases (stochastic effects). In ICPR103, [16], and also in [17] it is stated that the risk for cancer increases with 5 % per Sv in long-term for low exposure (up to 200 mSv) and 10 % per Sv for high exposure (from 200 mSv). This can be related to the collective dose:

# of death caused by cancer = collective dose (manSv) x 0.05 (1/Sv), for individual exposure ≤ 200 mSv.

or

# of death caused by cancer = collective dose (manSv) x 0.10 (1/Sv) for individual exposure > 200 mSv).

Hence, the total risk for death by cancer due to radiation exposure is independent of the individual dose exposure and only connected to the collective dose exposure. As an example a collective dose of 20 manSv results in one death due to cancer irrespectively if it is 20 000 people receiving 1 mSv or 200 people receiving 100 mSv as long as the maximum individual dose is less than 200 mSv.

4.3.4 Advantages, disadvantages and uncertainties

The advantage with the dose related risk metric is that it is rather straight forward to calculate from the release of radioactive material following a nuclear accident. The dose metric can also be connected to fatalities both in short and long term. It should also be easy to define consequence criterion to the dose metric. Both the individual and societal consequence can be estimated using dose metric (or fatality metric). The dose metric can also be used to improve plant design and emergency preparedness.

The disadvantage with the dose related risk metric is that it does not cover the complete consequences of a nuclear accident. The impact to the biosphere is not captured with the dose related risk metric, e.g. contamination/restrictions (evacuation) on land and sea, impact on wildlife is not covered by the dose related risk metric.

The uncertainties connected to dose and fatalities are the general uncertainties with respect to dispersion calculations (which also affect all other consequence metrics). Once the release and dispersion of radioactive material is calculated it is rather straight forward to calculate the dose exposure both on an individual and collective level if population densities are available. From the dose exposure it is easy to estimate fatalities. There are, however, uncertainties related to the validity of the linear, no threshold hypothesis used in the proposed way of calculating cancer deaths.

4.3.5 F/N-curve

An F/N-curve can be used to present the risk metric related to fatalities using a cumulative distribution function. Normally N is the number of fatalities and F is the frequency (probability per year) for N or more fatalities to occur. By using this risk metric one can compare the risk from a nuclear power plant with the risk from other hazardous industries. Consequences can be also scaled to per produced TWh_e if, for instance, a comparison of different energy sources is needed.

The F/N-curve (Figure 2) can also be used to express the dose risk metric by using collective dose or dose interval as N instead of fatalities.

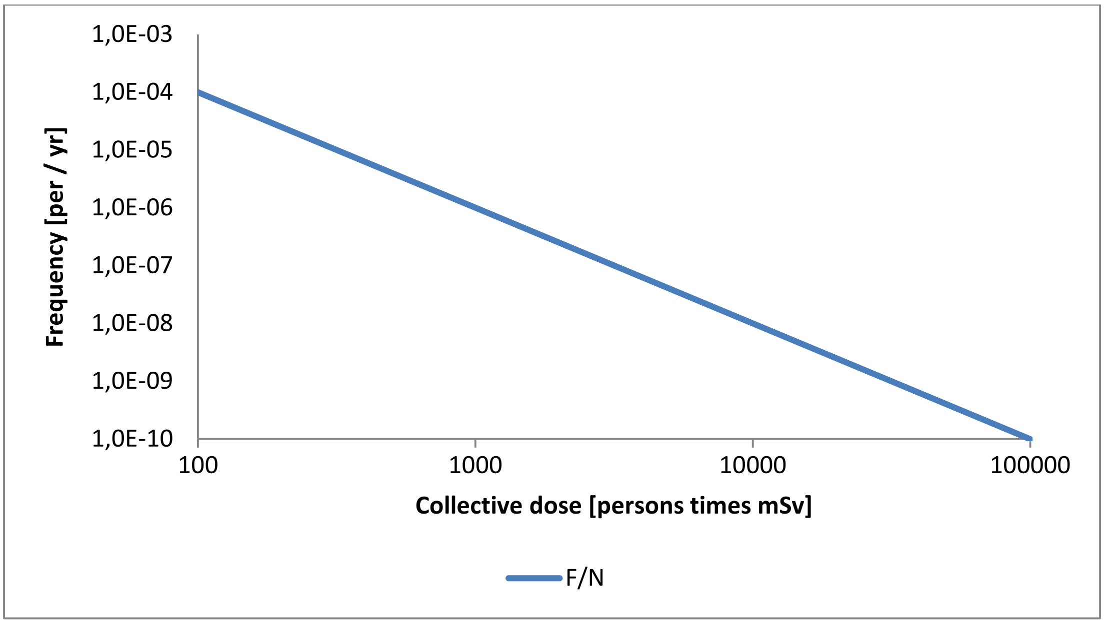

Figure 2. Example of an F/N-curve.

4.4 Enviromental impact

In the following section consequence metrics related to environmental impact are discussed.

4.4.1 Input

Realistic radiological consequences in Swedish NPPs

As mentioned above a project related to realistic radiological consequences has been performed during 2010–11 at the Swedish NPPs. The project calculated realistic radiological consequences for all anticipated operational transients and design basis accidents events [6]. The following metric was used in the project:

- Ground contamination level due to Cs-134 and Cs-137 [Bq]

This metric is connected to the requirement established by the Swedish government for severe accidents. This is judged to be fulfilled if the radioactive release after a severe accident is limited to below 0,1 % of the inventory of the caesium isotopes Cs-134 and Cs-137 in a core of 1800 MWth [7].

Fatal contamination level

Prompt fatalities can be related to the contamination of land. According to SKI/SSI [7] no short-term fatalities due to acute radiation syndrome occurs if the radioactive release after a severe accident is limited to 1 % of the inventory of a core of 1800 MWth. Hence, the contamination metric can be related to the dose metric of prompt fatalities.

Safety goal project

In section 2 different safety goals are presented for some countries worldwide. WENRA has set a qualitative safety goal that design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures. In Sweden, long-term significant ground contamination of large areas shall be avoided. Guidance suggests that up to 0.1% of core Cs released is deemed acceptable so some long term contamination is allowed.

Fukushima

In Figure 3 and Figure 4 some examples of evacuation zoning and contamination level of Cs-137 are shown. A maximum dose rate of 20 mSv/year will be allowed in zones were evacuation order are to be lifted. The contamination level in Figure 4 has been used estimated the doses received after the Fukushima accident by WHO [20].

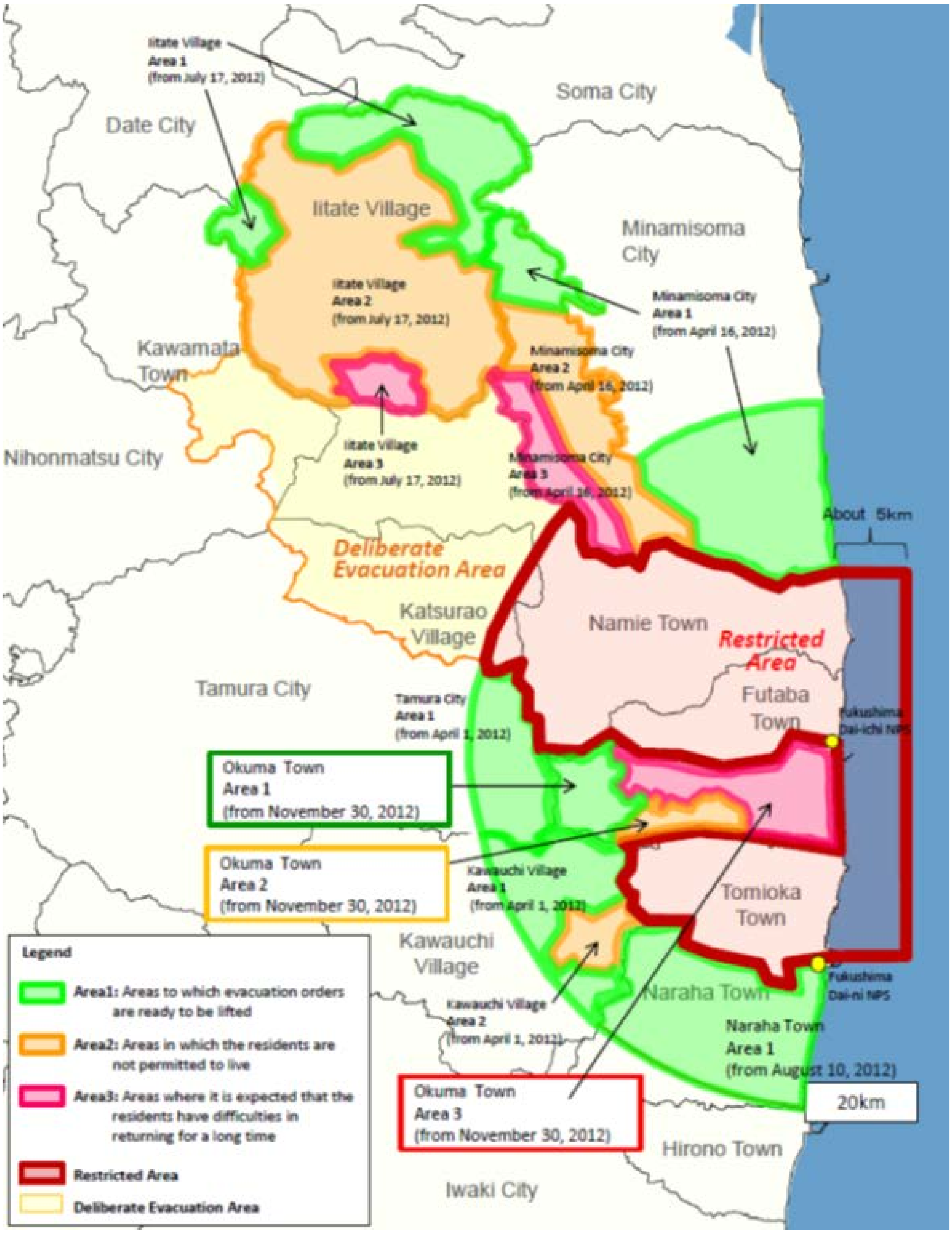

Figure 3. Restriction areas in Fukushima at the end of 2012 related to dose. Green < 20 mSv/year, Orange 20–50 mSv/year, Red > 50 mSv/year (source the Ministry of Economy, Trade and Industry (METI) in Japan).

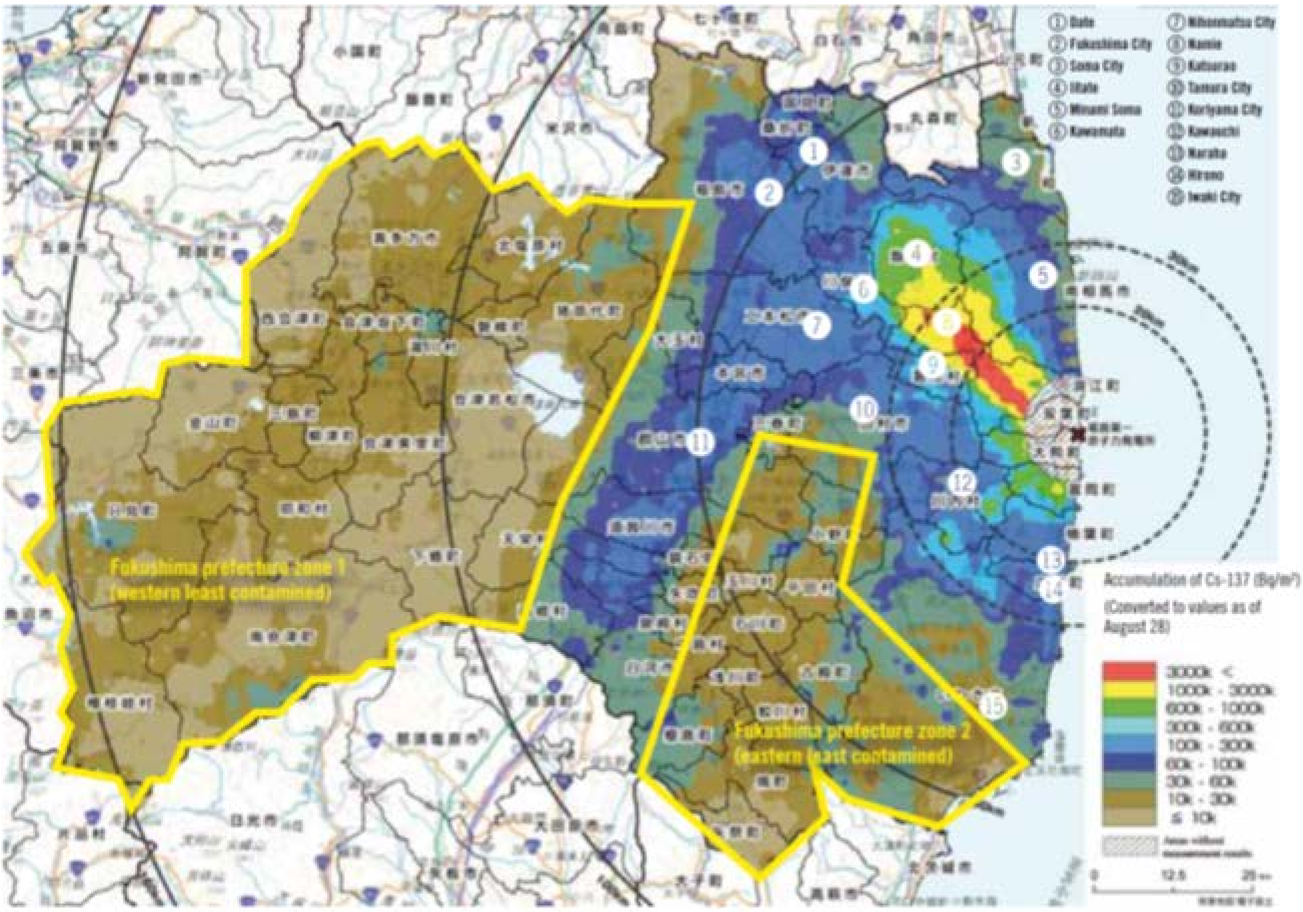

Figure 4. Accumulation of Cs-137 around Fukushima at the end of 2012 [20].

4.4.2 Identified metrics

Different levels of contamination can be used. One level of contamination could result in a restriction for living within a certain area and another level of contamination could result in restrictions from farming and harvest within a certain area.

The following metrics related to environmental impact are identified:

- Ground contamination level due to Cs-134 and Cs-137 [Bq/m^2] or [mSv/year]
- Non-usable area of land and sea [km^2].

4.4.3 Conversion between contamination level and dose

Once the contamination level is estimated [kBq/m^2] a dose rate can be estimated using conversion factors. Dose conversion factors can be found in [21] for different radionuclide, e.g., for Cs-134 the conversion factor is 5.4E-6 mSv/h/(kBq/m^2).

4.4.4 Contamination of different types of land

The contaminated land area based metric could be further refined into different types of land. Main categories are populated areas, sea, non-populated areas (wildlife) and agricultural areas (farming). The weighting between different types of land is a matter of economic impact metric (see Section 4.5).

It should be further noted that this metric is dependent on the contamination or dose based criteria applied for the restrictions in land use and food consumption.

4.4.5 Advantages, disadvantages and uncertainties

Similar to a dose related risk metric, it is rather straightforward to calculate an environmental impact metric at least in terms of affected land area (sea may be more challenging). This metric can be further refined from the time perspective point of view (temporary land use restrictions and long term restrictions) and the type of land point of view. Environmental impact metric is in many respects closely related to the health impact metric and these two metrics could be evaluated in an integrated manner. Environmental impact metric thus compensates part of the disadvantages of health impact metric.

The disadvantage is that there is not yet any commonly agreed approach to weigh different environmental impacts. A single number measuring the area of restricted land use does not reflect the differences between site locations, and it is also dependent on the land use restriction criterion. Type of land and time period of impact are relevant factors to be taken into account, but then conversion factors need to be defined if the results are to be compared. This leads to the definition of an economic impact metric.

The uncertainties connected with environmental impact are the general uncertainties with respect to dispersion calculations as well as the estimation of the long term impact on environment. The first issue is common to all other risk metrics, and the second one depends on the quality of environmental impact models. In practice, there should be sufficient input data for environmental impact estimation but the models include uncertainties, e.g., given that the release and dispersion can be calculated and given that the characteristics of the contaminated land area are known, it may be difficult to predict the time periods for land use restrictions and the significance for biosphere. Release to sea or river is even more complex to quantify but the air pathway is usually much more important than the sea pathway. Uncertainties are thus related to the definitions of the surrogate environmental impact metric that need to be applied.

4.5 Economic impact

In the following section metrics related to economic impact are discussed.

4.5.1 Input

OECD/NEA

In late 1990's, en expert group established by OECD/NEA prepared a guidance document for the consequence assessment of nuclear accidents [13]. The document provides a number of cost elements to be accounted (see Table 3) and discussion on cost assessment perspectives.

The economic effects associated with these consequences can be generally classified into two categories: direct and indirect. Direct economic consequences can be described in terms of cost of the implementation of countermeasures. The indirect economic consequences would cover the effects which are produced out of the areas directly impacted by the contamination, as for instance the impact on non-contaminated food marketing, on tourism, or on the nation's nuclear programme. Indirect consequences are normally difficult to quantify a priori, but they are amenable to an a posteriori evaluation. The report provides some examples of previous cost estimates, cost assessment approaches and a review of models and codes [13]. Rather obvious conclusions of the report are that there is no single cost of an accident and there is a large variation in the estimates.

There is also an ongoing activity at OECD/NEA to develop methodologies for estimating the costs of nuclear accidents. An expert group was established in 2013, and a study is expected to be finalised by the end of 2014.

Table 3. Cost categories of nuclear reactor accidents [13].

On-site Costs
Cost of decommissioning and decontamination Loss of capital (e.g. installed capacity)
Cost of countermeasures to reduce doses
Population movement - Transport away from the affected area - Temporary accommodation and food - Supervision of the evacuated area and monitoring of people - Loss of income for people unable to reach the workplace - Lost capital value and investment on land and property - Psychological effects of worry and upheaval Agricultural restrictions and countermeasures Decontamination - Cost of cleaning process, including the necessary equipment and materials, and the disposal and transportation of generated waste - Cost of labour - Cost of health effects induced in the workforce
Radiation-induced health effects in the exposed population
Cost of radiation-induced health effects: (1) early effects, (2) latent effects, (3) hereditary effects - Direct health care costs - Indirect costs, due to the loss of earnings during treatment and convalescence or of the total - Non-monetary costs, such as pain, grief and suffering associated with each effect
Psychological effects
Impact on the activity with which the installation is associated, for example the power programme
Impact on economic factors: employment, revenues, losses of capital, etc.
Long-term social and political impact
Environmental and ecological impact

Institut de Radioprotection et de Sûreté Nucléaire (IRSN)

IRSN has done a work on estimating the costs of nuclear accidents [15]. The work states that cost estimates should be comprehensive and if cost estimates are underestimated the value of accident prevention will also be underestimated. IRSN opposed the "consequence" approach which implies "zero Becquerel = zero cost" and the "economic" approach which considers a complete list of the effects of nuclear crisis including some cost items which correspond to zero Becquerel situations. Cost of an accident is divided into:

- On-site costs
- Off-site radiological costs
- Contaminated land areas
- Image costs
- Costs related to power production.

4.5.2 Identified metrics

The following metrics related to economic impact are identified:

- Total cost of accident, EUR.

4.5.3 Estimation of different economic impact

Estimation of different economic inputs consists of two major issues: selection of impacts to be included in the estimation and the conversion factors for non-monetary impacts (impacts primarily estimated in non-monetary scales).

Ideally, all costs of an accident should be accounted for, but this is practically impossible due to the multitude of stakeholders involved. Some perspective should be chosen for the estimation, e.g., the utility, the nuclear industry, the power production industry, or national level impact. Global impact is very difficult to estimate and may not be meaningful.

The list of economic impacts considered in the OECD/NEA [13] or IRSN study [15] could be used as references.

Depending on the decision making perspective, some of these costs may be ignored. In Task 0 of this project insurance companies expressed that the cost of loss of capital and image should not be included since it is a chosen risk for the company to act in the nuclear area [8]. Cost of loss of capital and image are, on the other hand, of major interest for the nuclear organisations.

4.5.4 Advantages, disadvantages and uncertainties

Economic impact has the obvious theoretical advantage that all impacts of an accident can be converted into a single metric, which allows consistent risk comparisons and cost-benefit analyses. In principle, this kind of risk metric should be applied in decision making, while the other risk metrics are surrogates to it.

In practice, it can be difficult to agree on what should be included in the quantification of economic impact and how to convert different impacts to a monetary value. This is a general problem for risk decision making and not specific to nuclear power plant risk analysis, although nuclear accidents have specific complicating aspects such as the multitude of impacts, significant number of stakeholders and the low probability of an accident.

Despite the difficulties evaluating economic impact, it should be sufficient to estimate the order of magnitude of different kinds of accidents, e.g., the TMI type of core damage accident with practically no external release would mean certain economic impact. Depending on the order of magnitude of release and direction of dispersion some other orders of magnitude of economic impact could be assumed. Knowledge from costs of other natural or industrial catastrophes could be also used as references to estimate the order of magnitude of a nuclear accident.

Despite the possible difficulties with converting non-monetary impacts to monetary scale, it might nevertheless be useful to try find some commonly agreed conversion factors. This process should lead to increased understanding of risks and facilitate risk communication. Given an economic impact assessment with explicit (parameterized) conversion factors, it is always possible to do sensitivity studies to determine the items that would be most critical to the economic impacts – even in the presence of uncertainties. An example of a multi-criteria decision analysis related to health, environmental, economic and societal impacts can be found in [22].

Since the economic impact assessment includes any consequences, the range of uncertainties is large and covers all kinds of uncertainties from the incompleteness issues, modelling uncertainties to parametric uncertainties.

4.6 Risk metrics for different stakeholders

Different stakeholders may need different risk metrics. Health effect and environmental impact metrics should be relevant to all stakeholders, but the way economic impact is assessed is more stakeholder dependent. The issue of selecting risk metrics for different stakeholders is thus mainly the question which costs are taken into account and in which way they are weighted. For instance, the safety authority may not necessarily want to take any position on the economic impact, while the utility and the insurance company may look at the economic impact on different risk perspectives.

It may be assumed that the Level 3 PSA is primarily done by the licensee and it would be advisable to consider a wide range of risk metrics. The aggregation of different impacts (health effect, environmental impact, economic impact) into a single impact metric should be done explicitly with parametric models, which allows different weightings. The issue of selecting impact metrics can be reduced to a discussion on weightings of various impacts.

5 Comparison with Level 1 and 2 PSA risk metrics

The risk metrics related to Level 1(core damage frequency) and 2 (unacceptable release frequency) PSA are to large extent not dependant on the siting (location) of the plant. The only impact from the location of the site in Level 1 and 2 PSA is from the determination of external events which to some extent are dependent of the location. In Level 3 PSA the location of the site is of paramount importance since e.g. metrological data and distance to population and agriculture areas are affecting the output. Hence, Level 3 PSA can give useful information about siting issues. Basically, Level 1 PSA analyses the plant systems which are designed to prevent core damage and Level 2 analyses the plant systems design to prevent and mitigate the consequences of a severe accident. Level 3 PSA will give useful information about both off-site emergency response or preparedness and plant safety systems.

Risk metrics for Level 2 PSA can be applicable as surrogates for Level 3 PSA risk metrics. There is a strong correlation between the release magnitude/timing metric and the health effect/environmental impact risk metrics. The correlation is site-specific. In practice, at a certain site it is only the effect of dispersion and evacuation which give variation in the consequence scale given certain release category.

The core damage Level 1 PSA risk metric is not a sufficient surrogate risk metric for Level 3 PSA purposes. On the other hand, if economic impact will be considered in level 3 PSA, it would be consistent to consider economic impacts even at level 1 PSA, i.e., to expand the consequence categories of Level 1 PSA to include even major economic losses (without a core damage). From the risk comparison point of view, there may be economically significant consequences without external release or even without core/fuel damage.

6 Conclusions and suggestions

A risk metric has two components: 1) probability metric and 2) consequence (or impact) metric. Regarding the probability metric, it is matter of choosing the normalization unit for risk comparison purposes. The consequence metric is associated with the impacts that are quantified in the consequence assessment part of Level 3 PSA.

The main probability unit used in the nuclear regulatory decision making is “probability per year per reactor”. Probability units “per lifetime” and “per produced energy (or electricity) over the complete fuel life cycle” can be considered in risk comparisons.

Table 4 summarises the main consequence metric categories (health effects, environmental impact and economic impact), their advantages, disadvantages and associated uncertainties as well as uses.

Table 4. Parameter, advantages, disadvantages, uncertainties and use for different consequence metrics.

	Consequence metric		
	Health effects	**Environmental impact**	**Economic impact**
Parameter or value	Dose [Sv] or [manSv] Fatalities (#) Short- and long-term effect	Contamination level [kBq/area] or [mSv/year] Restricted land and sea areal or "non-usable" land and sea areal (area)	Monetary units (e.g. [EUR] or [SEK]) Different costs are to be included depending on stakeholder (owner or insurance company)
Advantage	Relatively easy to estimate dose and connect dose to fatalities.	Relatively easy to estimate contamination of land and sea. Complements well the health effect based metric.	Most complete metric, everything is accounted for.
Disadvantage	Does not consider the total impact of a nuclear accident.	Contaminated area as a single metric does not characterise the site location. Use of multiple metric requires conversion factors between different environmental impacts.	Laborious to assess comprehensively and the impact is stakeholder dependent. May be difficult to agree on conversion factors for non-monetary costs.
Uncertainties	Long term health effect over the population is statistical estimate	Conversion factors between different environmental impacts	Large uncertainties in the estimation of cost. Which cost are to be included. How to estimate the cost of different factors. Political factors can affect the results.
Use	Improve plant design and emergency preparedness Requirements form authorities	Improve plant design and emergency preparedness Requirements form authorities	Improve plant design and emergency preparedness Communication with society Communication with insurance company Optimization of safety improvements

Health effects and environmental impact are rather similar metrics from the estimation and purpose point of view. The assessment of these metrics should be of interest for all stakeholders. It could be expected that even internationally the stakeholders could agree on which metric to use and risk criteria to be applied. At least for health effects, there are references for safety goals and associated numerical criteria. For the environmental impact, numerical criteria may not be necessary.

There are a number of open issues to be further explored, e.g., how far in time and place the estimations need to be done, i.e., what is the time frame for the risk metrics and how far away from the plant should the impact be accounted for? The pilot study, which is planned within the project [1], should elaborate more on these risk metrics when the scope of the study is determined. The pilot study should also elaborate how Level 2 PSA release category related risk metrics could be used as surrogates for Level 3 PSA criteria.

Economic impact is an ideal metric from decision making point of view and it would allow cost-benefit studies. In practice, it can be difficult to agree on what to include in the quantification of economic impact and how to convert different impacts into a monetary scale. Despite the difficulties to evaluate economic impact, one possibility could be to apply some simplified categorisation of economic impacts in terms of order of magnitude. It should be sufficient to estimate whether the cost is $\sim10^9$ € or $\sim10^{10}$ €. It is suggested that the pilot study should include at least a discussion of economic impacts within the framework of a licensee's risk analysis. This discussion should also explore conversion of non-monetary impacts to a monetary scale, e.g., doses and environmental impacts to euros. Main use of economic impact risk metric may be in cost-benefit assessments instead of being used in connection with numerical risk criteria.

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Working together
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Addressing off-site consequence criteria using Level 3 PSA

Task 2 - Regulations, guides and standards

Report for:
NPSAG / NKS-R

Report no: 21123-R01 Rev: 2

Date: 17 January 2014

Document history

Revision	Date	Description/changes	Changes made by
1	2014-1-17	Draft	Andrew Wallin Caldwell
2	2014-1-29	Final Version	Andrew Wallin Caldwell

Executive summary

The Level 3 PSA project as a whole aims to increase the knowledge in Level 3 PSA among NKS members and to set the frames for performing a state-of-the art Level 3 PSA. The objective of the work is to further explore the field of Level 3 PSA, in order to determine the driving forces for its utility, and by this, gaining experience and added quality to Level 1 and 2 PSA.

The purpose of Task 2 is to provide the ability to observe and influence the development of Level 3 PSA regulations, guides, and standards. This task will provide the environment to provide input to the Task 0 and Task 1 activities, as well as, provide feedback to external organizations based on the findings of the working group's activities.

This report represents the developments over the past year. In the project plan developed at the beginning of the project it was determined that the Task 2 activities would be focused on the first two years of the project.

Activities in this task include participation in the writing of an ANS/ASME Level 3 PSA standard, and participation in the recent IAEA Level 3 PSA activities.

The ANS/ASME Standard was intended to be the focus, however, since no in-person meeting took place during 2013 and modest progress over the past year a large majority of the work from members of the working group, to date in the area of the ANS/ASME 28.25 standard was provided in the thesis work referenced [1].

In light of the modest progress of the ANS/ASME standard over the past year, the IAEA activities in the field of Level 3 PSA have been actual focus of work of this Task. The IAEA held a technical meeting on Level 3 PSA in July of 2012. The attendees of the Technical Meeting provided guidance to the IAEA that further agency guidance would be useful for member countries. Subsequently, the IAEA provided a regional workshop in October 2013, and a Consultants Meeting in November 2013.

IAEA regional meeting included technical personnel from eastern European countries which were interested in Level 3 PSA. The IAEA hosted the meeting along with subject matter experts from the Netherlands, the UK, and Sweden. The meeting provided insight into developments in Level 3 PSA in eastern countries, which was relatively limited. The meeting also provided communication pathways to UK and The Netherlands which both have some Level 3 PSA work.

The objective of the IAEA Consultants meeting was to use the input from the IAEA 2012 Technical Meeting, and the IAEA Regional Workshop to develop an outline for creating a TECDOC, which may lead to further downstream agency publications (e.g. an IAEA Safety Series Guide (SSG)). Over the course of three days the attendees to the Consultant's meeting developed a large portion of the document outline. The IAEA intends to host further Consultant's meetings and eventually a Technical meeting in order to finalize the TECDOC and make final decisions on path forward for agency guidance on Level 3 PSA.

The work in the area of Level 3 PSA is poised to continue internationally. The project and specifically task 2 have provided resources for the working group and the project stakeholders to interact and influence some of these international developments.

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1 Introduction

The probabilistic assessment of off-site consequences, often referred to as Level 3 PSA, has been the subject of many large studies and international interest in the late 1980s, Organizations such as the IAEA, NEA, European Commission, and US NRC published reports or funded Level 3 PSA programs and studies. It was observed that very little has been done in the field since that time, but activities have started within some of these same organizations [1]. The purpose of Task 2 is to provide the ability to observe and influence the development of Level 3 PSA regulations, guides, and standards. This task has also provided input to the Task 0 and Task 1 activities, as well as, provided feedback to external organizations based on the findings of the working group's activities.

This report describes the work that has been performed within task 2 of the project over the past year, specifically, the work performed toward the ANS/ASME Standard 58.25, and two IAEA activities.

1.1 Background

Activity in the field of probabilistic offsite consequence analysis has had many peaks and valleys over the years. Internationally, and within the Nordic countries there was a large effort in the field of Level 3 PSA in the late 1980s, which included significant Probabilistic Consequence Analysis (PCA) methods work, large scope studies, and IAEA meetings and publications.

Several countries have been performing Level 3 PSA consistently for many years (e.g. the Netherlands, South Africa). However, generally speaking there was a significant drop-off in the work performed on Level 3 PSA methods and number of studies performed since the work of the late 1980s and early 1990s.

The interest in Level 3 PSA has risen in the last several years. This is based on several reasons, the fact that many of the large-scope well known studies are aging, the development and construction of new reactor units, and perhaps most significantly, the disasters at Fukushima. These reasons have prompted many in the Nuclear Safety Community to re-investigate Level 3 PSA.

1.2 Scope of work

At the onset of the project the primary focus of this task was to follow the ongoing work regarding the peer review standards ANS/ASME 58.24 (Level 2 PSA) and ANS/ASME 58.25 (Level 3 PSA). These standards have been under development in writing committee over the past several years. It is anticipated that it will take at least 1-2 years until these standards will be published. It was envisioned that this task will allow the project to influence and report on the progress of these standards.

The work performed under this task has also include monitoring and if possible participation in the development of international guides and regulations. This includes any developments made by the IAEA, the United State Nuclear Regulatory Commission, and similar organizations.

Finally, any additional, applicable regulations, and standards will be included in this task, particularly those identified in the work performed for Task 0 and Task 1. The extent that additional regulations and standards will be explored depends on the level of activity and involvement within the ANS and IAEA activities and available resources.

2 ANS/ASME Level 3 PSA standard 58.25

The ANS Standards 58.24 and 58.25 regarding Level 2 PSA and Level 3 PSA respectively have been under active development for several years. During this time a member of the working group has been actively involved in the 58.25 writing committee. This project will be integral in providing the resources to continue to engage in the ongoing work and report on the progress of these standards.

Since the work is relatively stagnant over the past year a large majority of the work to date in the area of the ANS/ASME 28.25 standard was provided in the thesis work provided in reference [1]. The following is an excerpt from that report:

The standard is being written by a committee of American Nuclear Society (ANS) and American Society of Mechanical Engineers (ASME) members. The committee was first funded and assembled in the early

2004. Since that time, a draft standard has been completed and released for review. To date, approximately 800 responses have been collected critiquing the draft version of the standard.

The ANS/ASME-58.25 standard provides requirements for application of risk-informed decisions related to the consequences of accidents involving release of radioactive materials to the environment. The consequences to be addressed include health effects (early and late) and longer term environmental impacts. These requirements are articulated for a range of technical Level 3 PSA areas in a specific structure. This structure is consistent with previously published ANS/ASME risk standards. The basis of this structure is built on the premise that the Standard is used to guide a Level 3 peer reviewer or auditor through the review of a Level 3 PSA analysis. This structure has proved useful for allowing some flexibility in applying PSA standards, which has limited the application of some standards and regulatory guidelines in the past. Examples of these issues have been experienced by the USNRC, which used a very prescriptive approach to probabilistic safety assessments (e.g. NUREG-1560), which was too restrictive and did not provide the flexibility for an analysis and provided little assistance in facilitating peer reviews.

The structure of the Level 3 standard, and the earlier PSA standards, is based on a hierarchy of technical elements and requirements. The framework for organizing the requirements first defines a set of *Technical Elements* of the analysis; Technical elements define significant fundamental tasks that are either important or necessary to perform for an analysis. For each technical element, High Level Requirements (HLRs) and subordinate Supporting Requirements (SRs) are defined. The High Level Requirements provide over-arching goals of each technical element. These HLR usually pertain to the data, modelling, and documentation, while the Supporting Requirements refer to specific actions while implementing the models, interpreting the data, or writing the documentation and presenting the results. Finally, each SR is divided into descriptions of minimum standards to fulfil three different "Capability Categories" (CCs). Each successive CC is defined for increased realism and site-specificity. Examples of the Technical elements are Release Categories, Protective Action Parameters, Dosimetry, Health Effects, and other broad processes that are integral to performing a Level 3 PSA. The Capability categories define somewhat specific details of the minimum requirement to achieve each of the three levels.

Since nearly all of the participants have a majority of their Level 3 PSA development or analysis experience with the MACCS code, as an unintended consequence, many of the Capability Category requirements were written such that a typical MACCS analysis would provide Capability Category II, where CC-I would be somewhat simplified analysis, and CC-III would stretch "beyond". Despite the group sharing a somewhat common background, the Standard Writing Committee actively made an effort to generalize the standard to apply to nuclear power, and non-nuclear power installations (e.g. fuel facilities, mars mission rocket launch accidents with nuclear payloads, etc.). This arises because the nature of Level 3 PSA and the fact that the analysis lends itself toward technology "neutrality." Most of the methodology integral to a Level 3 PSA is independent of the source of the radioactive release, given that source term information can be provided for the analysis. In practice, when drafting the standard and applying this neutrality, little of the language that was initially framed for reactor calculations had to be modified. While passages that required modification into a more general form did not greatly distort the standard. Yet, remaining general was a reoccurring difficulty in the drafting of the standard. A non-LWR (Light Water Reactor) application that was used as a reference to generalize the terminology of the Standard was a postulated rocket accident for an unmanned mission to mars. This program provided a basis for consideration of general requirements because the analysis was being performed with a different program than MACCS.

Another conscious decision of the writing committee was the omission of any particular consequence from being necessarily required for an analysis. Unlike the somewhat linear nature of the spectrum of extent of scenario development for Levels 1 and 2 analyses, like those shown in Figure 1. Rather, it was viewed by the writing committee that a Level 3 PSA analysis, a consequence analysis, does not necessarily imply health effects modelling and could be performed solely for economic or contamination purposes, an opinion not shared by all of the draft standard reviewers.

Figure 1. This diagram depicts the variation in scope that exists for the Spectrum of possible Levels 1, 2, and 3 PSAs. [2]

The nature of PSA standards separates them from other types of standards, for example standards pertaining to physical components (e.g. pressure vessels), or deterministic methodologies. PSA standards, therefore, define somewhat broad requirements and are entirely qualitative and rely on accurate terminology in the standard as well as interpretation by the users of the standard. This requires that the language of each requirement must be both precise and concise. The reviewers of the draft document had significant experience writing and reviewing PSA standards, perhaps more than much of the Standard Writing committee. This led to a healthy group dynamic, but also a rather large number of comments on the draft and the need for a significant revision following the initial balloting process.

The standard is progressing, but still undergoing major revision. The balloting of the draft standard has provided a substantial number of comments (approximately 800). This is, in part, because this standard is being written concurrently with a level 2 standard and ongoing addendums are being written for relevant-published standards, and this is also because the standard writing committee is relatively inexperienced with drafting standards. Due to the substantial volume of comments, and the extent of the revision to the draft Standard, it is somewhat apparent that the Standard will continue to be revised and reviewed for a few years before it is published. The draft document and specific text from the standard are not available for distribution at this time. [1]

3 IAEA activities in Level 3 PSA

Originally, it was envisioned that a significant portion of the work would be following the progress of the ASME/ANS 58.25 Level 3 PSA standard. The work in this area has been limited during the past year and a majority of the resources for this task of the project are being placed towards some of the IAEA agency activities surrounding Level 3 PSA.

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.

3.1 IAEA Consultant's Meeting (CM) on Level 3 PSA

An IAEA consultants meeting on Level 3 PSA took place in Vienna Austria from November 25-28, 2013.

The meeting included several individuals from countries with active Level 3 PSA projects.

The guidance from the attendees of the technical meeting guidance was that the IAEA should provide further guidance on Level 3 PSA. The purpose of the IAEA Consultant's meeting was to determine in what form the IAEAs guidance on Level 3 PSA should take.

The options that the group considered are shown in Table 1.

Table 1. Table of pros and cons of different possible TECDOC formats for Level 3 PSA.

Pros & Cons	Possible Options		
	~ Updated Safety Series No. 50-P-12	~ Document similar to TECDOC-1511 (2)	~ Report on overview of Current Practices (3)
Pros	- Useful introduction - Links to other refs - Good structure		- Overview of L-3 PSA methodology (blue book basis) - Examples (annexes) - Interface with Level 2 PSA - PCA codes - Motivation for Level 3 PSA - General principles of PSA (realism, etc.) -? Discussion on risk communication (optional) or Use of Level-3 PSA - ? Link to ASME PRA St on Level-3 PSA for the composition of chapters on the methodology
Cons	-Too many refs (+ -) -Obsolete refs	- Too much effort to provide quality - Not many explanations of WHY	
Comments			Start with blue book and extend Detailed guidelines (Pascal)
Decision			To produce a TECDOC - Use Blue Book on Level-3 PSA as a basis and provide up-to-date information - Extend to reflect on the recommendations of TM in July 2012 - Include use of Level-3 PSA and motivation for Level-3 PSA - Reflect on missing interface with Level 2 and Level 1 PSAs - Refer recent examples - Update references

			Note: After the TECDOC is done, the next step will be to develop a Safety Guide on Level-3 PSA

3.1.1 Participants

Ms Kuzmina Irina Ms Bobrova Ksenia Mr Lyubarskiy Artur	IAEA, SAS/NSNI	
Mr Caldwell Andrew Wallin,	Lloyd's Register Consulting - Energy AB	SWEDEN
Mr Compton Keith L.	U.S. Nuclear Regulatory Commission	USA
Mr Siu Nathan	U.S. Nuclear Regulatory Commission	USA
Ms Ghoorun Shivani	Koeberg Nuclear Power Station	SOUTH AFRICA
Ms Van Graan Henriette	National Nuclear Regulator	SOUTH AFRICA
Mr Steiner Pascal	Kernkraftwerk Gösgen-Däniken AG	SWIZERLAND

3.1.2 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.

3.1.3 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.[4].

3.1.4 Intended audience

This document is primarily intended to assist technical personnel with responsibilities in managing, performing or reviewing PSAs. The document is also intended to provide supporting information for users (e.g. decision makers) of Level 3 PSA results.

3.2 IAEA TC RER915 Regional Workshop on "Level 3 PSA development and related issues"

This meeting was valuable in showing the thoughts and competencies in the Eastern European region, as well as those from the other expert contributors, which were from the Netherlands, United Kingdom and Sweden.

This meeting marked the first IAEA Workshop on Level 3 PSA, and was held following the IAEA Technical Meeting on Level 3 PSA that took place in July of 2012, which was the first activity specifically discussing Level 3 PSA since the publication of the IAEA Safety Series No. 50-P-12 Publication titled, "Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 3).

The motivation for the meeting was due to the relative difficulty in finding information on Level 3 PSA. Due to this difficulty and many open questions in the Region, a 3-day workshop could provide significant insight into the basic constituents, uses, and scope of a Level 3 PSA.

3.2.1 Objective

The objectives of the meeting were stated by Artur Lyubarskiy:

- Present and discuss recent developments
- Current practices
- Application of Level 3 PSAs
    - Focus on NPP
- Available standards

3.2.2 Participants

The meeting included an IAEA representative and four subject matter experts:

- Artur Lyubarskiy (IAEA)
- John Preston (UK)
- Jacques Grupa (Netherlands)
- Andrew Wallin Caldwell (Sweden)

The meeting also included more than 30 participants from 15 different countries. These participants had either significant Level 1 & 2 experience, or deterministic radiological consequence analysis experience. There was almost no prior Level 3 PSA experience in the group of participating individuals.

Some of the countries represented were the following:

- Bulgaria
- Belarus
- Czech Republic
- Lithuania
- Russian Federation
- Slovenia
- Ukraine
- Armenia
- Croatia
- Hungary
- Netherlands (expert participant)
- Slovakia
- Sweden (expert participant)
- United Kingdom (expert participant)

3.2.3 IAEA & expert prepared presentations

The workshop consisted of lectures provided by the IAEA and subject matter experts, presentations provided by the participating countries, and a Question and answer section on the final day of the meeting. This section describes the two IAEA lectures, and five expert lectures.

3.2.4 Opening remarks & Output of the IAEA TM on Level-3-PSA

NRC Quality/Quantitative Safety Goals Specify Level 3 PSA type criteria as a basis, but the US along with most other countries do not require, or even regularly perform Level 3 PSA. Likewise, the IAEA has provided very little official documentation on the subject, and hasn't published anything on Level 3 PSA since the release of the Safety Series document No. 50-P-12, while, significant Agency guidance has been provided on Level 1 and Level 2 PSA.

To address this, a Technical Meeting was held in Vienna, Austria in July of 2012.

Finding from this Technical Meeting were the following:

There are many potential uses for Level 3 PSA

- Identification of cost-effective severe accident management measures
- Emergency planning and response
- Including siting for new NPPs
- Level 3 PSA can be a useful tool in risk communication.

The methodology is relatively well established.

The Agency should consider updating the Safety Series Document on Level 3 PSA because some of the guidance is no longer applicable, although the document is still very good, and relevant in its current state.

Fukushima has provided many lessons, and Level 3 PSA may have some potential areas of improvement.

- Consideration of multiple source terms
- Long duration releases
- Level 1 / Level 3 coupling of initiating events and off-site conditions

The main conclusions of the Technical meeting were the following.

- The Fukushima accident provided several insights into offsite consequence phenomena and areas for additional consideration in the context of Level 3 PSA

- The Level 3 PSA Methodology is application driven, and methods and level of detail can vary based on application
- Methods have advanced to a degree since the publication of the IAEA Safety Series No. 50-P-12, but not to a great extent
- Specific guidance for parameters and procedures is not universally defined
- Several important factors arise from upstream Level 1 and 2 PSA analyses and the interface between these levels, which are important for performing Level 3 PSA
    - Among these described were the specifics of the release such as the timing of the release,
    - Discussion on what defines a "representative set of source terms" for input to a Level 3 PSA was a very long and ongoing discussion.
- Motivation for and use of Level 3 PSA:
    - Many countries do not perform Level 3 PSA because there is no explicit regulatory requirement to do so, but there were many reasons identified for performing a Level 3 PSA. Some of the reasons expressed during the meeting were the following;
        - Using Level 3 PSA metrics for regulatory criteria (e.g. UK, the Netherlands) which often correlates quite directly to Quantitative Health Objectives
        - Input to Emergency Preparedness and Planning, (e.g. iodine prophylactic distribution)
        - New unit siting

Additional information about the Technical Meeting can be found in the Technical Meeting Report, which can be obtained through correspondence with the IAEA.

3.2.5 Participant presentations

Each of the participating countries provided presentations on the state of practice in terms of PSA, and radiological analyses. To a large extent these were limited to Level 1 and Level 2 PSA, which have been generally performed by each of the participating countries that currently have civilian nuclear power programs. A notable exception was Belarus, which has an active Level 3 PSA methods development program and is incorporating Level 3 PSA into the regulatory framework of Belarus.

A brief set of notes are provided for several of the presenting countries. Further detail of what was presented is provided in the presentation materials attached to the meeting report, and can be accessed upon request to the IAEA.

3.2.5.1 Hungary

Hungary does not currently impose Level 3 PSA requirements, but has a significant program for Level 1 & 2 PSA.

Interestingly, Level 3 PSA type calculation are required for all hazardous industries in Hungary with the exception of Nuclear installations. This gives the impetuous to explore such analyses because of the possibility for consistent regulations at sometime in the future. This being said, there is no current plan for Level 3 PSA.

Interest in bringing more state-of-the-art atmospheric dispersion methods toward Level 3 PSA.

3.2.5.2 Belarus

Criteria for Level 3 PSA in Belarus presented by Belarus was somewhat unclear. The presentation provided the following information:

General attributes corresponding Level 3 PSA:

risks to population criteria (according to this radiological system criteria):

The risk lies between 1.0E-6 and 1.E-4 (fatalities per year), which is recognized as limiting acceptable risk for population:

The implementation of the criteria, and decisions were expressed by Belarus with the following text: risk require the permanent control and special measures for its minimization

The levels between 1.0E-3 and 1.0E-4 (fatalities per year) is not acceptable for the population. A level less than 1.0E-6 is considered as negligible and do not require additional "minimizations".

Development of the program "RadRisk" is ongoing in Belarus, and it is intended to be used for conforming to the above mentioned criteria.

3.2.5.3 Slovenia

Very interested in uncertainties in PSA as a whole, and would be interested in Level 3 PSA type uncertainties.

3.2.5.4 Croatia

Croatia interest in Level 3 PSA for Emergency Planning Zone calculations. Currently, there is a difference in EPZ size between Slovenia and Croatia. Perhaps Level 3 PSA could be used as an objective tool for such a calculation.

Would the typography of the region cause issues with using current Level 3 PSA practices in Croatia?

3.2.5.5 Lithuania

No current Level 3 PSA criteria, calculations, and none currently planned.

3.2.5.6 Ukraine

No current Level 3 PSA criteria, calculations, and none currently planned.

3.2.5.7 Russia

No current Level 3 PSA criteria, calculations, and none currently planned.

Level 3 PSA calculations have been performed in the past.

3.2.5.8 Czech Republic

No current Level 3 PSA criteria, calculations, looking to begin work on Level 3 PSA during calendar year 2014. Accident scenarios have been defined.

3.2.5.9 Bulgaria

No current Level 3 PSA criteria, calculations, and none currently planned.

3.2.5.10 Poland

No current Level 3 PSA criteria, calculations, and none currently planned.

4 Other relevant regulations, guides, and standards

Any additional, applicable regulations and standards will be included in this task. The analyses and guidance used in the development of the containment venting systems that were designed for the Swedish and Finnish plants in previous decades is an item that has been identified at the beginning of this project.

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Appendix A

IAEA regional workshop question/answer

The questions were defined by meeting participants. Answers were formulated by Mr. Grupa and Mr. Wallin Caldwell. The following is an overview of the questions, and the formulated answers. Mr. Grupa produced a presentation describing his answers, which is included in the attached "documentnamedpartipantsquestionsandanswersJG.pdf"

Q: What should be the results of Level 3 PSA?

The results of Level 3 PSA can vary.

- Individual doses & individual risks, which can be further subdivided:
  - statistical distribution
  - position specific
  - cohorts - man, women, child
- Cumulative doses & risks
  - societal risks (number of early fatalities)
  - size of the area where early fatalities may occur (statistics)
  - collective doses
  - societal risks (number of late fatalities)
- Ground contamination
  - statistical distribution
  - position specific
  - size of the area where level X Bq/m2 is exceeded(statistics)
- Emergency measures and countermeasure effectiveness
  - Foodbans, sheltering, etc.
- Economic costs
  - Can include cost/benefit analyses for plant improvements

Q: What input data COSYMA software requires (For example weather condition change ….)

- source term: release fraction, release height, buildings, energy
- site data: radionuclide inventory, meterological data, site characteristics (flat, mountains), population data, land usage (agricultural), industries
- regional data: physical characteristics of people, food habits, living habits, radiobiological model food-chain
- national data: emergency response plan general data: plume growth for various turbulence classes, biophysical data (DCC, inhalation rates), radionuclide decay/chains

Q: Typically how many accident scenarios are analyzed for the Level-3 PSA purposes?

~20 in Dutch studies, but this can vary based on application

Q: Advanced models for dispersion (rather than Gaussian model) is needed to predict propagations beyond 40 km)

Other models are available, but not typically implemented in "fully-packaged" Probabilistic Consequence Analysis Codes. Programs such as COYSMA include some of these more computationally intense dispersion models, but they are not commonly implemented in fully probabilistic studies.

Beyond 40 km Gaussian plume is much less applicable. This is a driving factor why many Level 3 PSA studies do not extend further. This may change in the future, and more advanced models may be used for probabilistic analyses in the future.

Q: How we model changing weather conditions during long lasting releases (COSYMA)?

1. split release into release phases (usually one hour per phase)

2. use ¨measured*¨ hourly weather data (at least about five years)

Q: What is the difference between source terms/release categories:

There is general confusion between the term "Source Term" and "Release Categories". Release categories are groupings of Level 2 PSA results. These are often grouped phenomenologically, bypass sequences, early and late containment failure etc. The source term is the definition of release information for downstream dispersion analyses.

Source term in the Level 3 PSA sense of the word are the characteristics of a release, which include:

- Isotopics of release,(quantity and time history)
- Heat content of release
- Release Location (e.g. height)

A source term does not strictly need to be "probabilistically representative". Non-probabilistic source terms have been used for many off-site consequence deterministic studies. However, scommon practice for Level 3 PSA is to have "probabilistically representative" source terms which represent the range of source terms within the scope of the Level 3 PSA.

Release Categories are groups of sequences, divided into 10-20 categories. These grouping procedures are well documented in Level 2 PSA guides and standards. Often these Release Categories result in a single severe accident sequence analysis (MELCORE or MAAP analysis). The results of this analysis is usually used for the definition of a Level 3 PSA source term for the given Release Category.

Q: How we transfer Source terms to Level-3 PSA (e.g. height, energy, etc.). Level-3 PSA uses condensed information from Level-2 PSA on source terms (after several grouping process). How we can keep information on the initiating events that cause CD to Level-3 PSA (external hazards in particular)?

Maintaining Level 1 PSA conditions in Level 3 PSA is an issue that has gained attention after Fukushima. In general Level 1 PSA initiating events have been abstracted after compiling Level 2 PSA Plant Damage States, and also the subsequent consolidation into a set of Release Categories. The current state-of-practice does not maintain some of these external event/initiating event relationships on to Level 3 PSA.

Q: Groups of isotopes to be considered (16). What is the connection with release categories/source terms? HOW WE DEFINE REPPRESENATIVE ACCIDENT FOR FINAL ource term categories?

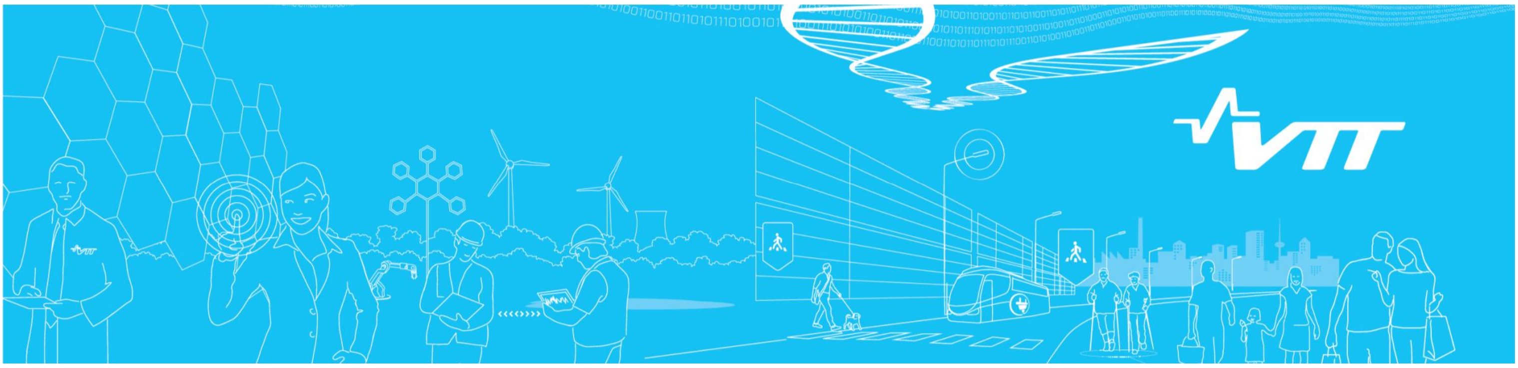
Isotopic grouping is usually already defined by the Probabilistic Consequence Analysis code used. What is important in Level 3 PSA may vary based on the application. Often we are interested in prompt fatalities, and latent fatalities. In these cases we a particularly interested in isotopes that drive these off-site health effects, but we must also be cognisant of release timing.

Q: Use of Level-3 PSA in Emergency Zones development (practical experience) + other application.

In principle Level 3 PSA results can be used to plan emergency measures, but other organisations are involved in national disaster plans. Often they follow their own track, using simple source terms and one or two deterministic weather types to set up and evaluate plans.

Consistency checks between license track, inspection track and

emergency track is useful but not often done systematically



RESEARCH REPORT

VTT-R-05661-14

Applying IDPSA in PSA level 3 – a pilot study

Authors: Ilkka Karanta, Tero Tyrväinen, Jukka Rossi

Confidentiality: Public

Report's title	
Applying IDPSA in PSA level 3 – a pilot study	
Customer, contact person, address VYR	**Order reference** SAFIR 42/2014
Project name Todennäköisyyspohjaisen riskianalyysin kehittäminen ja soveltaminen	**Project number/Short name** 85364/PRADA
Author(s) Ilkka Karanta, Tero Tyrväinen, Jukka Rossi	**Pages** 23
Keywords probabilistic safety analysis, consequence analysis	**Report identification code** VTT-R-05661-14

Summary

This report is a pilot study of applying deterministic and probabilistic methods in level 3 probabilistic safety analysis (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. The setup is as follows: the population of the major cities close to the site are 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. What radiological consequences (in terms of population dose and cancer deaths) would the radioactive release from the site have had under these presuppositions?

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 analyses.

Confidentiality	Public

Espoo 24.4.2015

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Preface

This report is a result of the PRADA project, which in turn is a part of the SAFIR 2014 research programme. On the other hand, the report is a part of Nordic cooperation in the "Addressing off-site consequence criteria using Level 3 PSA" project which has received funding from NKS and NPSAG.

Espoo 24.4.2015

Authors

Contents

1. Introduction

Integrated deterministic and probabilistic safety analysis (IDPSA) means the coupling of deterministic and probabilistic methods of safety analysis to address the mutual interactions of stochastic disturbances (e.g. equipment failure) and the deterministic response (transients) (Adolfsson et al. 2012). It has so far been used mainly in level 2 probabilistic safety analysis (PSA) of nuclear power plants.

Level 3 PSA concerns the consequences of a radioactive release from a nuclear power plant, or more precisely, the offsite dispersion and transport of radionuclides released to the environment and the health effects and other consequences of the postulated accidents (Lee and McCormick 2011).

From the beginning, both deterministic and probabilistic considerations have been a part of level 3 PSA (Bayer et al., 1981). However, the probabilistic side has usually been handled by adding random variables, with estimated or postulated probability distributions, to the deterministic variables describing e.g. atmospheric dispersion or dose of ionizing radiation to a group of people, and the analysis has normally been limited to Monte Carlo simulations and sensitivity analyses. Hence, the question arises whether more systematic and sophisticated probabilistic models might bring added value to level 3 analyses. This report describes an experiment in the application of event trees, throught IDPSA, to level 3 consequence analysis of health effects.

2. Goal

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:

- To illustrate how to apply a particular risk measure on level 3, namely the number of cancers resulting from a radioactive release.
- To enable comparison to the Swedish method of conducting level 3 PSA.
- 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.

3. Description of the pilot case

In this report, we do an exercise in alternative history: 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 (UNSCEAR 2013), 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 (UNSCEAR 2013).

Radionuclide	Total release (PBq) to the atmosphere
Te-132	29
I-131	120
I-132	29
I-133	9.6
Xe-133	7 300
Cs-134	9.0

Cs-136	1.8
Cs-137	8.8

For weather, we do not assume the weather of March 2011, but weather conditions in that part of Japan in March generally.

The evacuation proceeded in Fukushima as follows:

Table 2. Evacuation-related events in the Fukushima prefecture, March 2011 (UNSCEAR 2013).

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.

Name	Point of compass from Fukushima Daiichi	Distance from Fukushima Daiichi, kilometers	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

4. 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 – what the radiation doses to the population could 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, and with a release far more rapid than the actual – 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.

5. 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 (Savolainen and Vuori 1977), 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, first the event tree model is described in section 5.1, then the weather model (which gives the weather-related probabilities to the event tree) is explained in section 5.2, the evacuation and shielding models (which give the success probabilities of evacuation and shielding to the event tree model) are presented in section 5.3, and uncertainty distributions used in uncertainty analysis are presented in section 5.4.

5.1 Event tree model

The event tree model is presented in Appendix A. 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 5.3). 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.

5.2 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

A log-normal distribution was postulated for wind speed. From the Onahama statistics it was obtained that the average wind speed in March is 8 knots or 4.116 m/s. According to the site, the probability of wind speed exceeding 4 Beaufort in March is 19 %. 4 Beaufort is 20-28 km/h, and therefore this probability may be interpreted as the probability that wind speed exceeds 20 km/h = 5.556 m/s. For the log-normal distribution, the following two equations hold (Bury 1999):

$$\bar{x} = e^{\mu + \frac{\sigma^2}{2}} \tag{1}$$

$$F(x;\mu,\sigma) = \Phi\left(\frac{\ln(x)-\mu}{\sigma}\right) \tag{2}$$

Where $\Phi(x)$ is the standard cumulative normal distribution function, and μ and σ are the parameters of the log-normal distribution. These two nonlinear equations suffice to determine the parameters, noting that $\bar{x} = 4.116$ and $F(5.556;\mu,\sigma) = 1 - 0.19 = 0.81$. Through simple line search on equations on (1) and (2), it was found that the parameters of the log-normal distribution are approximately μ = 0.58144 and σ = 1.290995.

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. From the log-normal distribution with the parameters of the previous paragraph, the probabilities of the wind speeds are

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 range the wind speed used represents (m/s)	Probability of the wind speed range from the postulated log-normal distribution
0	0-4	0.733501384
8	4-12	0.196314116
16	12-	0.070185

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.

5.3 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.

5.4 Uncertainty distributions

To perform uncertainty analysis, uncertainty distributions are assigned to wind speed probabilities, wind direction probabilities, the probability of rain, population sheltering probability and the portion of the dose population is exposed when sheltered. With very little knowledge of the uncertainties, the uniform distributions presented in Table 6 are used.

Table 6. The limits of the uniform uncertainty distributions of the parameters.

Parameter	**Minimum**	**Maximum**
Probability of wind 16 m/s	0.0502	0.0902
Probability of wind 8 m/s	0.176	0.216
Probability of wind to Northwest	0.018	0.058
Probability of wind to West	0.007	0.047
Probability of wind to North	0.094	0.134
Probability of wind to South southwest	0.077	0.117
Probability of rain	0.158	0.358
Probability of sheltering	0.6	1.0
Portion of the dose population is exposed when sheltered	0.5	0.9

6. Results

The expected number of cancer deaths was 16. Table 7 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 7. Results of the event tree calculations.

Conditions	Northwest	West	North	South southwest
Wind 8 m/s, no rain, no sheltering	Prob = 1.1E-3 Cancers = 220	Prob = 7.7E-4 Cancers = 320	Prob = 3.3E-3 Cancers = 210	Prob = 2.8E-3 Cancers = 410
Wind 8 m/s, no rain, sheltering	Prob = 4.3E-3 Cancers = 150	Prob = 3.1E-3 Cancers = 220	Prob = 1.3E-2 Cancers = 150	Prob = 1.1E-2 Cancers = 290
Wind 8 m/s, rain, no sheltering	Prob = 3.7E-4 Cancers = 180	Prob = 2.7E-4 Cancers = 270	Prob = 1.1E-3 Cancers = 190	Prob = 9.6E-4 Cancers = 350
Wind 8 m/s, rain, sheltering	Prob = 1.5E-3 Cancers = 120	Prob = 1.1E-3 Cancers = 190	Prob = 4.6E-3 Cancers = 140	Prob = 3.8E-3 Cancers = 240
Wind 16 m/s, no rain, no	Prob = 3.9E-4	Prob = 2.8E-4	Prob = 1.2E-3	Prob = 1.0E-3

sheltering	Cancers = 220	Cancers = 320	Cancers = 210	Cancers = 410
Wind 16 m/s, no rain, sheltering	Prob = 1.6E-3 Cancers = 150	Prob = 1.1E-3 Cancers = 220	Prob = 4.7E-3 Cancers = 150	Prob = 4.0E-3 Cancers = 290
Wind 16 m/s, rain, no sheltering	Prob = 1.4E-4 Cancers = 180	Prob = 9.6E-5 Cancers = 270	Prob = 4.1E-4 Cancers = 190	Prob = 3.5E-4 Cancers = 350
Wind 16 m/s, rain, sheltering	Prob = 5.4E-4 Cancers = 120	Prob = 3.9E-4 Cancers = 190	Prob = 1.6E-3 Cancers = 140	Prob = 1.4E-3 Cancers = 240
Total	Prob = 1.0E-2 Cancers = 150	Prob = 7.2E-3 Cancers = 230	Prob = 3.0E-2 Cancers = 160	Prob = 2.6E-2 Cancers = 300
Expected cancers	1.5	1.7	4.8	7.8

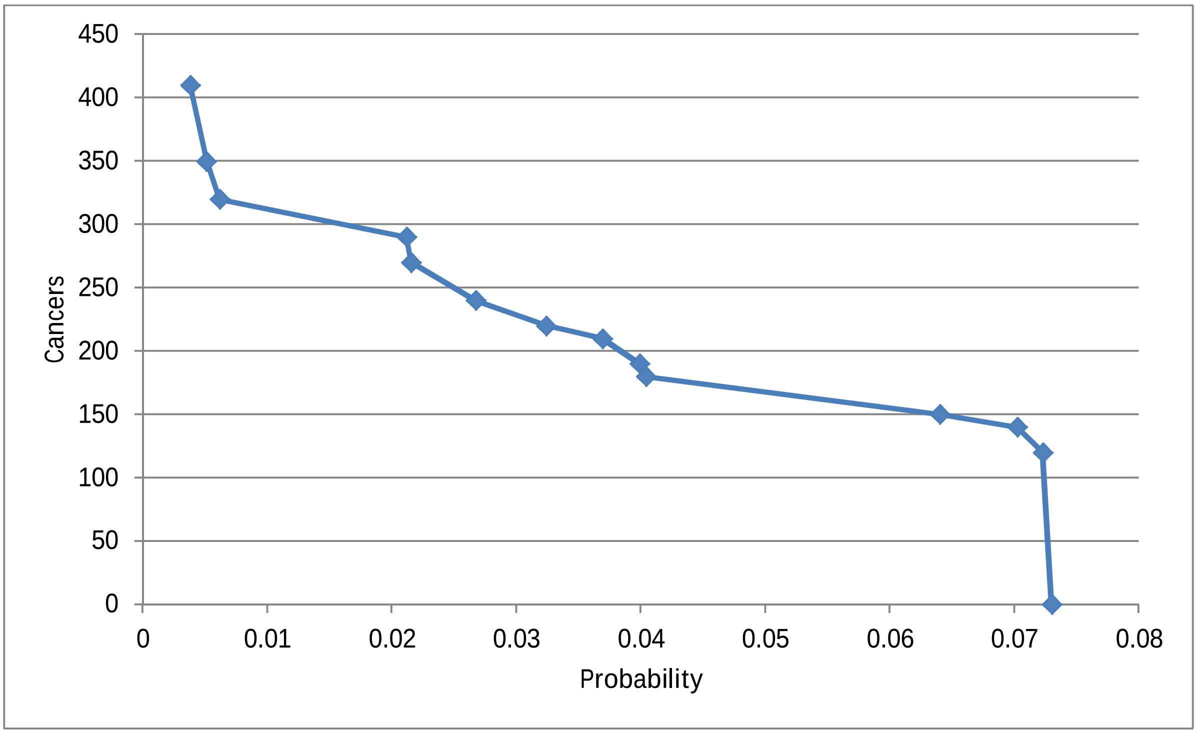

Figure 1: Farmer's curve.

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.

Uncertainty analysis with 10000 simulations based on uncertainties presented in Section 5.4 resulted with the cumulative distributions of cancer deaths and probability presented in Figures 4 and 5. The results indicate that the expected number of cancers is between 10 and 20 with a probability of 0.95 approximately. The probability of anyone getting a cancer is 0.1 at maximum.

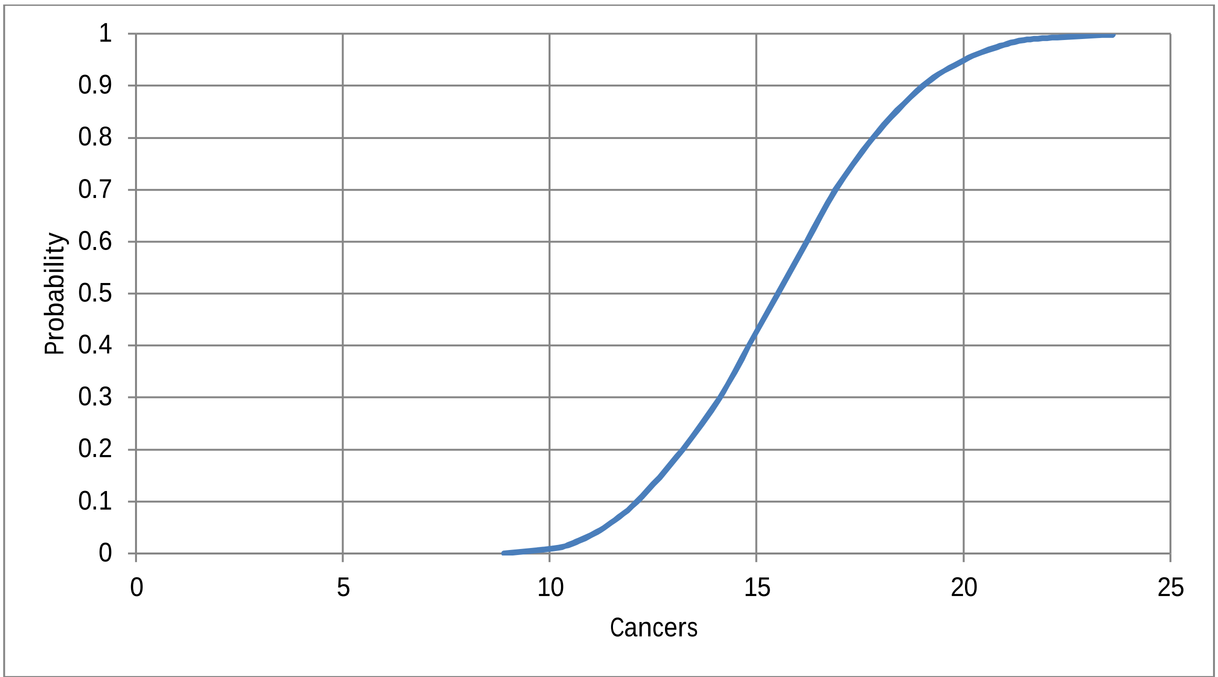

Figure 2: The cumulative uncertainty distribution of the number of cancers.

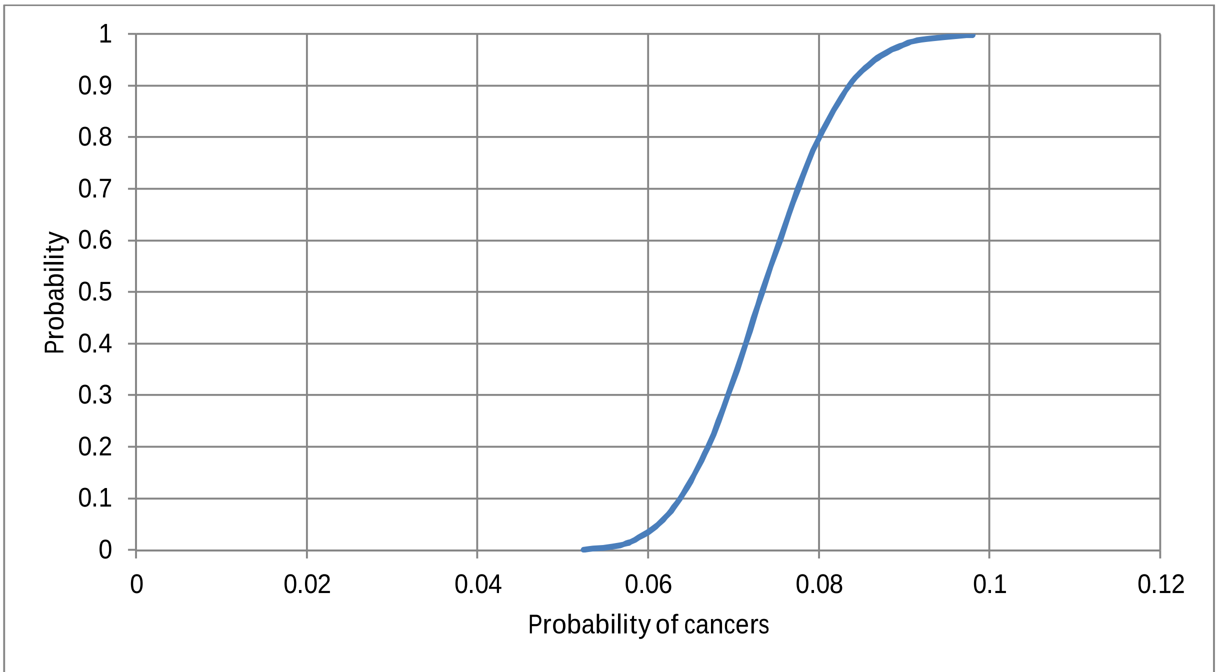

Figure 3: The cumulative uncertainty distribution of the probability of cancers.

7. 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.

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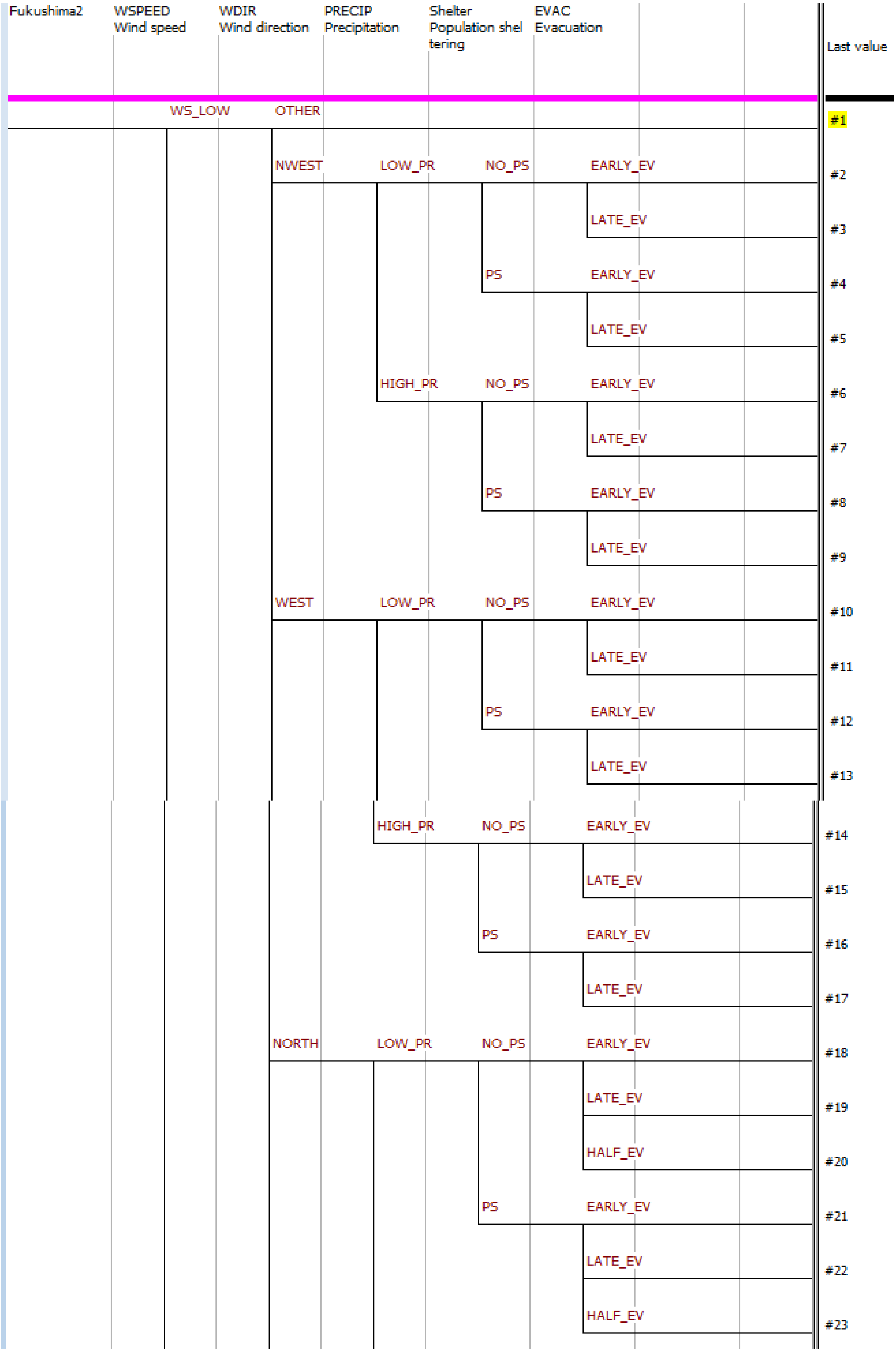
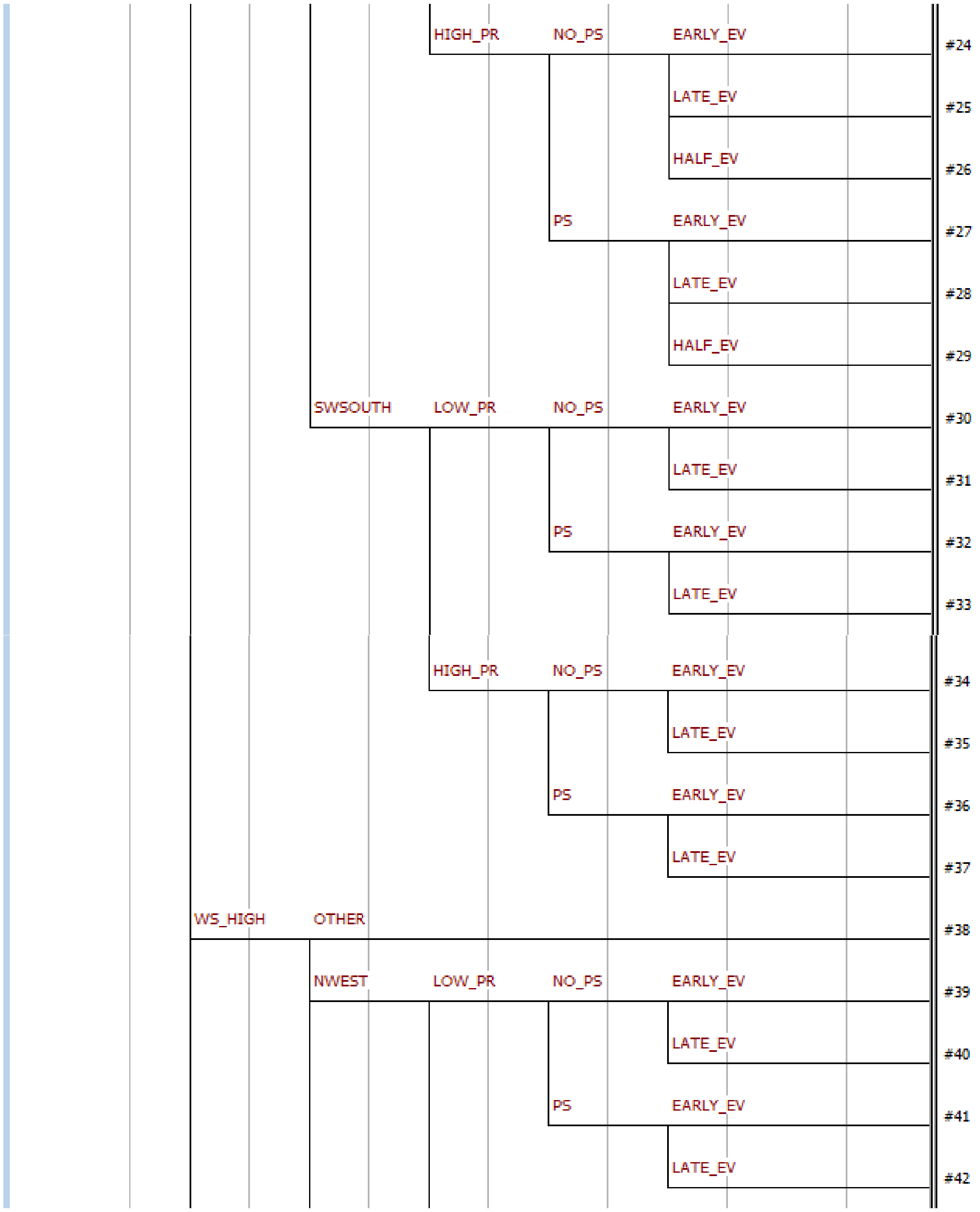
Appendix A

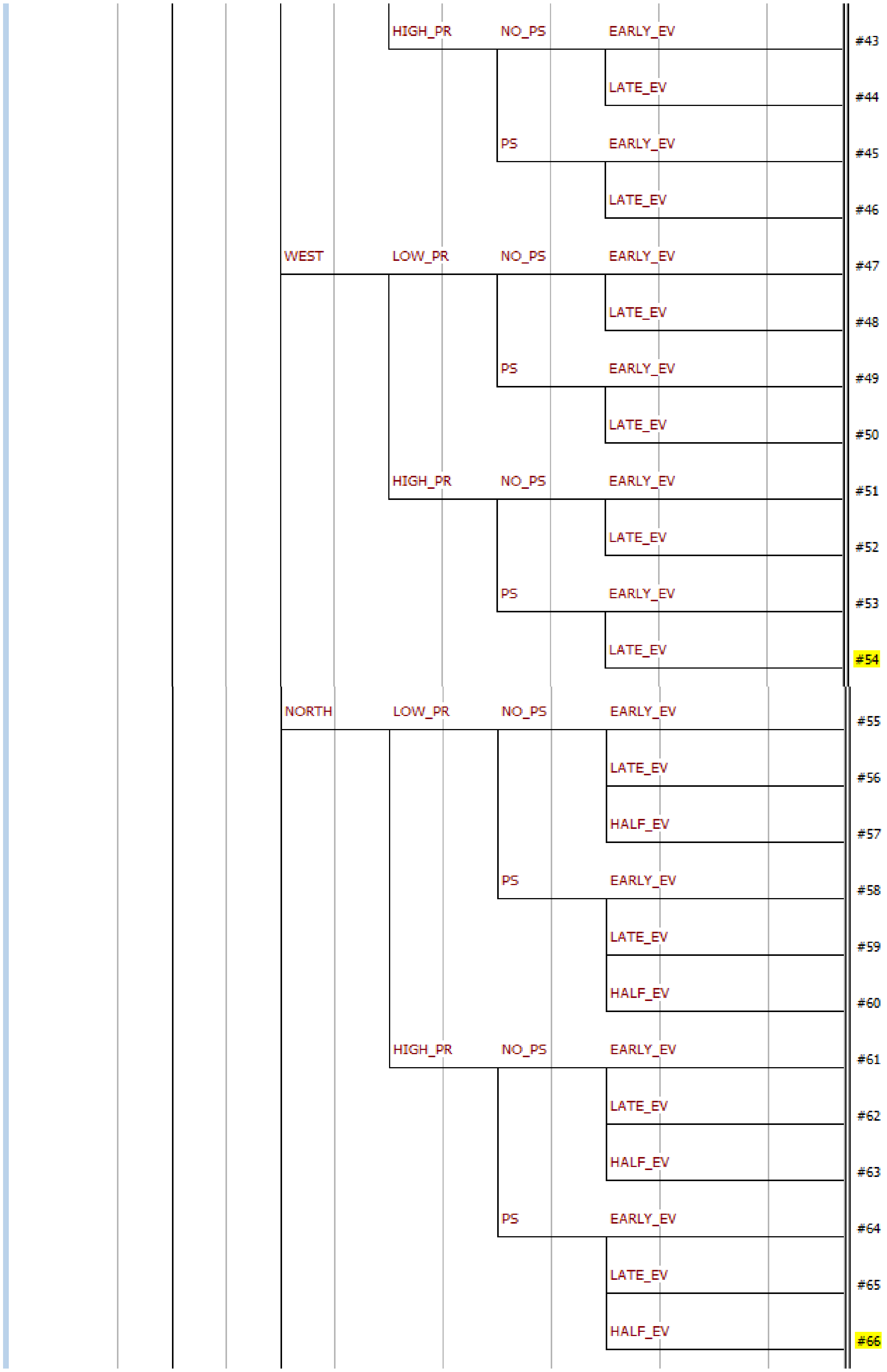
The event tree model is presented in the following. Some function names are explained in Table 8.

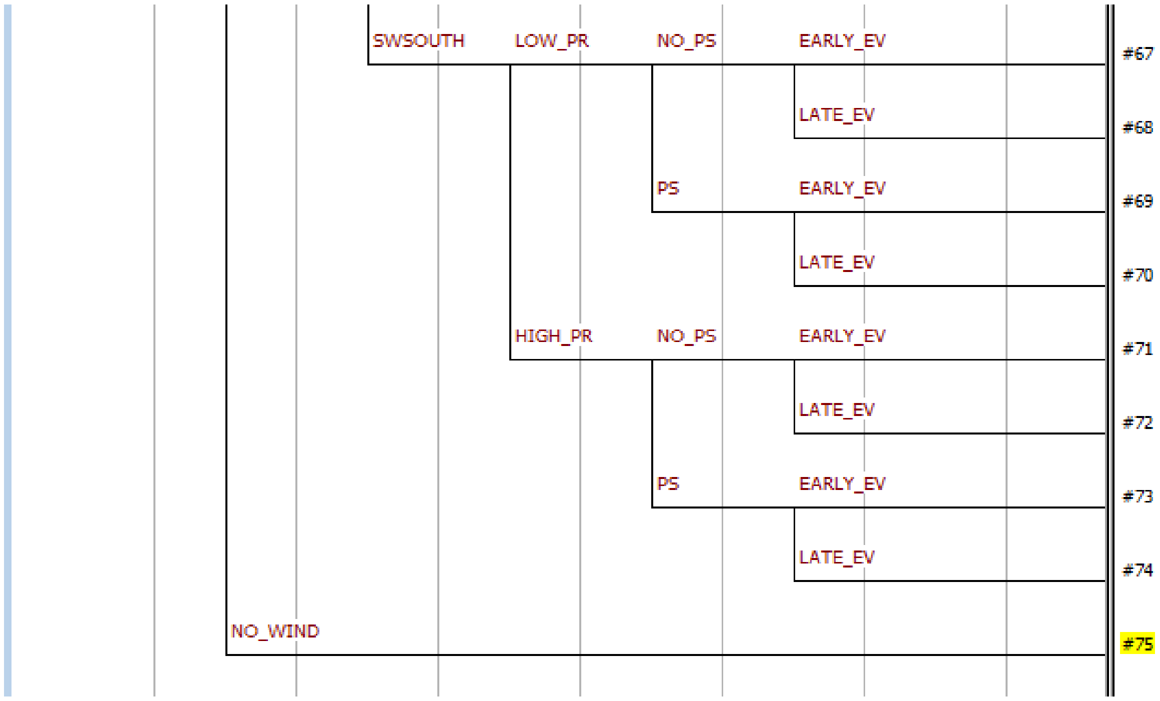
Table 8. Descriptions of functions in the event tree model.

Function	Desciption
WS_LOW	Wind speed 8 m/s
WS_HIGH	Wind speed 16 m/s
LOW_PR	No rain
HIGH_PR	Precipitation 5 mm/hour
NO_PS	No population sheltering
PS	Successful population sheltering
EARLY_EV	Evacuation successful
LATE_EV	No evacuation
HALF_EV	In the case wind direction north, Kakuda is evacuated but Minamisoma is not.


Fukushima2
WSPEED
Wind speed
WDIR
Wind direction
PRECIP
Precipitation
Shelter
Population shel tering
EVAC
Evacuation
Last value
WS_LOW
OTHER
#1
NWEST
LOW_PR
NO_PS
EARLY_EV
#2
LATE_EV
#3
PS
EARLY_EV
#4
LATE_EV
#5
HIGH_PR
NO_PS
EARLY_EV
#6
LATE_EV
#7
PS
EARLY_EV
#8
LATE_EV
#9
WEST
LOW_PR
NO_PS
EARLY_EV
#10
LATE_EV
#11
PS
EARLY_EV
#12
LATE_EV
#13
HIGH_PR
NO_PS
EARLY_EV
#14
LATE_EV
#15
PS
EARLY_EV
#16
LATE_EV
#17
NORTH
LOW_PR
NO_PS
EARLY_EV
#18
LATE_EV
#19
HALF_EV
#20
PS
EARLY_EV
#21
LATE_EV
#22
HALF_EV
#23


HIGH_PR
NO_PS
EARLY_EV
#24
LATE_EV
#25
HALF_EV
#26
PS
EARLY_EV
#27
LATE_EV
#28
HALF_EV
#29
SWSOUTH
LOW_PR
NO_PS
EARLY_EV
#30
LATE_EV
#31
PS
EARLY_EV
#32
LATE_EV
#33
HIGH_PR
NO_PS
EARLY_EV
#34
LATE_EV
#35
PS
EARLY_EV
#36
LATE_EV
#37
WS_HIGH
OTHER
#38
NWEST
LOW_PR
NO_PS
EARLY_EV
#39
LATE_EV
#40
PS
EARLY_EV
#41
LATE_EV
#42


HIGH_PR
NO_PS
EARLY_EV
#43
LATE_EV
#44
PS
EARLY_EV
#45
LATE_EV
#46
WEST
LOW_PR
NO_PS
EARLY_EV
#47
LATE_EV
#48
PS
EARLY_EV
#49
LATE_EV
#50
HIGH_PR
NO_PS
EARLY_EV
#51
LATE_EV
#52
PS
EARLY_EV
#53
LATE_EV
#54
NORTH
LOW_PR
NO_PS
EARLY_EV
#55
LATE_EV
#56
HALF_EV
#57
PS
EARLY_EV
#58
LATE_EV
#59
HALF_EV
#60
HIGH_PR
NO_PS
EARLY_EV
#61
LATE_EV
#62
HALF_EV
#63
PS
EARLY_EV
#64
LATE_EV
#65
HALF_EV
#66



The initial section:

```
real cancers,
     pdose,
     ws,
     dist1,
     dist2,
     time1,
     time2,
     shfactor

boolean popshe1,
        rain,
        popshe2

string dir

source cancers

routine init
  BinFreq = 1
return

routine finish
  cancers = pdose*0.05
return

class dir
routine binner active
  ('NWest',    'Expo'),
  ('West',     'Expo'),
  ('North',    'Expo'),
  ('SWSouth',  'Expo'),
```

```
  ('Other',    'Other')
return
```

WSPEED section:

```
real wsh, wsl

routine init
  wsh = raneven(0.050185, 0.090185)
  wsl = raneven(0.176314116, 0.216314116)
return


function real WS_HIGH
  ws = 16*0.001*60*60
return wsh

function nil NO_WIND
  ws = 0
  dir = 'Other'
return nil

function real WS_LOW
  ws = 8*0.001*60*60
return wsl
```

WDIR section

```
real nw, w, n, sws

routine init
  nw = raneven(0.018, 0.058)
  w = raneven(0.007, 0.047)
  n = raneven(0.094, 0.134)
  sws = raneven(0.077, 0.117)
return

function real NWEST
  dist1 = 64
  dist2 = 0
  dir = 'NWest'
return nw

function real WEST
  dist1 = 56
  dist2 = 0
  dir = 'West'
return w

function real NORTH
  dist1 = 27
  dist2 = 58
  dir = 'North'
return n

function real SWSOUTH
  dist1 = 48
```

```
  dist2 = 0
  dir = 'SWSouth'
return sws

function nil OTHER
  dist1 = 0
  dist2 = 0
  dir = 'Other'
  pdose = 0
return nil
```

PRECIP section

```
real hp

routine init
  hp = raneven(0.158, 0.358)
return

function real HIGH_PR
  rain = true
return hp

function nil LOW_PR
  rain = false
return nil
```

SHELTER section

```
real sp, sf

routine init
  sp = raneven(0.6, 1)
  sf = raneven(0.5, 0.9)
return

function real PS
  time1 = dist1/ws
  time2 = dist2/ws
  shfactor = sf
return sp

function nil NO_PS
  time1 = dist1/ws
  time2 = dist2/ws
  shfactor = 1
return nil
```

EVAC section

```
real l1, l2

function nil LATE_EV
  if samestr(dir, 'NWest') then
  begin
     if rain then
     begin
```

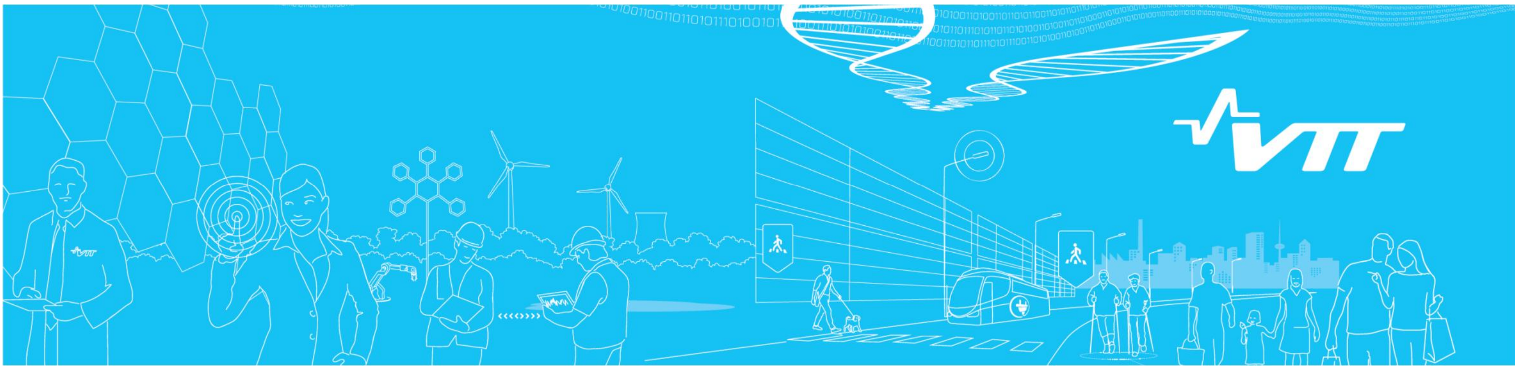
```
    if more(ws, 10) then
    begin
      pdose = shfactor*3528
    end
    else
    begin
      pdose = shfactor*3822
    end
  end
  else
  begin
    if more(ws, 10) then
    begin
      pdose = shfactor*4410
    end
    else
    begin
      pdose = shfactor*6174
    end
  end
end
else if samestr(dir, 'West') then
begin
  if rain then
  begin
    if more(ws, 10) then
    begin
      pdose = shfactor*5408
    end
    else
    begin
      pdose = shfactor*5746
    end
  end
  else
  begin
    if more(ws, 10) then
    begin
      pdose = shfactor*6422
    end
    else
    begin
      pdose = shfactor*8788
    end
  end
end
else if samestr(dir, 'North') then
begin
  if rain then
  begin
    if more(ws, 10) then
    begin
      pdose = shfactor*(3408+465)
    end
    else
    begin
```

```
        pdose = shfactor*(4828+496)
      end
    end
    else
    begin
      if more(ws, 10) then
      begin
        pdose = shfactor*(3692+558)
      end
      else
      begin
        pdose = shfactor*(5822+775)
      end
    end
  end
  else if samestr(dir, 'SWSouth') then
  begin
    if rain then
    begin
      if more(ws, 10) then
      begin
        pdose = shfactor*6900
      end
      else
      begin
        pdose = shfactor*8280
      end
    end
    else
    begin
      if more(ws, 10) then
      begin
        pdose = shfactor*8280
      end
      else
      begin
        pdose = shfactor*11385
      end
    end
  end
return nil

function real EARLY_EV
  l1 = time1/72
  pdose = 0
return l1

function real HALF_EV
  l2 = time2/72-time1/72
  if rain then
  begin
    if more(ws, 10) then
    begin
      pdose = shfactor*3408
    end
    else
```

```
      begin
        pdose = shfactor*4828
      end
    end
    else
    begin
      if more(ws, 10) then
      begin
        pdose = shfactor*3692
      end
      else
      begin
        pdose = shfactor*5822
      end
    end
return l2
```



RESEARCH REPORT

VTT-R-05819-15

Improvements to a level 3 PSA event tree model and case study

Authors: Ilkka Karanta, Tero Tyrväinen, Jukka Rossi

Confidentiality: Public

Report's title

Improvements to a level 3 PSA event tree model and case study

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VYR	SAFIR 4/2015
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Ilkka Karanta, Tero Tyrväinen, Jukka Rossi	21/10
Keywords	**Report identification code**
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Summary

This report presents implemented and potential improvements to a level 3 PSA model and its analysis previously developed in the PRADA project. The model is an event tree model, where wind direction, wind speed, precipitation, success of evacuation and success of sheltering (if evacuation is unsuccessful) are the nodes. The case modelled was an alternative take on the Fukushima Daiichi nuclear accident: what radiological consequences would the accident have had if the population in nearby big cities had been in place and not dislocated due to the tsunami. The radiological consequences were found to be small even under rather conservative assumptions.

The most important change to the event tree model introduced in this report is that wind speed is now drawn from a Weibull distribution in a Monte Carlo simulation. Also evacuation modelling was improved slightly.

In the uncertainty analysis, the most important change concerns the handling of source term uncertainties. Population dose is composed from doses caused by each radionuclide, and thus the amounts of radionuclides in the release can be made random variables and subjected to Monte Carlo simulation. An uncertainty distribution was attached also to wind speed parameters and evacuation success probability in the uncertainty analysis.

The radiological consequences to the general population were minor also in this study. This gives support to the hypothesis that the very small (according to UNSCEAR) radiological consequences to the general population in the Fukushima Daiichi nuclear accident were not a matter of good luck but rather something to be expected concerning the case.

Confidentiality	Public

Espoo 1.2.2016

Written by	**Reviewed by**	**Accepted by**
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Preface

This report was written as a part of PRAMEA project, which is a part of the Finnish Nuclear Power Plant Safety Research programme 2015-2018 SAFIR2018. The report is also a part of Nordic cooperation within the “Addressing off-site consequence criteria using level 3 PSA” project which has received funding from NKS and NPSAG. The authors wish to thank Dr. Mikko Ilvonen for advice.

Espoo 1.2.2016

Authors

Contents

1. Introduction

Level 3 probabilistic safety analyses (PSA) are a compromise between modelling and analysis accuracy on the one hand, and computational demands on the other hand. In (Karanta et al. 2015), a lightweight approach to level 3 PSA was presented. It consists of a PSA event tree model for weather and countermeasure variables, and utilizes a level 3 code named ARANO in calculating the population doses for each outcome. The model used is simple in many respects: weather is modelled as essentially static (no change in wind direction or speed), evacuation success is expressed as a single probability figure based on a very simple model, and sheltering success is also expressed as a single probability. An uncertainty analysis was conducted, but it did not incorporate uncertainty in the source term, and the uncertainty distributions of model variables were determined by judgment of the report authors.

The case was a study in alternative history: what if the Fukushima Daiichi nuclear power plant accident, with its source term, had happened without the earthquake and tsunami? In this case, the population of the Fukushima prefecture would have been in their homes and workplaces, whereas in reality, the population of large areas within the prefecture either had died in the tsunami, or had already been evacuated because of it. The motivation of this piece of alternative history was to shed light on the following question: was the near absence of radiological consequences in the area – according to UNSCEAR report (UNSCEAR 2013), no radiological deaths or cancers due to the accident have occurred nor will be expected within the next 85 years – due to the fact that large areas were already depopulated, plain good luck, or were the insignificant radiological consequences something to be expected, given the weather conditions in that part of Japan, the fact that Fukushima Daiichi is on the Pacific coast and thus approximately 50 % of wind directions result in negligible dose to the general population, and the effectiveness with which the Japanese officials carried out the evacuation of the area.

The results of that study indicated that at the distances of 27…64 km no acute health effects were expected because individual doses remained below 1 Sv. They also indicated that late health effects, measured by the number of expected cancer deaths, were minor. This supports the hypothesis that insignificant radiological consequences outside the evacuation planning zone are to be expected rather than being good luck.

This report considers improvements to the model and analyses of that report. Uncertainty analysis, which did not take uncertainty related to the source term into account in the original report, is improved upon in this regard. Dynamic models for weather, evacuation and sheltering are considered.

2. Event tree model

The consequences of the accident are analysed using an event tree model that utilises deterministic dispersion calculations. The event tree structure is quite similar to the event tree in the previous study (Karanta et al. 2015). The main difference is that wind speed is not divided into branches, but it is sampled from a distribution on each simulation round (see Section 4.1). In this way the complete wind speed distribution and its effect on the population doses can be included in the model. The event tree can be found in Appendix A.

The probabilistic analysis was performed using VTT's FinPSA Level 2 software (Mätäsniemi et al. 2015), while the previous study was performed using SPSA. Both tools work almost identically. The CETL (containment event tree language), a programming language integrated to FinPSA Level 2, was utilized in the implementation of the improvements; all the code samples in this report are written with it. The supporting deterministic computations were performed mainly with ARANO software (Savolainen and Vuori 1977).

3. Uncertainty analyses

The main improvements of the uncertainty analysis, when compared to (Karanta et al. 2015), were the incorporation of uncertainty in the source term in the uncertainty analysis, and the changes brought about by the improved handling of wind speed.

Most of the uncertainty distributions presented in (Karanta et al. 2015) were used in this analysis too, excluding wind speed related distributions. The following improvements were made:

- In the previous study, the number of cancer deaths was calculated from the population dose by multiplying it by 0.05. In the new model, instead of being constant, the factor was assumed to follow uniform distribution between 0.03 and 0.07.
- Wind speed uncertainties were handled as presented in Section 4.1.
- The uncertainties of the source term were handled as presented in this section.
- Uncertainty distribution was assigned for evacuation time as presented in Section 5.

3.1 Uncertainty analysis of source term

Usually, the uncertainty related to the source term is an output of level 2 PSA analyses. In our case, such analyses were not available, and the uncertainty in the source term had to be assessed by other means.

Several estimates for the source term uncertainty in the Fukushima Daiichi accident exist, all of them related to the estimation of the source term.

The UNSCEAR report takes its source term from (Terada et al. 2012), This paper does not contain a proper uncertainty analysis, but on p. 145 it says ”the mean differences in logarithm of measurements and calculations [..] and the standard deviations of the differences [..] are also shown in Figure 4”. That figure contains the standard deviations of a “refined model”, where surface deposition measurements have been taken into account, and they are 0.93 (I-131) ja 0.91 (Cs-137) (unit: Bq/m^2). However, these uncertainty estimates do not take measurement uncertainty and uncertainty resulting from sampling into account; furthermore, it is unclear how these estimates could be transformed into ordinary uncertainty estimates. Therefore they were not used.

Reference (Stohl et al. 2011) presents another estimate, an uncertainty range expressed for Xe-133: 12.2-18.3 EBq. Presumably this is the 95 % confidence interval of the amount of Xe-133 in the release. The trustworthiness of this estimate is reduced because their baseline estimate (15.3 EBq) is more than double the estimates obtained by other researchers, and the baseline estimate for Cs-137 is more than four times that obtained by other researchers, as noted also by the UNSCEAR report (UNSCEAR 2014). Thus, this uncertainty estimate was not used either.

Reference (Winiarek et al. 2014) states that “total released quantity of caesium-137 in the interval 11.6 - 19.3 PBq with an estimated standard deviation range of 15-20 % depending on the method and the data sets”. This uncertainty estimate appears to be the most plausible available because their source term is well in line with those obtained by other researchers. Thus it was used. Uncertainty estimates for all radioisotopes were set to 17.5% from their baseline estimates corresponding to the caesium-137’s 15-20 % uncertainty range.

3.2 Propagation of source term uncertainty to population doses

In the uncertainty analysis, it was assumed that if a source term is scaled with a particular factor, the population dose can be scaled with the same factor. At least, the computation in ARANO software works this way. Therefore, it was possible to perform the uncertainty analysis by scaling baseline results in FinPSA instead of performing Monte Carlo analysis in ARANO.

The population doses were calculated for each radionuclide separately using the baseline release values from (Karanta et al. 2015). Fractions of different radionuclides of the total population dose were examined in order to calculate a scaling factor to be used in uncertainty analysis. For each radionuclide, an uncertainty distribution was created with the fraction as the mean value. On each simulation round, a weight of each radionuclide was drawn from the distribution, and the weights were summed up to obtain a scaling factor of the source term uncertainty.

It was found out that the fractions of different radionuclides of the total population dose depended slightly on wind speed and distance (while the doses changed significantly as can be seen from Section 4.1). In the case of no precipitation, it was mainly the fraction of xenon that changed according to wind speed and distance, while other fractions were approximately scaled according to the fraction of xenon. The baseline fractions that were used in the uncertainty analysis are presented in Table 1. The dependence to wind speed and distance was modelled quite roughly because the effect on the total results was assumed to be small. Table 2 presents the mean fraction of xenon in different cases. In the analysis, the fractions of other nuclides presented in Table 1 were scaled according to the fraction of xenon so that the sum of fractions was 100%.

Table 1: Mean fractions of different radionuclides of the total population dose when there is no precipitation.

Radionuclide	Fraction (%)
Te-132	11
I-131	51
I-132	0.5

I-133	1
Xe-133	6
Cs-134	11.5
Cs-136	1
Cs-137	18

Table 2: Mean fractions of xenon of the total population dose in different cases when there is no precipitation.

City	v ≤ 1	1 < v ≤ 4	4 < v ≤ 8	8 < v ≤ 16	v > 16
Minamisoma	8%	6%	4%	3%	3%
Iwaki	13%	8%	5%	4%	3%
Koriyama	22%	8%	5%	4%	3%
Kakuda	22%	8%	5%	4%	3%
Fukushima	26%	8%	5%	4%	3%

The fractions of I-132 and I-133 were so small that the iodine nuclides were grouped together for the uncertainty analysis. The correlation of radionuclides was handled simply by a correlation factor that was common for Te-132, iodine, Cs-134, Cs-136 and Cs-137. For the correlation factor, normal distribution was assumed with the mean of 0.5 and standard deviation of 17.5% of the mean.

In the case of precipitation, it was assumed that the entire population dose comes from xenon, meaning also that the dose is much smaller than without precipitation (for justification, see Section 4.4). This is not true with very high wind speeds, but the assumption was made to keep the model simple enough. Very high wind speeds are quite unlikely, and the only effect of the assumption is that the uncertainty distributions are slightly wider than they would be with more accurate modelling.

The scaling factors (st2 for Minamisoma and st for other cities) were calculated using the following code:

```
  $ source term uncertainty computation
  cor = rannorm(0.5, 0.0875) $ correlation factor
  i = rannorm(0.525, 0.0919)
  te = (1-cor)*rannorm(0.11, 0.0193)+i*0.11/0.525*cor
  xe = rannorm(0.06, 0.0105)
  cs134 = (1-cor)*rannorm(0.115, 0.0201)+i*0.115/0.525*cor
  cs136 = (1-cor)*rannorm(0.01, 0.00175)+i*0.01/0.525*cor
  cs137 = (1-cor)*rannorm(0.18, 0.0315)+i*0.18/0.525*cor
  st = te+i+xe+cs134+cs136+cs137 $ factor of source term
uncertainty
  st2 = te+i+xe+cs134+cs136+cs137
```

```
  if rain then
  begin
    $ when it rains, the whole population dose comes from
xenon
    st = xe/0.06
    st2 = xe/0.06
  end
  else
  begin
    $ factor of source term uncertainty changed according to
wind speed
    $ the fraction of xenon depends on wind speed
    $ st2 is for Minamisoma and st for other cities
    if wind_speed > 16 then
    begin
      st = (st-xe)*103/100+xe*0.5
      st2 = (st2-xe)*103/100+xe*0.5
    end
    else if wind_speed > 8 then
    begin
      st = (st-xe)*104/100+xe*4/6
      st2 = (st2-xe)*103/100+xe*0.5
    end
    else if wind_speed > 4 then
    begin
      st = (st-xe)*105/100+xe*5/6
      st2 = (st2-xe)*104/100+xe*4/6
    end
    else if wind_speed > 1 then
    begin
      st = (st-xe)*98/100+xe*8/6
    end
    else
    begin
      if samestr(dir, 'NWest') then
      begin
        st = (st-xe)*80/100+xe*26/6
      end
      else if samestr(dir, 'SWSouth') then
      begin
        st = (st-xe)*93/100+xe*13/6
      end
      else
      begin
        st = (st-xe)*84/100+xe*22/6
      end
      st2 = (st2-xe)*98/100+xe*8/6
    end
  end
```

For each city and weather condition, the total baseline population dose was scaled by the scaling factor (st or st2) on each simulation round. The distributions of the scaling factor for Iwaki with and without precipitation are presented in Figures 1 and 2. The distribution is slightly wider in the case of precipitation: the 5th percentile was 0.71 and the 95th percentile was 1.29.

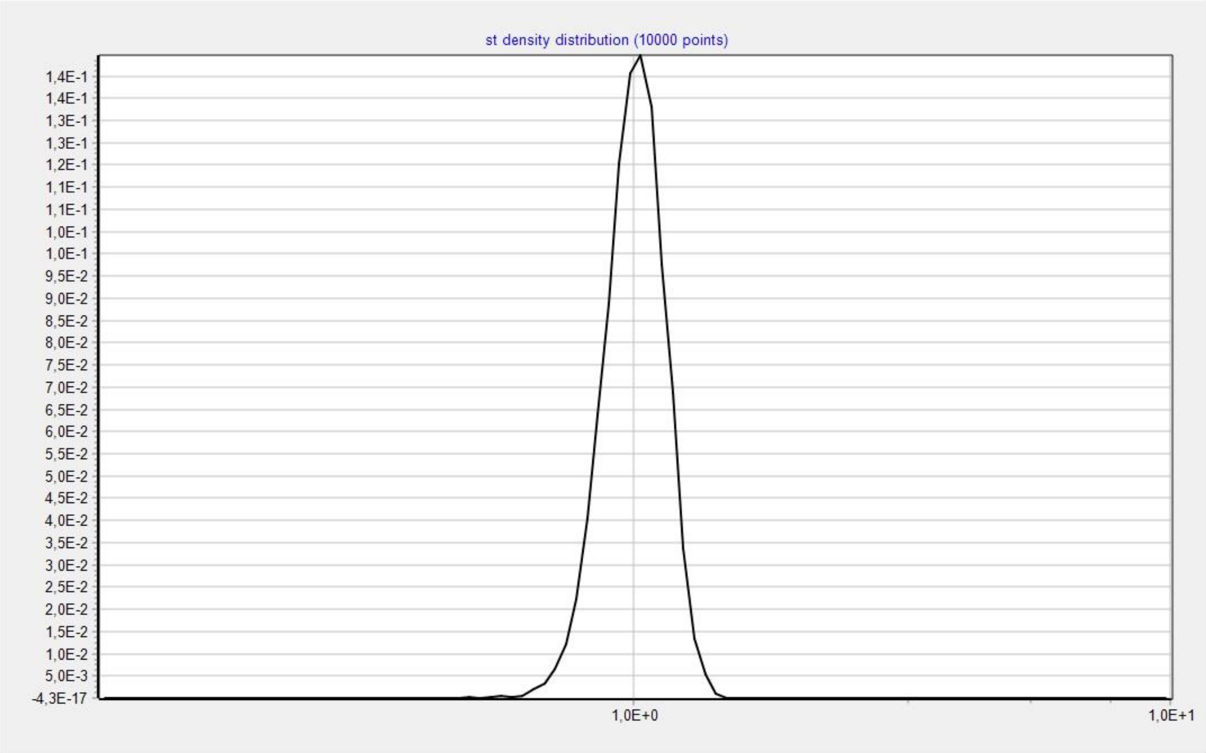

Figure 1: The uncertainty distribution of the scaling factor for Iwaki with no precipitation.

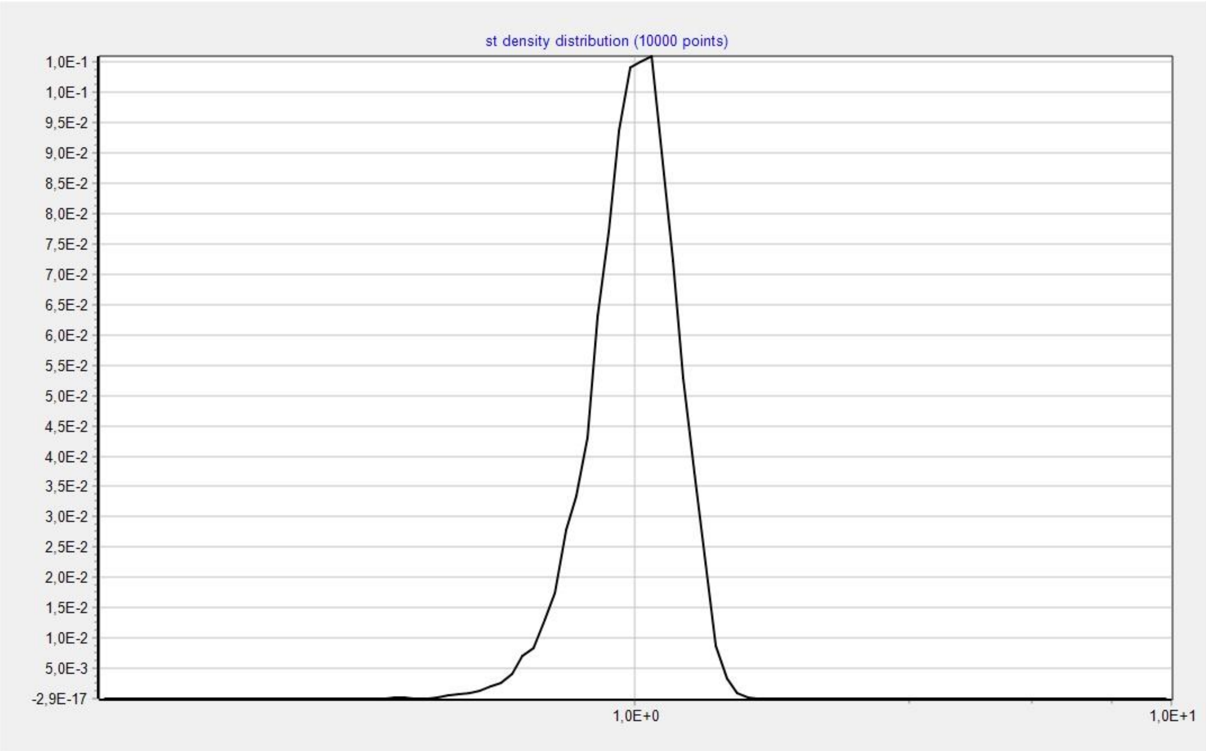

Figure 2: The uncertainty distribution of the scaling factor for Iwaki with precipitation.

4. Effects of weather dynamics

Improvements were made to the model concerning weather factors. Furthermore, computational experiments were conducted to determine what phenomena in weather dynamics have considerable effect on individual and population doses. This chapter describes these improvements and experiments, and provides some discussion on the effect of weather dynamics.

4.1 Accounting wind speed in the model

In the previous model (Karanta et al. 2015), wind speed was modelled very simply in the event tree: there were three branches, for 0 km/h, 8 km/h, and 16 km/h wind speeds. The probabilities of these three wind speed classes were estimated from a lognormal distribution.

The handling of wind speed was modified considerably. It was not handled as branches in the event tree. Instead, a wind speed was sampled from a distribution on each simulation round and the corresponding population doses were calculated as functions of the wind speed.

Population doses were calculated using various wind speeds, from 0.5 m/s to 40 m/s, in ARANO. The result was a vector of population doses for each city in both rain and no rain conditions. 40 m/s is the wind speed of a typhoon, and therefore no population doses for higher wind speeds were needed. Population dose curves as functions of wind speed are presented in Figures 3 and 4.

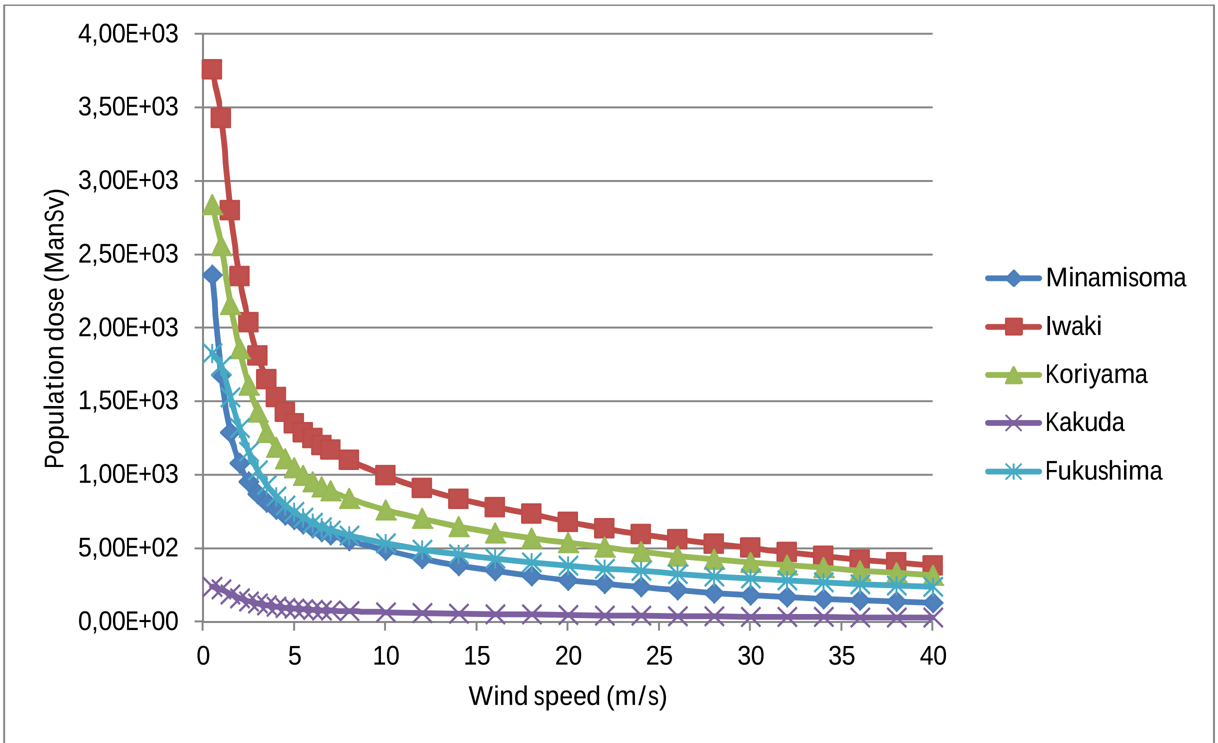

Figure 3: The population doses calculated in ARANO as functions of wind speed in the case of no rain.

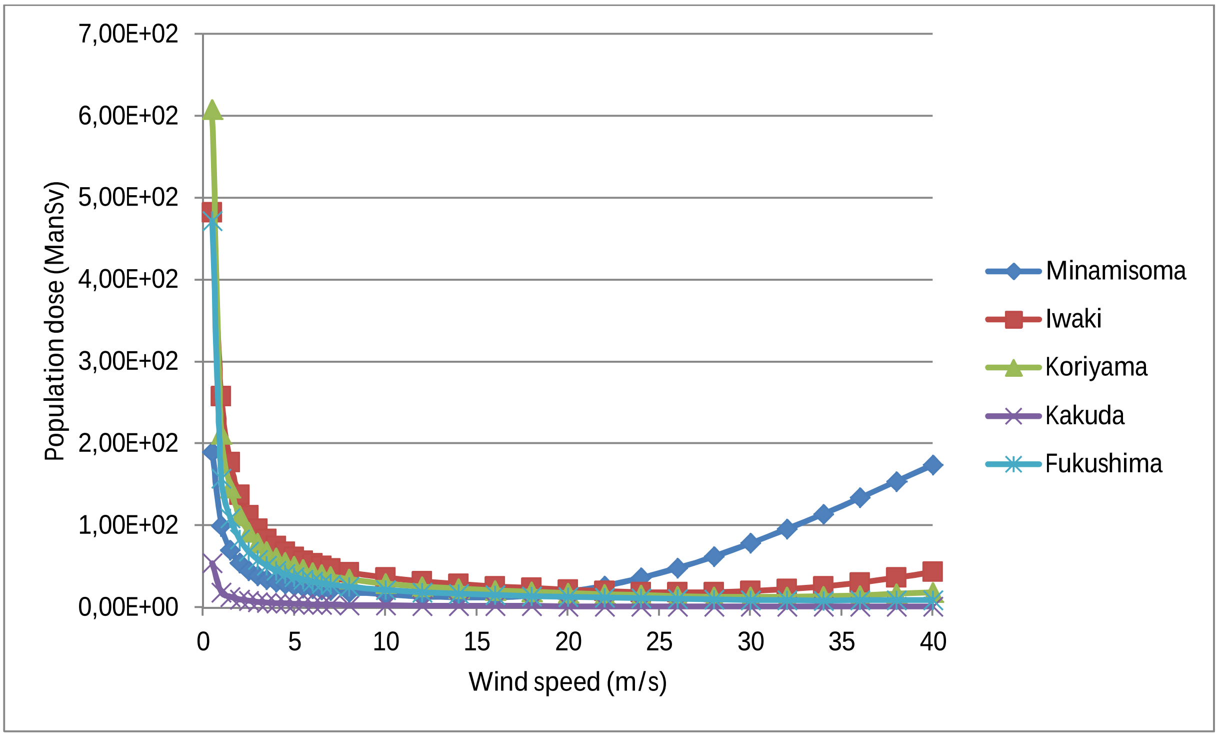

Figure 4: The population doses calculated in ARANO as functions of wind speed in the case of rain.

In Figure 4, dose is increased with increasing wind speed in the towns located at shorter distances from the release point. This is due to minor scavenging of the plume at high wind speeds: the rain does not have time to wash the aerosols down before the plume reaches the city and aerosols are washed down on it by the rain as ground deposit.

Wind speed was handled with Monte Carlo simulation. On each simulation round, the wind speed was drawn from a probability distribution, and the corresponding population doses were looked up from the wind speed / population dose vectors for each city in both rain and no rain conditions. The population doses were calculated from the elements of the vectors by linear interpolation. For wind speeds smaller than 0.5 m/s, the population doses of 0.5 m/s were used, and for wind speeds larger than 40 m/s, the population doses of 40 m/s were used. The simulation gave a probability distribution for population doses, from which e.g. mean population dose can be calculated.

The probability distribution used in simulation was the Weibull distribution. Although many probability distributions have been used for wind speeds (Carta et al. 2009), Weibull distribution remains the most popular, because daytime wind speed observations are generally consistent with it (night time wind speeds are positively skewed when compared with the Weibull distribution) (Monahan et al. 2011). The parameters of the Weibull distribution were estimated so that they fit the statistical information available from Onahama, a city in the Fukushima prefecture some 60 kilometers south of the Fukushima Daiichi site (the same weather statistics data was used in (Karanta et al. 2015)). The wind speed statistics are from the www site http://www.windfinder.com/windstatistics/onahama, and they are as follows: mean wind speed in March 8 knots, probability of wind speed exceeding or equalling 4 Beaufort 0.19. The wind speed distribution is presented in Figure 5.

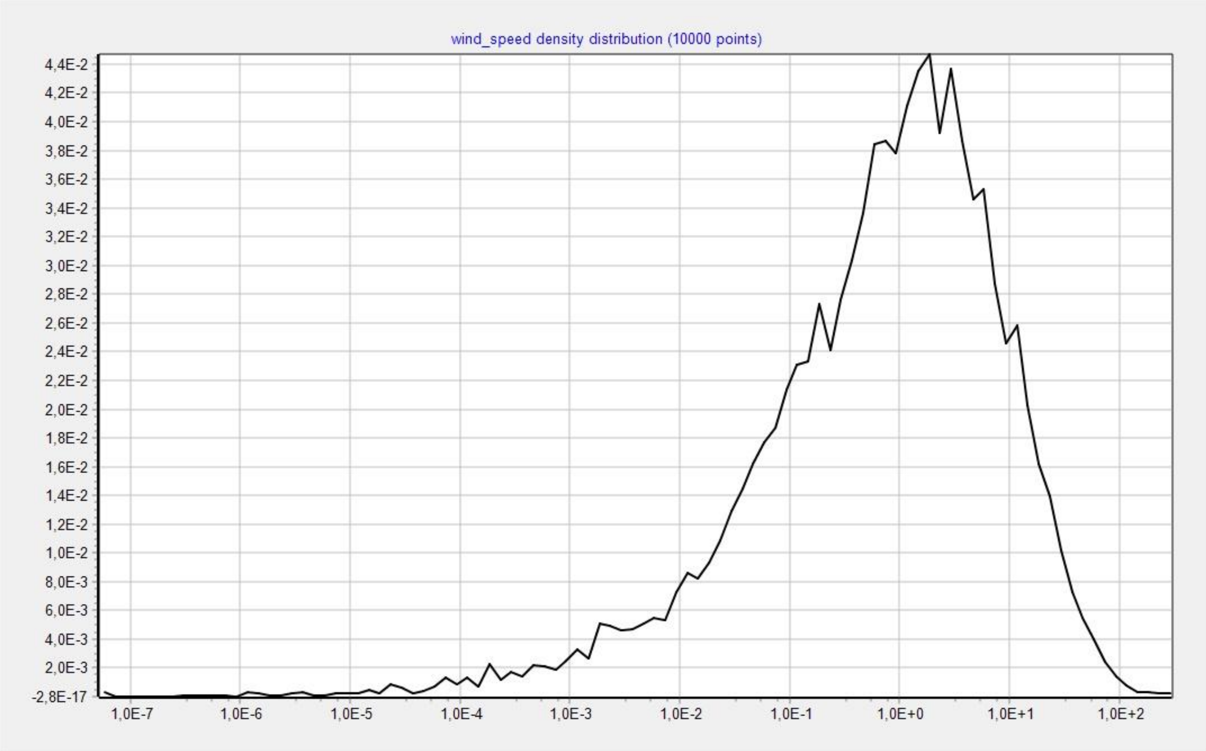

Figure 5: The distribution of wind speed (unit: m/s).

4.2 Effect of wind speed and direction changes

Wind speed change may either increase or decrease population dose when compared to the situation where wind speed is constant.

- The wind calms down while the release plume is over a population center. In this case, the population's dose is expected to increase when compared to the constant wind speed situation.
- The wind speeds up when the plume is over a population center. In this case, the (local) population's dose is expected to decrease when compared to the constant wind speed situation.
- The wind calms down before the release plume reaches the population. This prevents the population from being subjected to ionizing radiation from the release altogether.

Wind direction changes may also increase or decrease population dose:

- If the wind was initially blowing towards a sparsely populated or nonpopulated area (e.g. sea), and then turns toward a densely populated area, population dose will increase unless evacuation is carried out in time. This applies also to situations where the wind was initially blowing towards a densely populated area, and then turns towards another densely populated area (instead of continuing towards a sparsely populated area).
- If the wind was initially blowing towards a population center, and then turns towards a sparsely populated area, the dose will decrease. This applies both in situations where the radioactive plume had not yet reached the population center, and in situations where, without change in wind direction, the plume would have continued towards another population center.

Near a nuclear power plant, most directions do not contain a population center within short distance. This applies especially to NPP's located on a coast; in Fukushima Daiichi, for example, more than half of directions from the plant are either uninhabited (the Pacific ocean) or have relatively low population density close to the site. Furthermore, if a radioactive plume turns towards a population center after moving through a less populated area, it has moved longer than it would have if it had moved in one direction only, and thus has lost more of its radioisotopes on the way than a plume that has moved in one direction only. Thus, there is some justification in saying that having the plume move in one direction only (as in ARANO), one gets more conservative estimates for population doses.

Wind speed and direction changes could be incorporated in the model through combining wind speed and wind direction (direction is now handled as branches in the event tree) handling into a procedure that would calculate plume paths through Monte Carlo, and calculate doses if the path crosses a population center. This, however, would require weather data of the wind conditions near Fukushima Daiichi (from which wind speed statistics, including time correlations, could be estimated); such data was not available for this study. Furthermore, implementing this would require a major programming effort, which was beyond the resources of the project.

4.3 Effect of rain timing

Rainfall has quite a different effect on population dose depending on whether it occurs before the plume has reached a population center, or while the plume is above a population center. Before the plume reaches the general population, rainfall washes aerosols from the plume, and with the aerosols, the most harmful radioisotopes (Iodine and Cesium) get washed down to sparsely inhabited or uninhabited regions. While the plume is above a population center, rainfall washes aerosols to the ground there, and thus contributes to surface deposition.

Population dose as a function of start time of rain before the plume reaches the population center (Iwaki) is depicted in Figure 6. Rain intensity was assumed to be 5 mm/h, stability class C, and wind speed 4 m/s. As can be seen, even half an hour's rain reduces the population dose to less than 2 % of the original if the rain occurs before the population center.

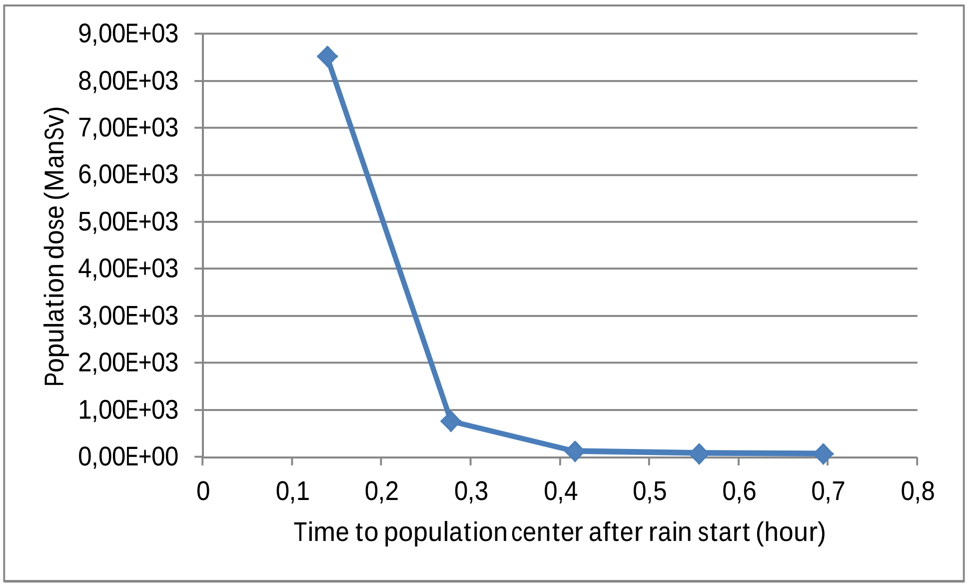

Figure 6: Population dose (manSv) as a function of the time (hours) it takes the plume to reach a population center after start of rain.

Various computational experiments were conducted. Of these, perhaps the most illuminating concerns the comparison of three situations:

- one in which there is no rain,
- one in which rain starts when the release plume is approaching a city but stops before the plume enters the city,
- one in which rain starts when the release plume is approaching a city and continues as the plume flows above the city.

The case of Iwaki was considered (48 km from Fukushima Daiichi, 345 000 inhabitants in 2011). The values of the weather variables are wind speed 4 m/s, rain intensity (when there is rain) 2 mm/h, stability class C, Table 3 summarizes the results.

Table 3. Effect of rain timing on individual and collective doses.

Rain timing	maximal individual dose (Sv)	collective population dose (manSv)
no rain	4,43E-03	1530
rain starts 4 km (about 17 min) before plume reaches city, and continues while the plume is above the city	1.67E-02	5770
rain starts 4 km (about 17 min) before plume reaches city, and stops 1 km before the plume reaches city	7,86E-04	271

The effect of rain timing is quite dramatic. If the rain starts and stops before the city, the doses are more than 20 times less the doses when the rain starts before the city but continues when the release plume is above the city. When compared to the case that there is no rain, the washing effect of rain before the city still reduces population dose to less than one-fifth.

The computational experiments confirm the intuitive idea that rainfall between the nuclear accident site and a population center is a blessing, but rainfall within the population center (after the release plume has arrived) is a curse.

4.4 Effect of the intensity of rain

Rain intensity affects deposition of volatile compounds in the release due to the washing effect: water droplets wash aerosol particles from the air, depositing the volatile compounds to where the rain falls. This washing effect does not affect noble gases, and so the plume still contains radioisotopes, but it is effectively stripped off the most harmful radioisotopes (Iodine and Cesium, in particular). In this section the effect of rain intensity is quantified.

Rainfall intensity is classified according to the rate of precipitation [American Meteorological Society 2015]:

- Light rain (precipitation rate is < 2.5 mm per hour)

- Moderate rain (precipitation rate is between 2.5 mm - 7.6 mm per hour)
- Heavy rain (precipitation rate is > 7.6 mm per hour)

The Met Office of United Kingdom [Met Office 2007] uses the scale slight 0 – 2 mm/h, moderate 2 – 10 mm/h, heavy 10 – 50 mm/h, and violent > 50 mm/h.

A complex relationship exists between rainfall intensity, rain duration and frequency (Koutsoyiannis et al. 1998), and this relationship probably has great effect on the probability that rain washes the aerosols of a release plume before it reaches a given geographical location. However, here only rainfall whose intensity is static in time is considered.

The first computational experiment concerned the effect of rainfall on the release plume moving from Fukushima Daiichi to Iwaki (48 km south of Fukushima Daiichi, 345 000 inhabitants). Its results are shown in Figure 7. As can be seen, rainfall more intensive than approximately 4 mm/h (moderate rain) will wash all aerosols from the plume. The only radionuclides that remain in the plume are noble gases, and they cause a rather small population dose (approximately 21.3 manSv).

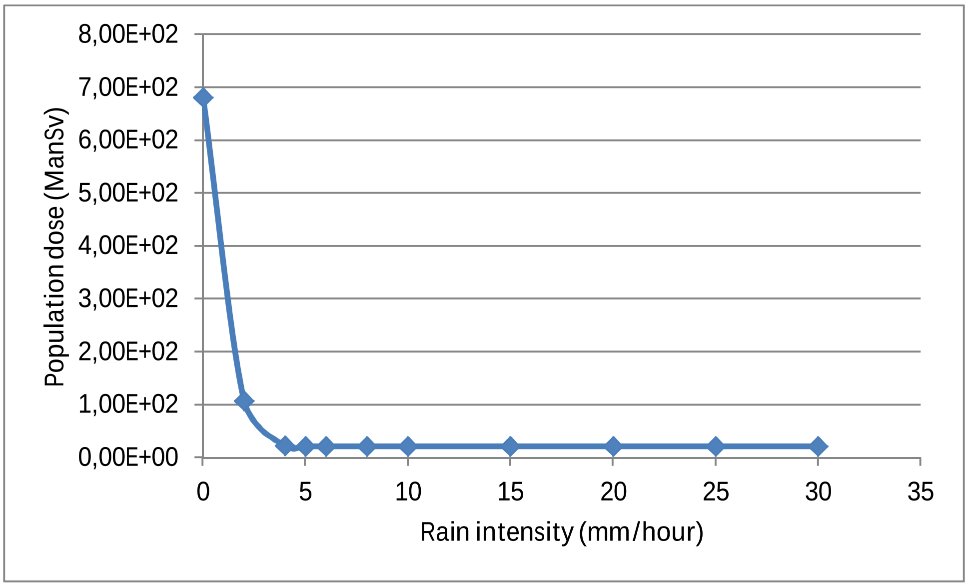

Figure 7. Population dose in Iwaki (48 km from Fukushima Daiichi) as a function of rainfall intensity. Wind speed 20 m/s, rain all the way from Fukushima Daiichi to Iwaki.

5. Dynamic models of evacuation

Evacuation is perhaps the most important early countermeasure, because if the population is transported away from the area before the plume arrival, radiation exposure is completely avoided.

The importance of evacuation has received attention in the nuclear safety community from early on (NUREG 1980). In practice, a nuclear power plant site is surrounded by a low population density zone. For the radiation protection purposes the NPP is surrounded by an emergency planning zone (EPZ), which spans up to 16-25 kilometers from the site, depending on the country. In Finland the EPZ is divided into two zones: the protection zone extends to about five kilometres from the power plant and the emergency planning zone is applied for an area within a radius of about 20 km. The planning principle of these zones is that there shall not be need for evacuation beyond the protective zone due to a severe reactor accident and no need for sheltering beyond the preparedness zone. As the

Fukushima accident demonstrated, there may be need for evacuation beyond the EPZ to reduce the collective dose.

Evacuation is a complex phenomenon, as there are many factors to be taken into account:

- Trip generation time is the time from the issuance of evacuation recommendation to the beginning of household's departure from the EPZ (Urbanik 2000). It consists of notification time, or the time from the decision to evacuate to getting the message through to people in the area to be evacuated, and mobilization time, or the time span between receiving notification and departure from home (Tweedie et al. 1986).

- The road network of the EPZ affects evacuation success in many ways. The road network near the Finnish NPP's is not very complex, but if evacuation would have to be extended to nearby cities, the models of the road network would become quite large. The location of workplaces and residences in the network is of importance, because people normally drive to home from work before they evacuate; if the workplaces are located so that this causes a lot of traffic crossing the radial evacuation traffic, this may cause delays. Also people who work outside the EPZ but live within it will return to their homes before they evacuate, which affects their evacuation time. Road capacity is of importance, too: big roads with more than two lanes will have larger capacity for the evacuation traffic, and traffic control actions affect evacuation times, too. If the rate of evacuation trip departures exceeds road capacity, traffic slows down and the time required by the excess trip demand has to be added to the evacuation times.

- The spatial distribution of population varies by time in ways that affect evacuation times. Holiday seasons have considerable effect (e.g. many inhabitants in the EPZ of the Loviisa power plant live there only in the summer holiday season). Time of day affects the distribution: in the daytime, people who work in or near the plant are at their workplaces, while they are at home in the night time; the reverse is true of people who live in the EPZ but work elsewhere.

- The number of available vehicles in relation to population size affects evacuation time. If there is sufficient vehicle capacity, each vehicle has to make only one trip; otherwise, some vehicles have to return and fetch more people. In addition to cars of the people living or working in the EPZ, also public transit has to be taken into account.

- Evacuation of public buildings such as schools and hospitals have to be considered by a model that concerns both building evacuation and evacuation trips through the road network according to the evacuation plans for those buildings; the model should take into account the fact that the population in such buildings have limited mobility.

- Evacuation management and control – e.g. the control of traffic by police, and evacuation instructions given by officials concerning e.g. evacuation timing and routing – affect evacuation times considerably as they e.g. prevent traffic congestion.

- Weather factors and time of day affect traffic. For example, rain, snowstorms, and heavy wind may cause traffic to slow down. If the nuclear accident being considered in the level 3 analysis has been caused by e.g. a flood or an earthquake, also the road network may have been damaged.

Some codes for evacuation modelling and analysis, such as I-DYNEV and OREMS, exist. Furthermore, the former is public domain, or at least was at the time of writing of (Urbanik 2000). However, I-DYNEV could not be found at the site of the U.S. Federal Emergency Management Agency (FEMA) for the purpose of this study, or even a contact address where it could be obtained. Even if the codes were available, proper modelling and analysis of evacuation would be a major undertaking due to the complicating factors listed above.

Statistical models that could have been used to provide evacuation time estimates for the level 3 model were not to be found for this study.

The way that evacuation is taken into account in the model was improved in the following way. Hitherto, if evacuation failed, the whole population was assumed to be in the city and be subjected to radiation for three days. This is overly conservative, because it is most probable that a part of the population has been evacuated when the plume arrives, even if the evacuation of all people has failed. Now, if evacuation fails, the population is assumed to be subjected to radiation for only the time from the plume arrival to three days. This is a conservative assumption, and still represents an upper limit to total population dose. In practice, this was implemented so that the total population doses, in the case that evacuation had failed, were scaled down with $(T_e – T) / T_e$, where T_e is the evacuation time distributed normally with mean 72 hours and standard deviation 7.2 hours and T is the time that the plume arrived in the city (in practice, T=S/v, where S is the city's distance from Fukushima Daiichi, and v is the wind speed). This is still an approximation, because the dose rates from cloudshine and inhalation are zero after the plume has moved past the person (population) considered.

6. Modelling of sheltering

Sheltering is another short-term countermeasure. Although not as effective as evacuation, it may significantly reduce the population dose.

There are several factors that affect the effectiveness of sheltering:

- Timing of sheltering recommendation.
- Time span to get the sheltering recommendation through to the general population. Several factors bring uncertainty to this: the communication channels available/used, the spatial distribution of the people.
- The proportion of people who choose not to obey the sheltering recommendation but e.g. decide to leave the EPZ.
- The time it takes each individual to arrive at a shelter, starting from where they were at the time they received sheltering recommendation.
- The quality of the shelters (ventilation, permeability of walls by ionizing radiation etc.).
- The time people leave the shelters (e.g. after the sheltering recommendation has been cancelled).

The authors of this report do not know of mathematical models of sheltering in scientific literature. Such models would contain many parameters for which statistical data is not available (e.g. the proportion of people who ignore the sheltering recommendation), and which should therefore be estimated by expert judgment. Such expert judgment exercises have been conducted e.g. by the European Union (Goossens et al. 2001). Large uncertainties would thus be associated with such models.

7. Results

The results from the simulation runs of the improved model (see Section 2) were as follows. The expected number of cancer deaths was 3.6. Direct comparison to the results in (Karanta et al. 2015) cannot be made because there was an input error in the calculations in the previous model. However, the model from (Karanta et al. 2015) was recalculated with correct input, and the expected number of cancer deaths was 1.2. The difference can be explained by the differences in wind speed modelling. Previously, the doses were assumed to be zero if the wind speed was smaller than 4 m/s, while speeds from 0.5 to 4 m/s can actually lead to high doses. Also, the probability of wind speed smaller than 4 m/s is large as can be seen from Figure 5.

The mean value of the probability of more than 0.1 cancer deaths was 0.16. The maximum of this probability was 0.22. Again, these higher numbers compared to (Karanta et al. 2015) can be explained by the wind speed modelling.

Figure 8 presents the complementary cumulative distribution of the number of cancer deaths. The probability for 20 cancer deaths is around 0.1. The probability for 60 cancer deaths is around 0.01, while the probability for 100 cancer deaths is less than 0.001.

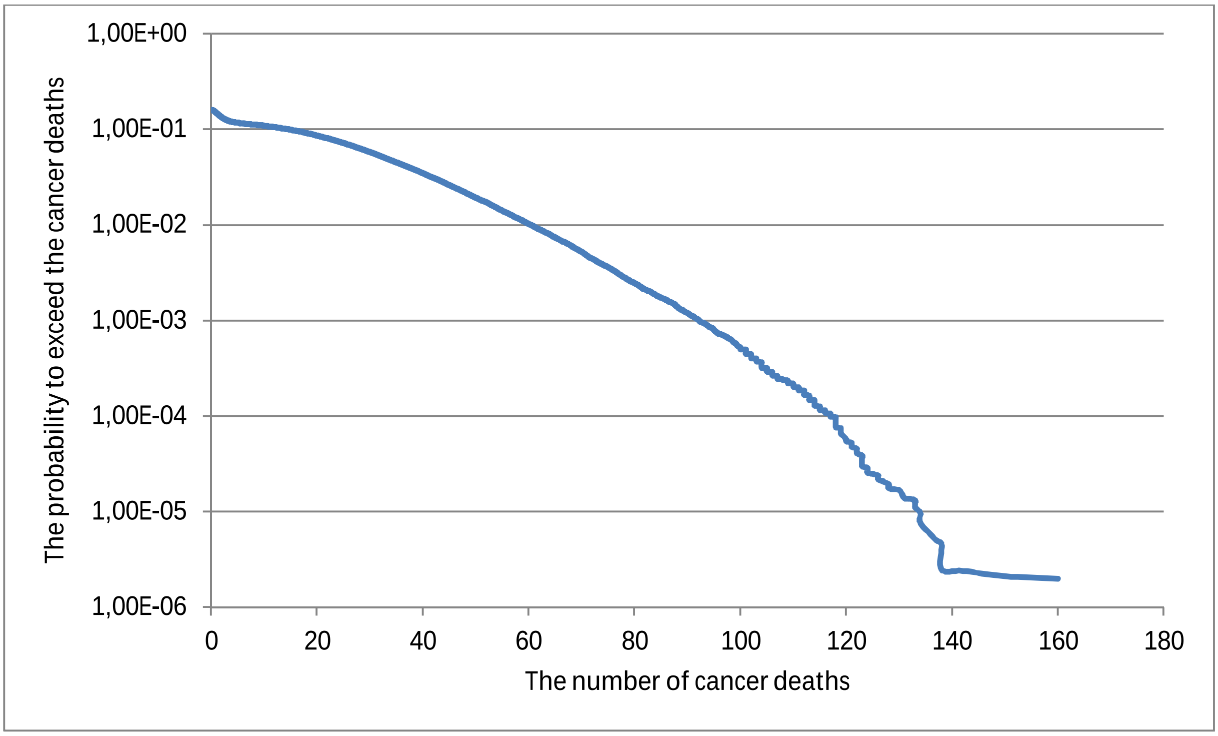

Figure 8: Complementary cumulative distribution of the number of cancer deaths.

Uncertainties on cancer deaths can also be viewed based on the scatter plot between the number of cancer deaths and the probability ('Freq') presented in Figure 9. This scatter plot contains a point (if not 0) from each sequence from each simulation round. Notice that the probability of anyone dying of cancer cannot be judged based on this graph because each point represents only one event. On one simulation round, the probability of cancer deaths is the sum of the probabilities of the sequences with non-zero population dose consequences. The two high density areas in the graph represent cases of rain and no rain; note that this bimodality is most probably an artefact of the model (with just two values for rain intensity), and would likely disappear if a continuous model for rain intensity would be introduced. The cancer death numbers are much lower with rain. The highest cancer death numbers are from southwest south direction where the largest and second nearest city is located, but around 100 cancer deaths can come also from directions west and north.

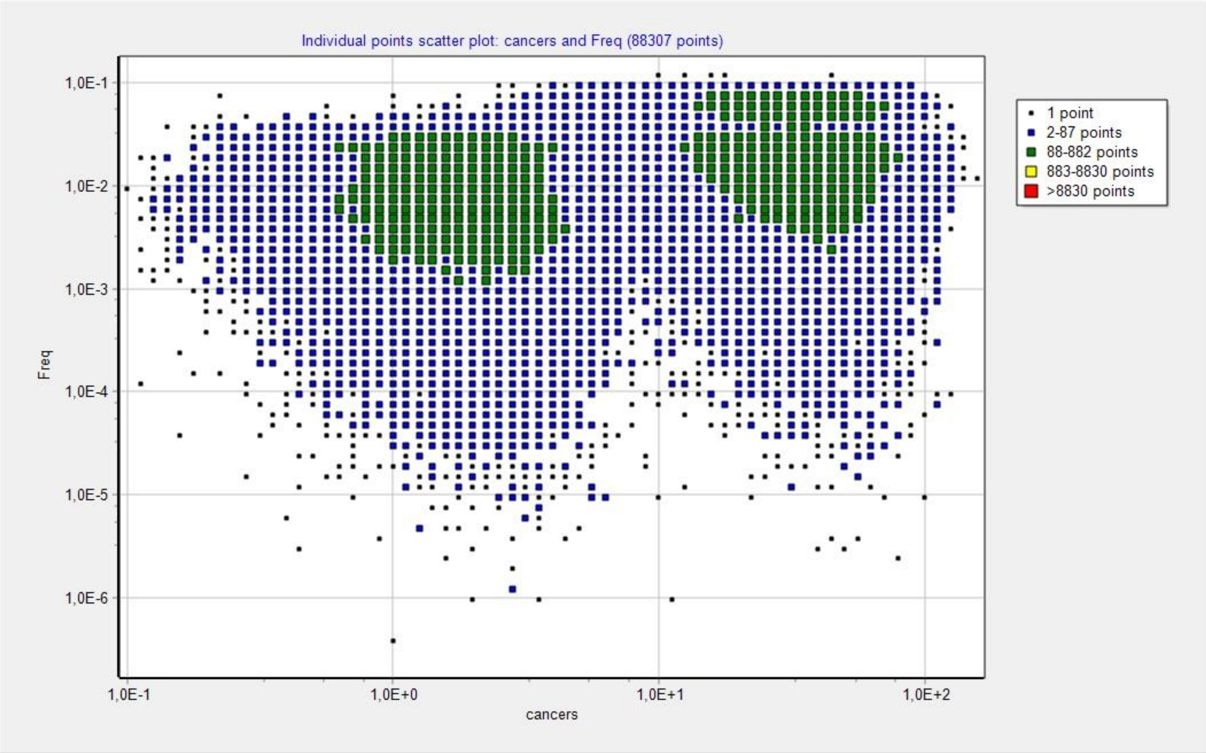

Figure 9: The scatter plot between the number of cancer deaths and the probability.

The effect of the wind speed on the number of cancers can be judged based on the scatter plot presented in Figure 10. This plot contains one weighted point from each simulation round (except from the rounds with very small population doses), and the cancer death numbers are therefore smaller than the largest values in Figure 9. The largest cancer death numbers are obtained with wind speeds from 1 to 10 m/s and especially from 1 to 2.5 m/s.

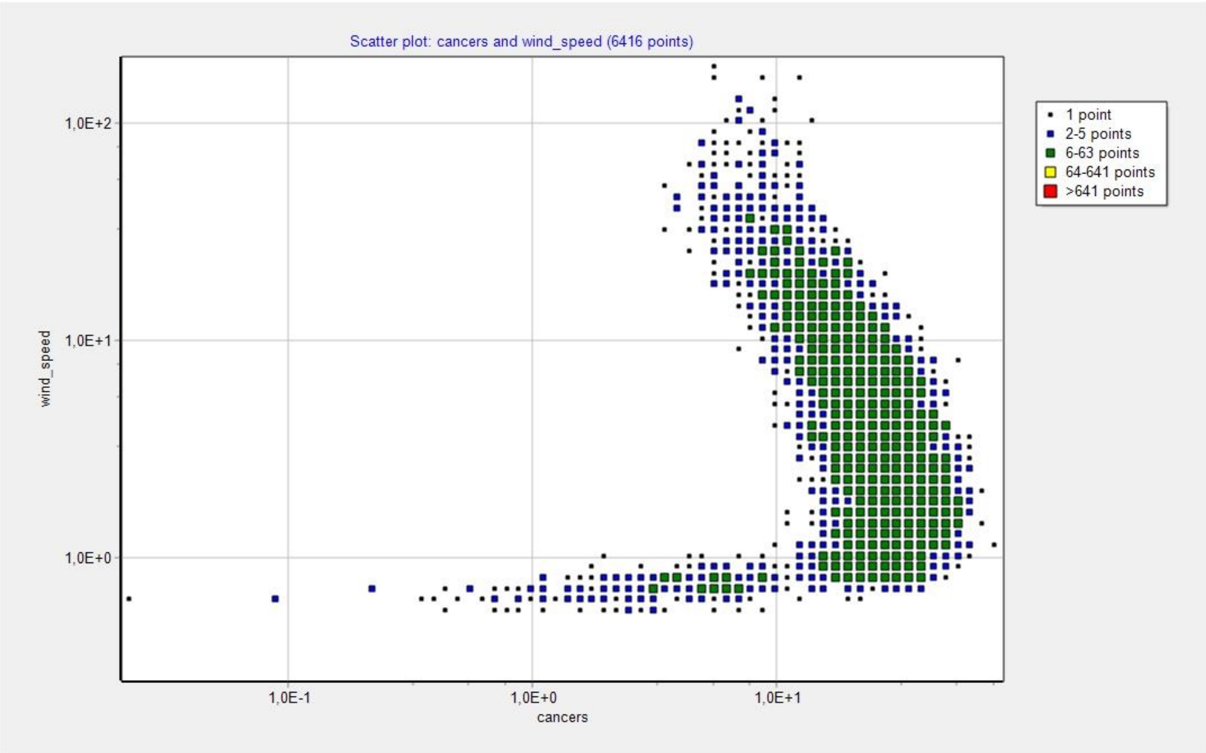

Figure 10: Scatter plot between the number of cancers and the wind speed.

8. Conclusions

The level 3 PSA model was improved compared to the one previously developed. The main improvement in the model is that wind speed is not handled coarsely in wind speed classes, but rather as a continuous random variable following the Weibull distribution; also the evacuation model has been improved. Source term uncertainty has been included and handling of wind speed and evacuation uncertainties has been improved in the model. Computational experiments on the effects of various weather factors on population dose have also been conducted.

The central factors affecting population dose are wind direction, wind speed, precipitation and evacuation. If the wind blows to the right direction (in the Fukushima case, towards the Pacific ocean) or if evacuation is conducted in time, the general population receives essentially no dose of ionizing radiation. High-speed wind takes the radioactive plume quickly away from a population center, with relatively little ground and surface deposition; low-speed wind does the opposite, and thus, increases the population dose. Rain washes aerosols from the radioactive plume, and leaves essentially only noble gases if long-lasting and/or intensive enough; if the rain starts well before a population center, this leads to lesser population dose, but if it starts above the population center, this leads to higher population dose due to increased ground deposition.

Considering the topic of the case study, the results support the conclusion indicated by results in (Karanta et al. 2015): the minor radiological consequences to the general population in the Fukushima Daiichi nuclear accident were not a matter of good luck, but rather what one would expect, given Fukushima Daiichi's location, weather in that part of Japan in March, and the effectiveness of evacuation in the region.

There are several important research issues that would need further work. The dynamics of precipitation (the relationship between rain intensity, duration and frequency), and its incorporation into level 3 PSA analyses, would merit a more thorough treatment. Proper evacuation modelling, with traffic modelling software, and the analysis of the model, would give more justified estimates of evacuation times and also uncertainty distributions for them. Modelling of sheltering would shed light on the actual effectiveness of sheltering, and would also increase the plausibility of the level 3 PSA model.

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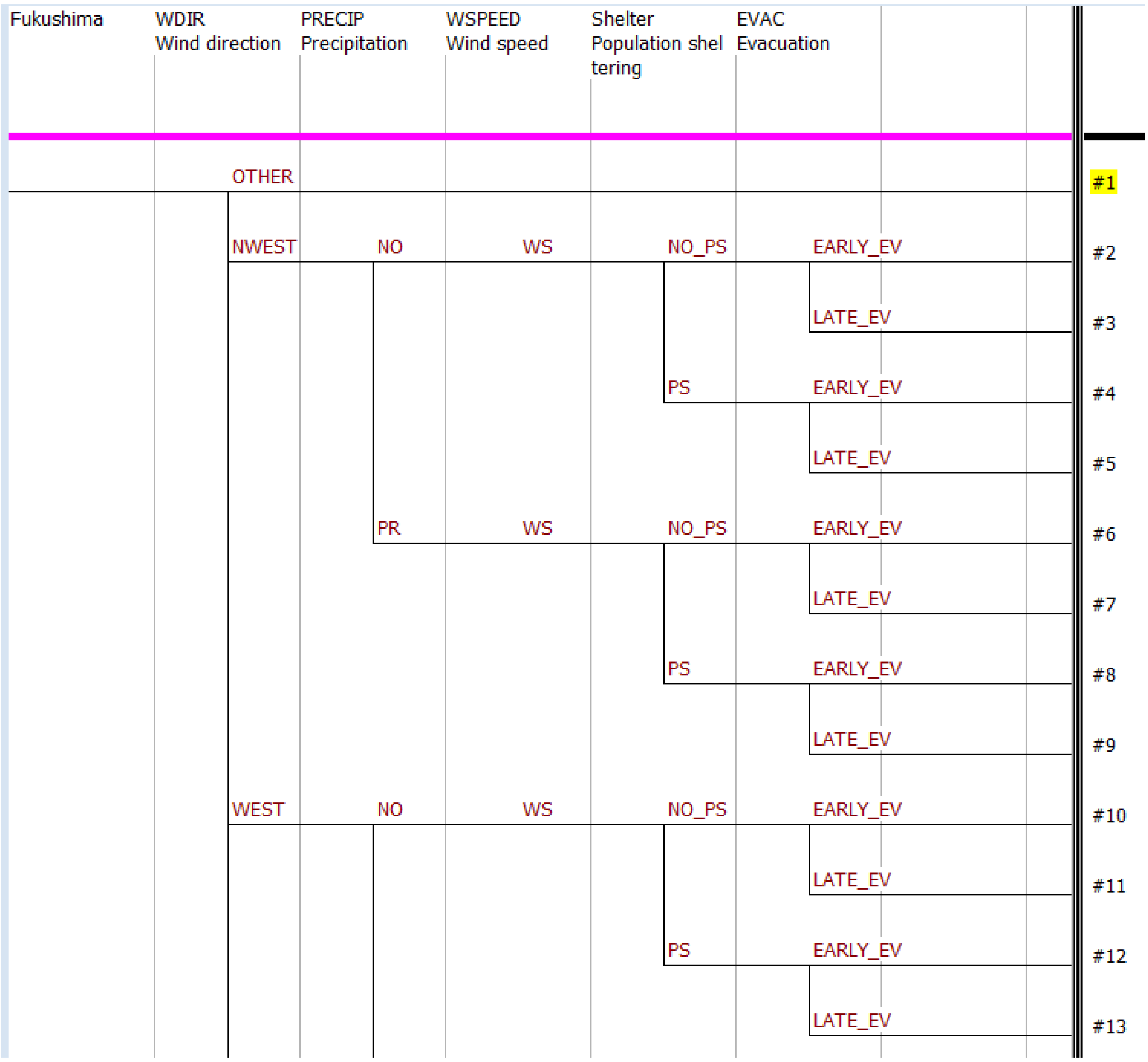
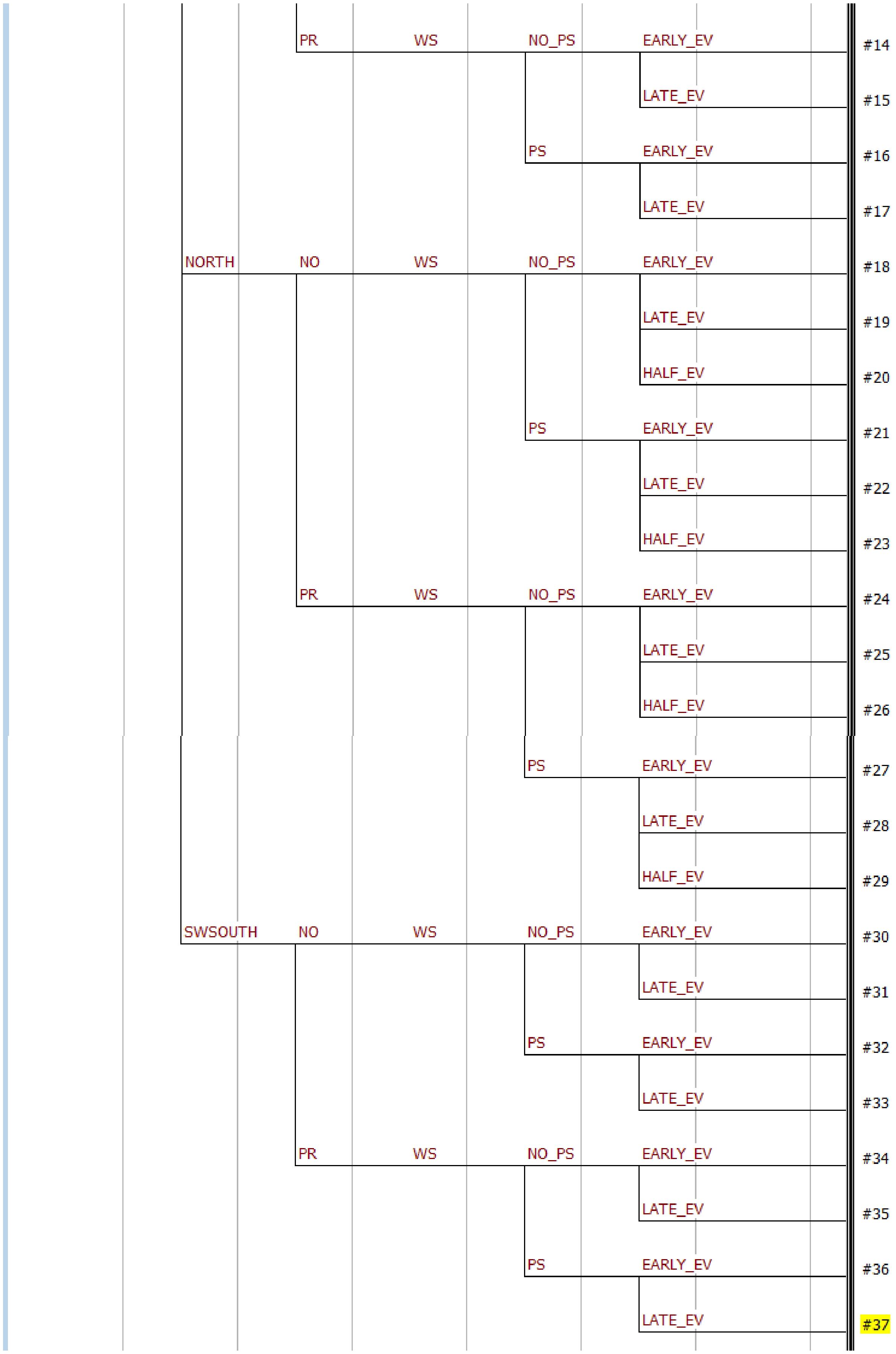
Appendix A: the event tree model

The event tree model is presented in the following. Some function names are explained in Table 4.

Table 4. Descriptions of functions in the event tree model.

Function	Desciption
NO	No rain
PR	Precipitation 5 mm/hour
WS	Wind speed computation
NO_PS	No population sheltering
PS	Successful population sheltering
EARLY_EV	Evacuation completely successful
LATE_EV	Evacuation not completely successfully
HALF_EV	In the case wind direction north, Kakuda is evacuated but Minamisoma is not.




PR
WS
NO_PS
EARLY_EV
#14
LATE_EV
#15
PS
EARLY_EV
#16
LATE_EV
#17
NORTH
NO
WS
NO_PS
EARLY_EV
#18
LATE_EV
#19
HALF_EV
#20
PS
EARLY_EV
#21
LATE_EV
#22
HALF_EV
#23
PR
WS
NO_PS
EARLY_EV
#24
LATE_EV
#25
HALF_EV
#26
PS
EARLY_EV
#27
LATE_EV
#28
HALF_EV
#29
SWSOUTH
NO
WS
NO_PS
EARLY_EV
#30
LATE_EV
#31
PS
EARLY_EV
#32
LATE_EV
#33
PR
WS
NO_PS
EARLY_EV
#34
LATE_EV
#35
PS
EARLY_EV
#36
LATE_EV
#37

The initial section:

```
real cancers, $ the number of cancer deaths
     wind_speed,
     dist1, $ distance from the plant
     dist2,
     time1, $ time when the plume arrives
     time2,
     shfactor, $ sheltering factor
     pdose, $ baseline population dose based on wind
     pdose2,
     st, $ factor of source term uncertainty
     st2,
     cancer_factor $ number of cancers multiplied from pdose

real cor, evac_factor, evac_factor2
real te, i, xe, cs134, cs136, cs137

boolean rain

string dir

source cancers

collect wind_speed, st

routine init
  BinFreq = 1

  $ the number of cancer death multiplied from the population
dose
  cancer_factor = raneven(0.03, 0.07)

  $ source term uncertainty computation
  cor = rannorm(0.5, 0.0875) $ correlation factor
  i = rannorm(0.525, 0.0919)
  te = (1-cor)*rannorm(0.11, 0.0193)+i*0.11/0.525*cor
  xe = rannorm(0.06, 0.0105)
  cs134 = (1-cor)*rannorm(0.115, 0.0201)+i*0.115/0.525*cor
  cs136 = (1-cor)*rannorm(0.01, 0.00175)+i*0.01/0.525*cor
  cs137 = (1-cor)*rannorm(0.18, 0.0315)+i*0.18/0.525*cor
  st = te+i+xe+cs134+cs136+cs137 $ factor of source term
uncertainty
  st2 = te+i+xe+cs134+cs136+cs137
return

routine finish
  if rain then
  begin
    $ when it rains, the whole population dose comes from
xenon
    st = xe/0.06
    st2 = xe/0.06
```

```
  end
  else
  begin
    $ factor of source term uncertainty changed according to
wind speed
    $ the fraction of xenon depends on wind speed
    $ st2 is for Minamisoma and st for other cities
    if wind_speed > 16 then
    begin
      st = (st-xe)*103/100+xe*0.5
      st2 = (st2-xe)*103/100+xe*0.5
    end
    else if wind_speed > 8 then
    begin
      st = (st-xe)*104/100+xe*4/6
      st2 = (st2-xe)*103/100+xe*0.5
    end
    else if wind_speed > 4 then
    begin
      st = (st-xe)*105/100+xe*5/6
      st2 = (st2-xe)*104/100+xe*4/6
    end
    else if wind_speed > 1 then
    begin
      st = (st-xe)*98/100+xe*8/6
    end
    else
    begin
      if samestr(dir, 'NWest') then
      begin
        st = (st-xe)*80/100+xe*26/6
      end
      else if samestr(dir, 'SWSouth') then
      begin
        st = (st-xe)*93/100+xe*13/6
      end
      else
      begin
        st = (st-xe)*84/100+xe*22/6
      end
      st2 = (st2-xe)*98/100+xe*8/6
    end
  end

  $ The number of cancer deaths calculated.
  $ Baseline population dose is scaled according to
  $ source term uncertainty, evacuation factor, sheltering
factor.
  $ Population dose is multiplied by the cancer factor.
  $ When wind direction is north, there are two cities with
different
```

```
  $ population doses, evacuation factors and source term
uncertainty factors.
  if samestr(dir, 'North') then
  begin
    cancers =
(pdose2*evac_factor2*st+pdose*evac_factor*st2)*shfactor*cancer
_factor
  end
  else
  begin
    cancers = pdose*shfactor*st*evac_factor*cancer_factor
  end

  if cancers < 0.1 then dir = 'Other'
return

class dir
routine binner active
  ('NWest',    'Expo'),
  ('West',     'Expo'),
  ('North',    'Expo'),
  ('SWSouth',  'Expo'),
  ('Other',    'Other')
return
```

WDIR section

```
real nw, w, n, sws

routine init
  nw = raneven(0.018, 0.058)
  w = raneven(0.007, 0.047)
  n = raneven(0.094, 0.134)
  sws = raneven(0.077, 0.117)
return

$ Fukushima city is located in northwest
function real NWEST
  dist1 = 64
  dist2 = 0
  dir = 'NWest'
return nw

$ Koriyama is located in west
function real WEST
  dist1 = 56
  dist2 = 0
  dir = 'West'
return w

$ Minamisoma and Kakuda are located in north
function real NORTH
  dist1 = 27
```

```
  dist2 = 58 $ dist2 for Kakuda
  dir = 'North'
return n

$ Iwaki is located in southwest south
function real SWSOUTH
  dist1 = 48
  dist2 = 0
  dir = 'SWSouth'
return sws

$ No cities in other directions
function nil OTHER
  dist1 = 0
  dist2 = 0
  dir = 'Other'
  pdose = 0
  pdose2 = 0
return nil
```

PRECIP section

```
real hp

routine init
  hp = raneven(0.158, 0.358)
return

function real PR
  rain = true
return hp

function nil NO
  rain = false
return nil
```

WSPEED section:

```
real u, a, b, w

integer j

$ wind speeds and corresponding baseline population dose for
$ each city in both rain and no rain conditions.
vector(31) speeds = (0.5, 1, 1.5, 2, 2.5, 3, 3.5, 4, 4.5, 5,
5.5, 6, 6.5, 7, 8, 10, 12, 14, 16, 18, 20, 22, 24, 26, 28, 30,
32, 34, 36, 38, 40),
           doseMn = (2360, 1680, 1290, 1080, 954, 870, 814,
768, 729, 696, 667, 640, 614, 591, 551, 489, 430, 384, 348,
313, 284, 259, 238, 216, 197, 182, 168, 156, 146, 137, 129),
           doseIn = (3760, 3430, 2800, 2350, 2040, 1810, 1650,
1530, 1430, 1350, 1290, 1250, 1200, 1170, 1100, 996, 908, 838,
```

```
781, 734, 681, 635, 596, 562, 532, 503, 473, 447, 424, 403,
384),
           doseKOn = (2840, 2560, 2160, 1860, 1610, 1430,
1290, 1190, 1110, 1050, 995, 952, 920, 891, 841, 762, 703,
649, 606, 569, 538, 508, 477, 450, 386, 369, 350, 333, 317),
           doseKAn = (241, 220, 188, 162, 140, 125, 113, 104,
97, 91.4, 86.6, 82.5, 79.7, 77.2, 72.8, 66.1, 60.9, 56.4,
52.6, 49.5, 46.8, 44.5, 41.7, 39.4, 37.3, 35.4, 33.8, 32.3,
30.9, 29.4, 28),
           doseFn = (1830, 1740, 1530, 1320, 1160, 1030, 932,
855, 794, 748, 709, 675, 645, 624, 589, 534, 493, 460, 429,
404, 382, 363, 347, 328, 310, 295, 282, 269, 258, 248, 238),
           doseMp = (190, 100, 70.3, 54.6, 45, 38.7, 34.4,
31.1, 28.5, 26.3, 24.5, 23, 21.6, 20.5, 18.6, 15.8, 13.3,
12.1, 12.4, 14.7, 19.2, 26.3, 36, 48.1, 62.2, 78.3, 95.7, 114,
134, 154, 174),
           doseIp = (482, 258, 177, 137, 112, 95.8, 83.7,
74.5, 67.3, 61.5, 57.1, 53.5, 50.3, 47.6, 43, 36.4, 31.8,
28.4, 25.8, 23.7, 21.4, 19.7, 18.6, 18.2, 18.6, 19.8, 22.2,
25.6, 30.2, 36.1, 43.1),
           doseKOp = (608, 212, 146, 112, 92, 78.4, 68.5,
60.9, 55, 50.2, 46.2, 43, 40.5, 38.2, 34.6, 29.2, 25.5, 22.7,
20.6, 18.9, 17.5, 16.2, 14.9, 13.9, 13.3, 13, 13.1, 13.7,
14.8, 16.5, 18.7),
           doseKAp = (54.1, 18.7, 12.9, 9.85, 8.1, 6.9, 6.03,
5.36, 4.84, 4.41, 4.06, 3.77, 3.55, 3.35, 3.03, 2.56, 2.23,
1.99, 1.8, 1.66, 1.53, 1.43, 1.32, 1.22, 1.16, 1.12, 1.11,
1.14, 1.2, 1.31, 1.46),
           doseFp = (472,, 158, 109, 83.5, 68.4, 58.3, 50.9,
45.3, 40.8, 37.2, 34.3, 31.8, 29.7, 28, 25.3, 21.3, 18.6,
16.6, 15, 13.8, 12.8, 11.9, 11.2, 10.4, 9.68, 9.15, 8.78, 8.6,
8.63, 8.88, 9.38)

routine init
  $ wind speed is sampled from Weibull distribution
  a = raneven(0.4365, 0.5335)
  b = raneven(1.7487, 2.1373)
  u = random()
  wind_speed = b*pow(-ln(1-u),1/a) $ a = 0.485, b = 1.943
return

$ population dose is calculated based on wind speed
function nil WS
  $ finding right place (index j) in the vector
  j = 1
  while (wind_speed > speeds(j)) and (j < 31) do
  begin
    j = j+1
  end

  $ w is a weight for interpolation between two values
  if same(j, 1) then
```

```
  begin
    $ pdose is the dose of wind speed 0.5
    w = 0
    j = 2
  end
  else
  begin
    w = (wind_speed-speeds(j-1))/(speeds(j)-speeds(j-1))
  end
  if w > 1 then w = 1 $ pdose is the dose of wind speed 40

  $ population dose is calculated by interpolation
  if rain then
  begin
    if samestr(dir, 'NWest') then
    begin
      pdose = (1-w)*doseFp(j-1)+w*doseFp(j)
    end
    else if samestr(dir, 'West') then
    begin
      pdose = (1-w)*doseKOp(j-1)+w*doseKOp(j)
    end
    else if samestr(dir, 'North') then
    begin
      pdose = (1-w)*doseMp(j-1)+w*doseMp(j)
      pdose2 = (1-w)*doseKAp(j-1)+w*doseKAp(j) $ pdose2 for
Kakuda
    end
    else if samestr(dir, 'SWSouth') then
    begin
      pdose = (1-w)*doseIp(j-1)+w*doseIp(j)
    end
  end
  else
  begin
    if samestr(dir, 'NWest') then
    begin
      pdose = (1-w)*doseFn(j-1)+w*doseFn(j)
    end
    else if samestr(dir, 'West') then
    begin
      pdose = (1-w)*doseKOn(j-1)+w*doseKOn(j)
    end
    else if samestr(dir, 'North') then
    begin
      pdose = (1-w)*doseMn(j-1)+w*doseMn(j)
      pdose2 = (1-w)*doseKAn(j-1)+w*doseKAn(j) $ pdose2 for
Kakuda
    end
    else if samestr(dir, 'SWSouth') then
    begin
      pdose = (1-w)*doseIn(j-1)+w*doseIn(j)
```

```
    end
  end
return nil
```

SHELTER section

```
real sp, sf

routine init
  sp = raneven(0.6, 1) $ sheltering probability
  sf = raneven(0.5, 0.9) $ sheltering factor
return

function real PS
  time1 = dist1/wind_speed
  time2 = dist2/wind_speed $ time2 for Kakuda
  shfactor = sf
return sp

function nil NO_PS
  time1 = dist1/wind_speed
  time2 = dist2/wind_speed $ time2 for Kakuda
  shfactor = 1
return nil
```

EVAC section

```
real l1, l2, evtime

routine init
  evtime = rannorm(72, 7.2) $ evacuation completed at this
time
return
$ to how large portion of the dose the population is exposed
$ evac_factor2 is for Kakuda, evac_factor for others
function nil LATE_EV
  evac_factor = (evtime-time1)/evtime
  evac_factor2 = (evtime-time2)/evtime
  if evac_factor < 0 then evac_factor = 0
  if evac_factor2 < 0 then evac_factor2 = 0
return nil

$ if the plume comes after evacuation time, population dose is
0
function real EARLY_EV
  l1 = time1/evtime
  if l1 > 1 then l1 = 1 else l1 = 0
  pdose = 0
  pdose2 = 0
  dir = 'Other'
return l1

$ Kakuda evacuated, but Minamisoma not
```

```
function real HALF_EV
  if (time1/evtime < 1) and (time2/evtime > 1) then
  begin
     l2 = 1
  end
  else
  begin
     l2 = 0
  end

  pdose2 = 0

  evac_factor = (evtime-time1)/evtime
  if evac_factor < 0 then evac_factor = 0
return l2
```

Level 3 PSA – Swedish Pilot Study

Pilot Project Plan Report

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1. Introduction

The pilot project is separated into two parallel activities: the *Swedish* Pilot Project, and *Finnish* Pilot Project. The contents of this report are related to the Swedish pilot project.

This technical 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.

2. Project Goals

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 project plan and were used as a basis for the formulation of the scope of analysis of the project.

The project goals identified are the following:

- The pilot project should clarify what insights that can be gained from a Level 3 PSA.
    - The project should demonstrate what additional can be gained in addition to Level 2 PSA (e.g. when threshold criteria are imposed on nuclear releases what if threshold is exceeded marginally or substantially)
- The project should demonstrate and report on the resources required to perform a Level 3 PSA
- The project should develop a clearer understanding of what the key uncertainties of Level 3 PSA are.
- The project should determine how the existing release category structure fits-in to off-site consequence needs.
    - To what extent does the existing Level 1 and Level 2 PSA framework in the Nordic countries accommodate Level 3 PSA analysis, and what is lacking in existing studies?
- The project should further investigate the application of the risk metrics proposed in Task 1.
- The lessons learned during the pilot project should be applied/ communicated in Guidance Document.
- The pilot project should identify development needs and future work
- The pilot project should provide additional, practical insight, for contributing to external organizations e.g. IAEA
- The pilot project should try to determine the risk importance of filtered containment venting systems
- The pilot project should investigate to what extent can "shortcuts" and surrogate metrics provide insight to off-site consequence analysis.

3. 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.

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.

5. Scope of Analysis Report

The Scope of Analysis report will describe how the project intends to satisfy the project goals provided in Section 3. 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.

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.

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.

Level 3 PSA – Swedish Pilot Study

Swedish Pilot - Input Specification Report

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[4]VTT

Table of contents

1. Introduction

This technical report represents the first in series of reports that will be released detailing the Swedish Level 3 PSA pilot project.

The pilot project has been separated into four primary reports. These reports will cover the following topics: Input specification, Scope of Analysis, Methodology Specification, Application and result interpretation.

The pilot project is separated into two parallel activities. The "Swedish" and "Finnish". Pilot projects. The contents of this report are mostly related to the Swedish pilot project.

2. Purpose

The purpose of this report is to 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.

3. Input data requirements for Level 3 PSA

The Swedish project will be 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 2014 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.

3.1. 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." [3]

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.

3.1.1. 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 3.4.1 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 analyzed 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.

3.1.2. 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.

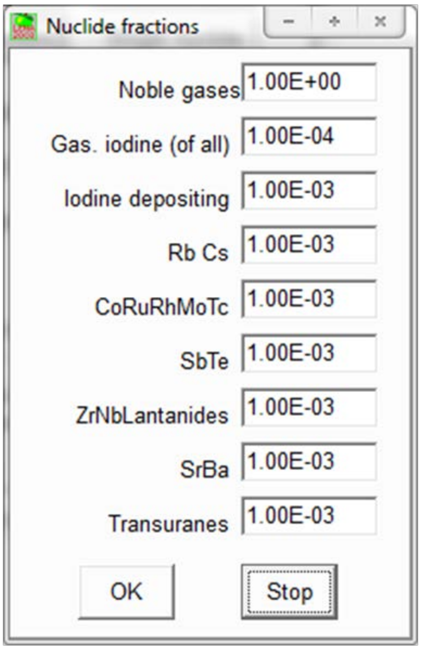

Figure 1. Isotopic fractions as shown in the LENA Graphical User Interface.

3.1.3. 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.

3.1.4. 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.

3.1.5. 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.

3.1.6. 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. 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. 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.

3.2.2. 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.

3.2.3. 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.

3.2.4. 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.

3.3. 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.

3.3.1. 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.

3.4. 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.

3.4.1. 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.

3.4.2. 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 [12].

4. 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.

4.1. 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 4.2 .

4.1.1. 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 1.

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 1. 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	[7]
EPR	- EPR Level 2 PSA largely available (UK-EPR submittals) - Relevant due to current Finnish- construction	- Not especially relevant to current operating Swedish reactors.	[8], [9]
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	

4.1.2. 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).

4.1.2.1. 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. [10]

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.

4.1.2.2. 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 4.1.1. Like WASH 1400, the results have been argued as being conservative, (however less so than those in WASH 1400). [11]

4.1.2.3. 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. [4]

4.2. Source term input

The previous section, Section 4.1, 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 2. 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 2. 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

4.3. 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 [2]. 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 2 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 3 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/

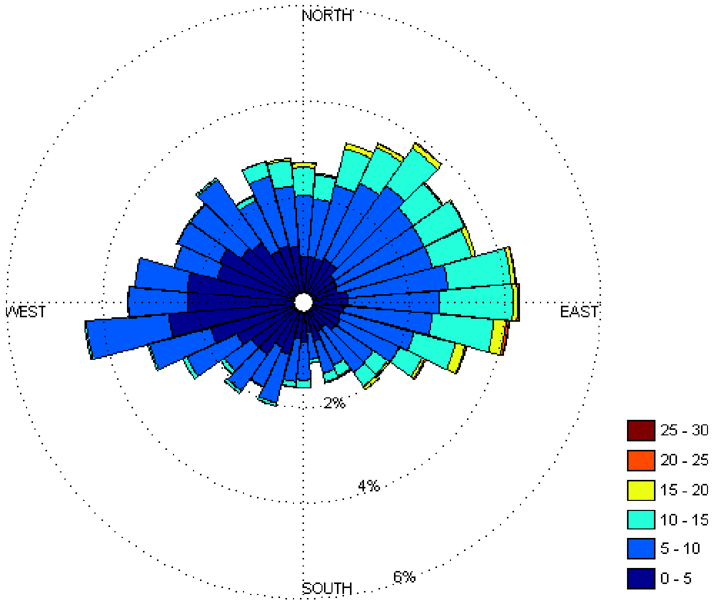

Figure 2. Wind rose showing distribution of information used in this analysis. The weather information was provided by SSM and characteristic of weather conditions in southern Sweden. Velocities scaled in meters per second [m/s]

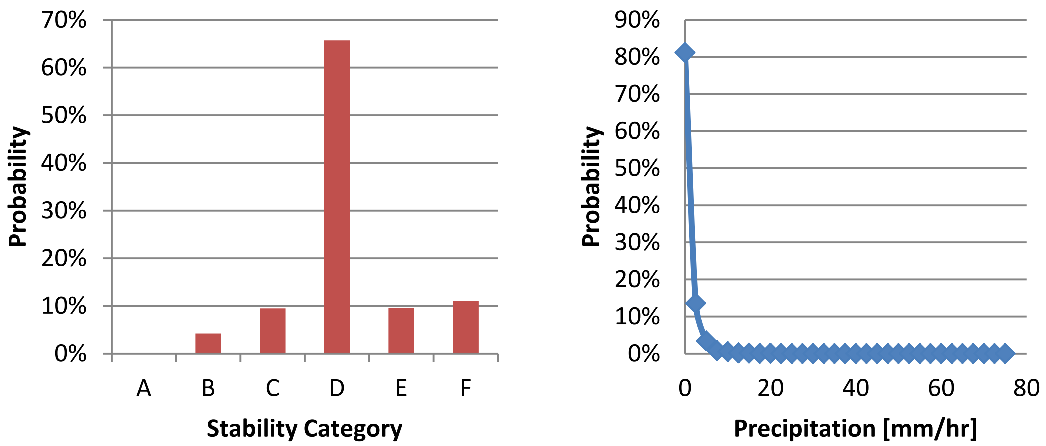

Figure 3. (left) The distribution of stability categories used in this analysis. A representative characteristic of the Swedish climate is the complete lack of the highly unstable atmospheric class A. (right) This figure shows the distribution of precipitation by probability.

4.4. 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 [2].

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 4. 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.

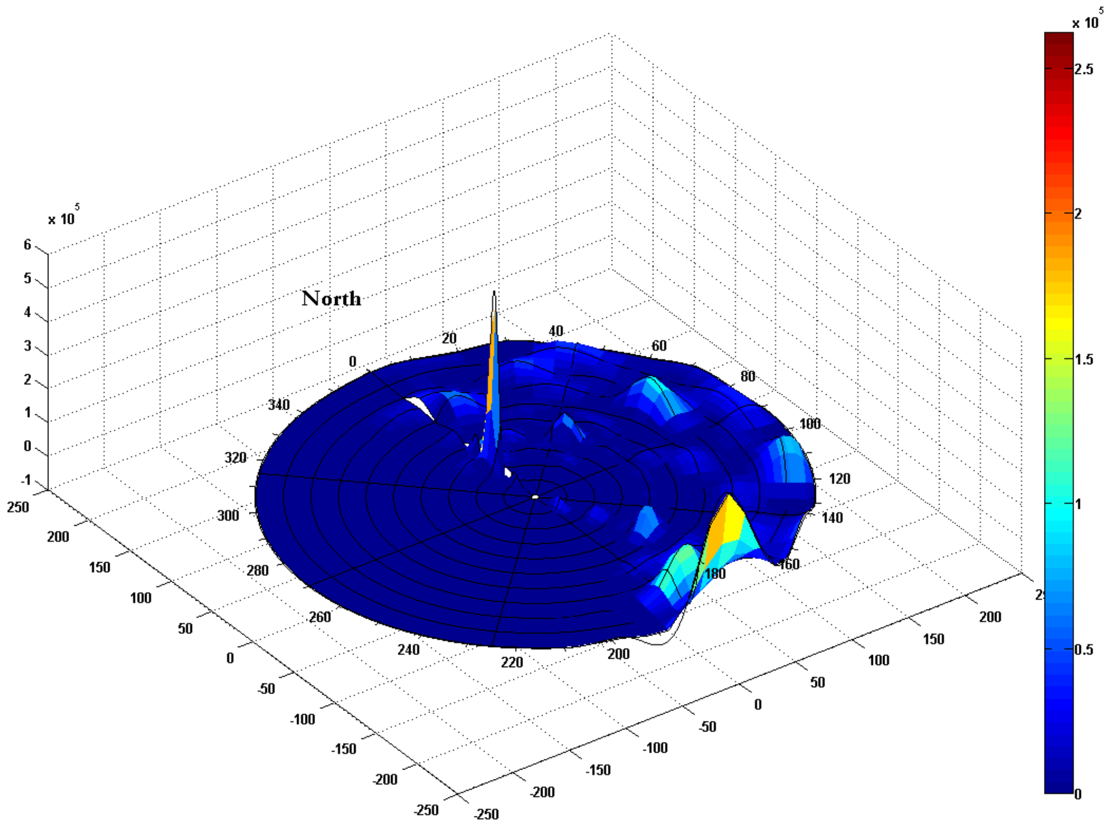

Figure 4. Surface plot representation of population distribution used in this analysis, representative of southern Sweden with maximum distance stretching 200 km and a coastal location. The colour bar to the right of the graphic shows the correspondence between the colours of the surface plot and the population per sector.

4.5. 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 3. More discussion on how countermeasures will be calculated will be provided in the Methodology Specification Report.

Table 3. SOARCA Surrey Shielding Factors [13]

	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

4.6. 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.

5. LENA input format requirements

A probabilistic version of the LENA program, LENA-P, was included in an international code comparison, during a significant international effort in the field of Probabilistic consequence analysis, which took place during the 1980s and early 1990s [4]. Since this large benchmarking study, LENA has been utilized, maintained, and even updated, but the program has been largely useful in the field of emergency preparedness. Most of the probabilistic capabilities that had been incorporated in LENA-P are no longer maintained, or, have since been removed. These probabilistic capabilities enabled the program to provide weather and release information and run LENA in a large batch configuration without a Graphical User Interface.

The LENA program is now largely a GUI controlled program, which does not allow the user to provide input data in a large single batch operation, or to easily automate the process of running the program. This makes large scale probabilistic analyses quite difficult to perform in LENA, in its current configuration. There are many methodological limitations in using the LENA program as well, however, the discussion of those considerations will be provided in a later report, which will specifically address the subject of methodology.

The LENA program categorizes two main input types for the purposes of performing an analysis. These data are "Weather data", and "Release data. The data that comprise these categories and the formats that LENA requires are summarized in this section.

5.1. Weather data

LENA uses a straight-line Gaussian plume dispersion model, which has very minimal weather data requirements. These are however, consistent with the state-of-practice in Level 3 PSA.

5.1.1. Data requirement

LENA requires the following parameters, and units for performing the Gaussian plume and dose conversion calculations:

- Wind speed [m/s],
- Wind direction [degrees],
- Mixing height [m],
- Pasquill Stability class [A-F (1-6 in batch input file)]
- Precipitation [mm/h].

These data are assumed to be for the local conditions at the point of release. Precipitation is handled differently depending on if the user is inputting data using LENA's batch mode or using the GUI interface. The batch mode method assumes the precipitation is present and uniform for the duration of the release. The GUI version of LENA allows the user to specify where along the plume trajectory there is precipitation or after during what time window there is precipitation. This functionality is not available if one performs an analysis in batch mode. The following weather data may be inputted when using the GUI for a single plume calculation:

- Distance to the beginning of rain [m],
- Distance to the end of rain [m],
- or
- Rain starts [h],
- Rain stops [h].

5.1.2. Format

When running LENA in batch mode the user must specify each of the bulleted items from Section 5.1.1 for batch files. These data are separated into cases by placing the data on separate lines. The input file formatting is shown in Figure 5.

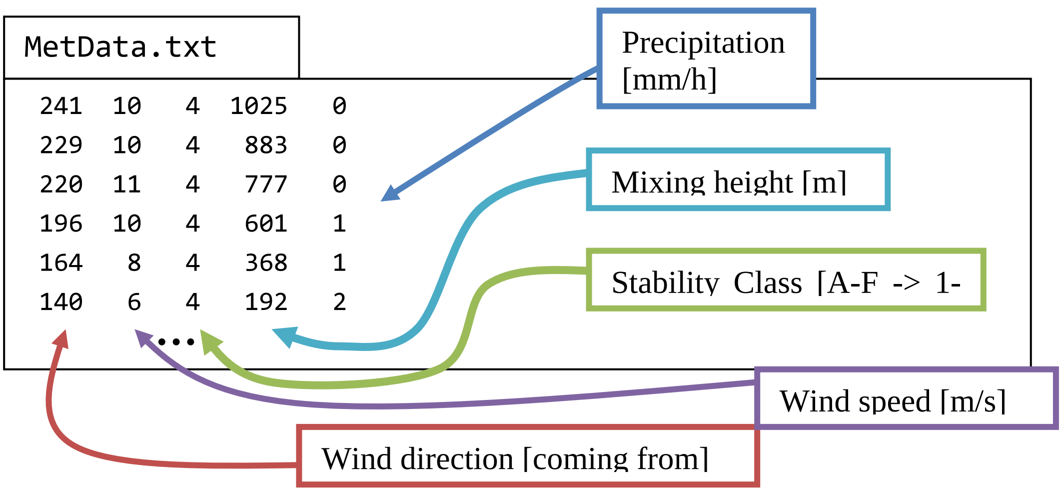

Figure 5. Meteorological data input for LENA batch mode.

5.2. Release data

The release data that LENA requires can be loaded from a set of files or input in the graphical user interface (GUI). The input of these data cannot be done in a batch mode as is possible with the weather information.

The source term input for LENA requires the following information:

- Reactor core power [MWth],
- Heat of plume [MWth],
- Release height [m],
- Delay [h],
- Duration of release[h],
- Release fractions [%],
- Release coordinates.

The release coordinates are not really of significant importance since probabilistic studies are performed in batch mode, therefore the user must take care to track the directions of the plume calculation.

5.2.1. Format

The release data can be defined using a series of GUI dialogues or through importing several input files. In order to reduce user errors while performing the analyses it is recommended to develop the source term files and importing them in LENA. The source term information is defined in three files, a "Block" file (*.BLK), an "Inventory file" (*.INV), and a Source Term file (*.SCT).

The Block file (*.BLK) defines the release coordinates (plant coordinates), the thermal output of the plant, the name of the inventory file, and whether the inventory file reflects the absolute number of radionuclides in the core at the time of the accident or a relative nuclide fraction. The Block file is shown in Figure 6.

The inventory file (*.INV) is a long list which defines the radionuclide abundances for the entirety of the nuclear core, at the time of the accident. The eventual release that will be modelled is calculated as a fraction of this nuclide inventory. The a sample of an inventory file is shown in Figure 7.

The final file, which makes up the source term is the source term file (*.SCT). This file defines the release fractions (fractions of the core inventory defined in the inventory file) as well as the release energy, release height, delay and duration of the release. The LENA GUI interface for defining the source term is shown along with the source term file which provides an analogous input in Figure 8.

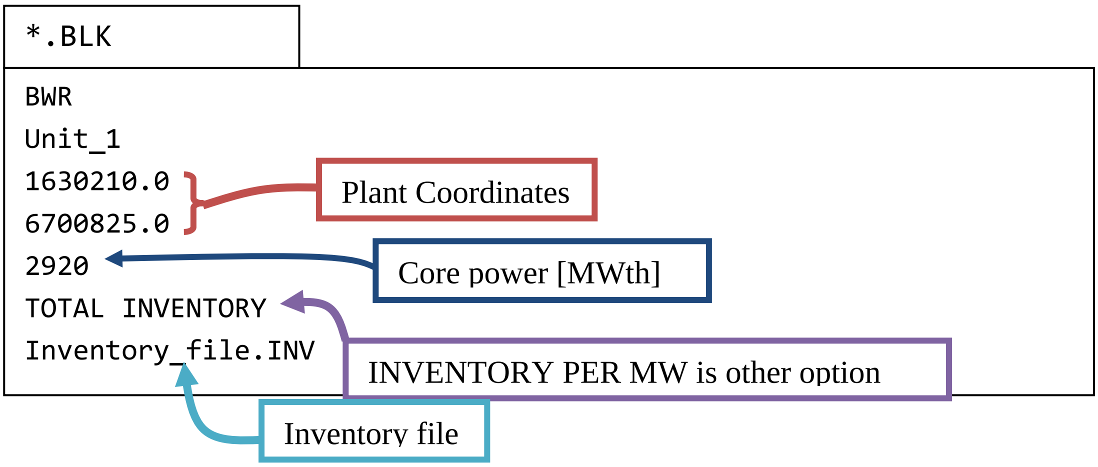

Figure 6. "Block" input file for input of plant coordinates, core thermal power, absolute/relative inventory, and inventory file.

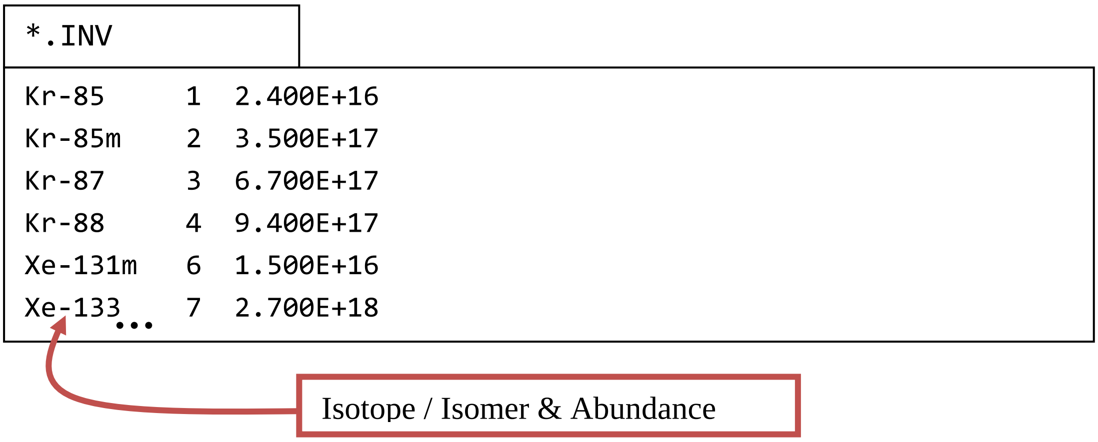

Figure 7. Meteorological data input for LENA batch mode.

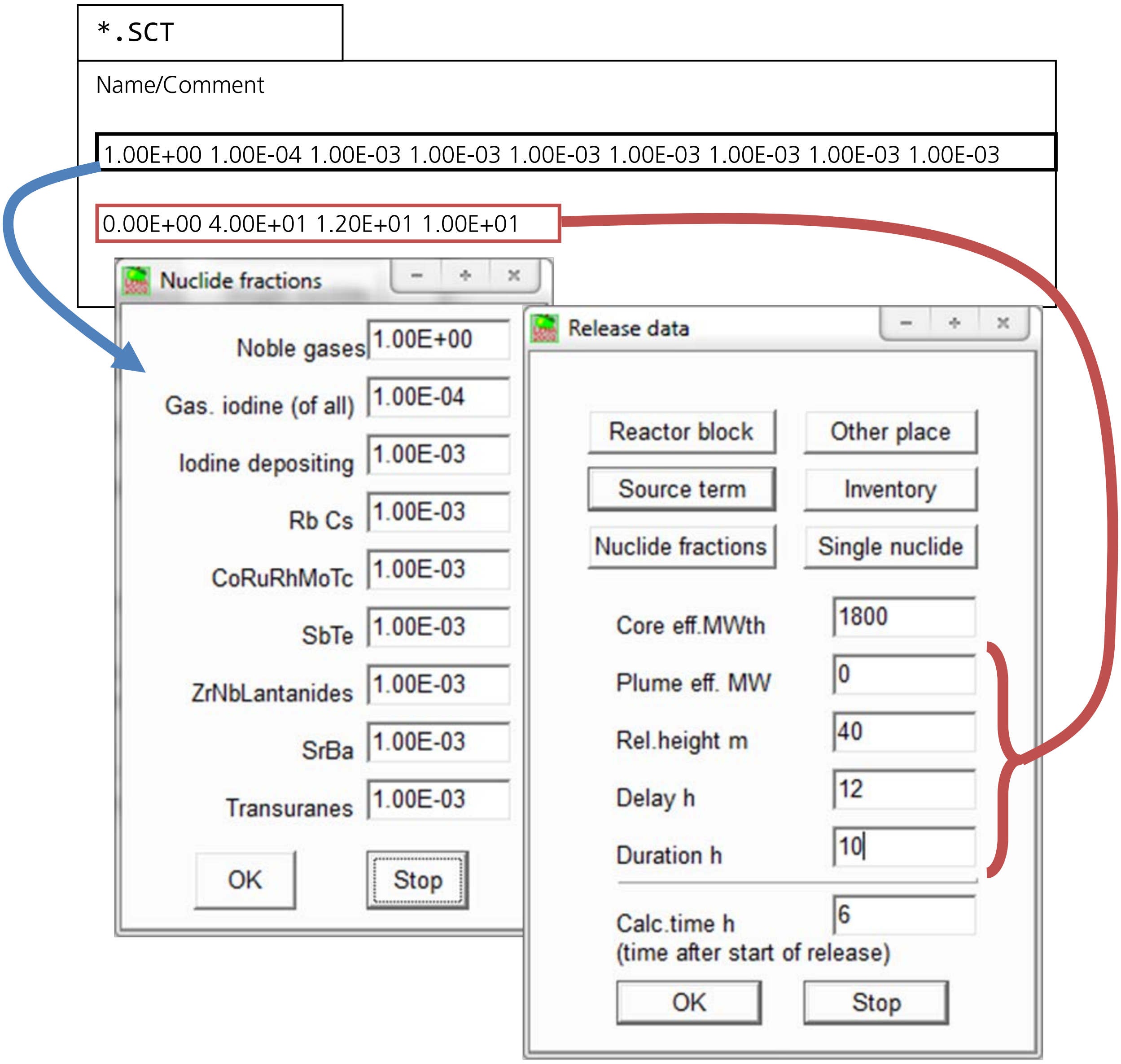

Figure 8. LENA 2.3.0 Source term file and GUI input comparison

5.3. Output options

The graphical user interface easily allows the user to access the calculation results (concentrations and relevant doses) in many ways. The user can click on the map, specify a down wind and cross-wind distance, or save to a file of results with distances 1-75 km.

For the purposes of a Level 3 PSA, LENAs batch mode for weather input will be the most useful way to perform many calculations for various weather conditions, and various distances. In order to run LENA in batch mode one must define all of the source term parameters as discussed in Section 5.2, then select "Doses > Batchrun" from the menu bar. The batch run will request a file that defines the downwind distances that shall be calculated and the meteorological data file shown in Figure 5.

5.4. Advanced options

LENAs interface is designed to be simplistic, with most "advanced" options set to default settings, so that in an emergency situation the software can be quickly used by emergency preparedness groups. For use in probabilistic risk studies these assumptions may need revision, or might be of interest in the form of sensitivity studies, such as:

- Shielding factors
- Filter factor
- Deposition velocities

Fortunately, these advanced parameters can be modified. The factors that can be modified in the LENA program via the GUI interface are shown in Figure 9. These cannot be saved to be used in subsequent sessions.

The advanced options may also be accessed via the LENA2003.INI file. This file includes configuration settings for the Graphical User Interface and graphical output (e.g. language, directory designations, map coloration), which are largely irrelevant for this Level 3 PSA analysis. The configuration file also allows for the modification of each of the dispersion parameters and coefficients shown in Figure 9. For a full discussion on each of the modifications that can be made in the LENA configuration file and how they affect the methods in LENA consult the user manual [6].

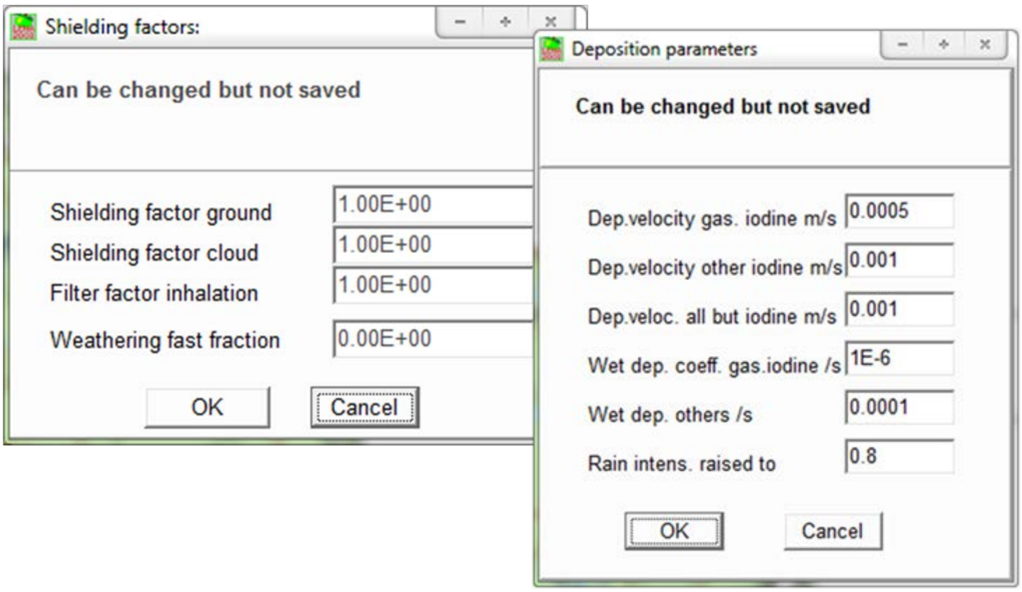

Figure 9. Advanced options in LENA can be changed through the GUI interfaced (but not saved for later sessions), or modified directly in the *.INI file.

6. References

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Level 3 PSA – Swedish Pilot Study

Swedish Pilot Project Analysis Scope Specification

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Table of contents

1. Introduction

This report 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. Therefore, it is instructive to list the main goals which the scope of analysis would ideally seek to cover:

1. Cover which types of insights can be attained from a L3 PSA
    a. Discrimination of consequences which exceed a regulatory risk threshold, eg released activity, marginally or substantially.
    b. Seek to establish to which extent L2 PSA output may be relevant as a surrogate for L3 PSA insights.
2. Indicate resources required for performing a L3 PSA
3. Identify any key uncertainties in the analysis
4. Indicate how existing plant L2 PSA structure would interface with a L3 PSA analysis
5. Gain insights into the use of L3 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 L3 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.

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 affects the analysis scope is discussed in more detail in separate sections below, as these constraints serves 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 PCA code.

The next sections presents 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.

1.1. Purpose

1.2. 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.

1.3. 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. 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 L3 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 ^{137}Cs.

An illustration of the available LENA output is given in Figure 1.

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.

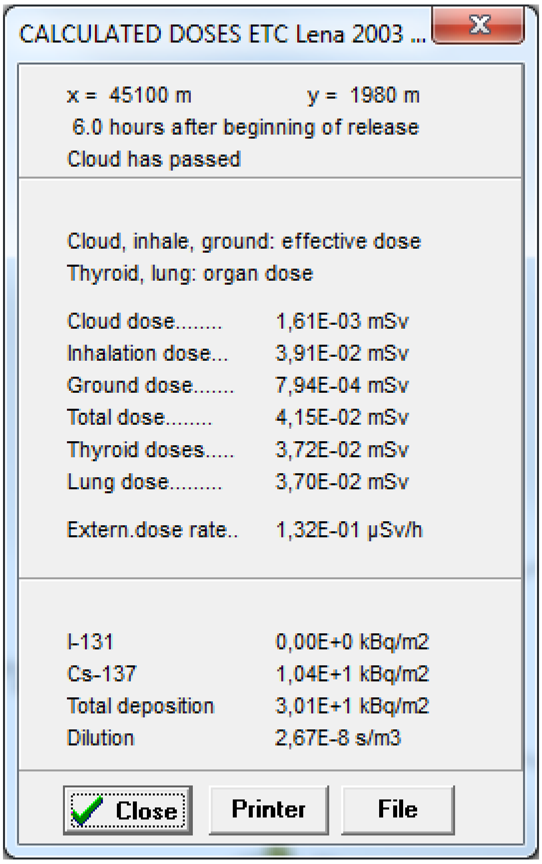

Figure 1: Core output produced by LENA 2003 (note that the specified release in this case only consisted of Cs).

3. 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 which serves as input data to the LENA code.

As reported in the Input Specification [1], 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 [2] 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 was 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 [3] for more information) for a reactor of the size of the EPR (see [6]) 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 1 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 1: Selection matrix for SA scenarios for further Level 3 PSA analysis. RC numbers and Cs release fraction in parenthesis are from [2].

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 1, 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 1.

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.

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

Performing probabilistic consequence analysis calculations for the specified scenarios listed in Table 1, 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 2 below.

The analysis scope envisions looking at three key analysis characteristics together with a set of 5 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, [3] 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. One particularly simple way of performing the latent cancer fatalities evaluation is given in [4] (see the methodology specification report [5] for more details).

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.

Table 2: 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 maximum 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 threshold	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.

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). Further details of the methodology behind the risk measure is given in the Level 3 PSA project's methodology specification report (see [5]).

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 [4] 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 2 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.

5. Summary of analysis scope for the Swedish Pilot Project

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 1 combined with the Level 3 PSA analysis cases proposed in Table 2.

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.

As can be seen, 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.

7. References

[1] Level 3 PSA Project---Swedish Pilot Project: Input Specification Report, 2015

[2] Hinkley Point C Pre-Construction Safety Report (PCSR), Sub Chapter 15.4, Level 2 PSA, 30/07/2012, Doc. No. HPC-NNBOSL-U0-000-RES-000035, NNB Generation Company LTD

[3] Villkor för fortsatt tillstånd enligt 5 § lagen (1984:3) om kärnteknisk verksamhet att driva kärnkraftsreaktorerna Ringhals 1, 2, 3 och 4, Regeringsbeslut 11, 1986-02-27

[4] Sunde, C & Holmberg, J.-E. 2014. "Addressing Off-site Consequence Criteria Using Level 3 PSA Task 1 – Appropriate Risk Metrics, Final Report". Risk Pilot.

[5] Level 3 PSA project's methodology specification report, 2016.

[6] Hinkley Point C Pre-Construction Safety Report (PCSR), Sub Chapter 4.1, Summary Description, 26/03/2011, UKEPR-0002-041 Issue 03, NNB Generation Company LTD

Level 3 PSA – Swedish Pilot Study

Methodology Specification

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Table of contents

1. Introduction

This report details the methods used in the Swedish Pilot Study. The methods employed in the spreading calculation were performed with LENA 4.0. Specific discussion of the LENA software is not provided in this report. A complete description of the methods and models in the LENA dispersion analysis are available in references [1] and [2].

1.1. 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.

1.2. 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. Source Term

Source terms are limited to the information provided by the Pre Construction Safety Report (PCSR) for the UK EPR [10].

Single phase calculations with the exception of the late failure releases which are split into two release phases

2.1. General inputs

General inputs are provided to LENA in the *.BLK file. This file includes the reactor type (e.g. BWR, PWR, etc.), plant coordinates, reactor thermal power rating, and a reference to the inventory file.

This study is a generic study based on an example population distribution and not a specific plant site as detailed in the input specification report [9]. Therefore, the general input file only details the thermal power of the reactor, which for the EPR is 4500 MWth.

2.2. Reactor core inventory

A specification of the reactor core inventory is also required for LENA. This information is provided to the program in the *.INV file. This file specifies the quantitiy of the calculated nuclides in the reactor at the time of the accident in units of Bequrels [Bq].

The reactor inventories were not explicitly provided in the publicly available PCSR report [10]. What is provided in the PCSR are release fractions and the quantities released. These were in close agreement with publications provided to the USNRC for licensing of the US EPR, but far beneath the very “conservative” values provided by AREVA in the response to Requests for Additional Information [11].

Ultimately, the inventories for the study were calculated by combining the information that was provided in the PCSR for release fractions and quantities released. Since release fractions and release quantities were provided for each release category the calculated inventories could be compared for the same nuclide across each of the release categories. It was found that there was a very slight variation in the calculated nuclide inventories between release categories, which provided some confidence in the calculated values. Figure 1 shows a histogram of the standard deviations of each of the calculated inventories for all isotopes divided by the mean. Since only three significant digits were provided in the tables much of this variation is likely due to rounding error.

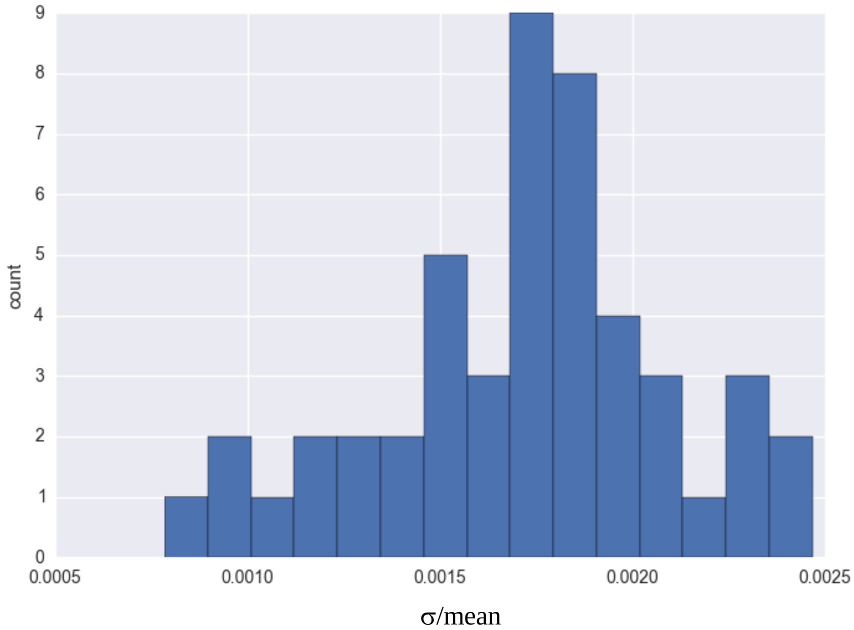


Figure 1. Verification calculation of inventory values derived from UK-EPR PCSR.

Further, these values were compared to generic PWR inventories and were found to be in fairly close agreement with general PWR inventories when scaled to the large thermal power of the EPR of 4500 MWth. Several isotopes that are included in LENA were not provided in the release quantity table in the PCSR. These isotopes were calculated using generic data provided by LENA for 4-Loop PWR inventories.

2.3. Release Fractions

Release fractions are derived from the values provided in the UK EPR PCSR. The values provided correspond to the MAAP release fraction output (‘FREL’). These do not have a 1-1 correspondence to the values used by LENA therefore a bit of work is performed to translate the 12 FREL values from MAAP to the 9 Groups required for input into LENA. The methods used for each of the groups are provided in subsections 3.3.1 through 3.3.9.

For extended releases the release fractions are provided in two phases in the UK EPR PCSR. The early and late phase of the releases are input into LENA separately and then doses are calculated as the sum of the two phases.

No corrections are made for the values that are actually released. This implies that a significant over-estimation is made for many of the source terms.

2.3.1. Group 1 (noble gasses)

Noble gases are treated the same between MAAP and LENA therefore the release fractions can be taken directly from FREL 1 to LENA Group 1.

2.3.2. Group 2 (organic iodine)

In the UK EPR PCSR source term tables organic iodine is explicitly provided in release fraction group FREL 2a. These values are used to calculate the organic iodine release fractions.

2.3.3. Group 3 (inorganic iodine)

The values provided in the UK EPR PCSR for FREL 2a corresponds to the inorganic iodine release. These values are therefore used for Group 3 of LENAs calculations.

2.3.4. Group 4 (cesium)

Group 4 release fraction group in LENA represents the fractional release of cesium. In MAAP cesium is provided in two separate groups (namely, FREL 2 [CsI] and FREL 6 [CsOH]). To remedy this discrepancy some considerations and approximations need to be made. If the total fraction of CsOH is approximately the same or larger than the fraction CsI then it would be okay to approximate the cesium release as the quantity CsOH. In other cases the total cesium release can be approximated using the following formula:

$$z = \frac{x \cdot n_{CsI} + y \cdot n_{CsOH}}{n_{Cs}} \tag{1}$$

In this formula z, x, and y are respectively the fractions of Cs (to LENA), CsI, and CsOH (from MAAP). The values n is the molar mass of the respective chemical compounds (based on their subscripts).

2.3.5. Group 5 (Co, Ru, Rh, Mo, Tc)

LENA group 5 correlates to MAAP group 5.

2.3.6. Group 6 (Sb, Te)

Similar to how cesium is handled Sb and Te occur in several MAAP groups (FREL 3, 10, and 11). Using the same methodology applied in equation 1 the fraction of Te can be derived. On occasion the fraction Sb (MAAP group 10) can dominate this group and therefore the derivation of Te may be irrelevant. Furthermore, for large releases of iodine the whole of group 6 may be dwarfed by the effects of iodine.

2.3.7. Group 7 (Zr, Nb, La, Lanthanides)

LENA group 7 correlates to MAAP group 8.

2.3.8. Group 8 (Sr, Ba)

Sr and Ba are represented in two MAAP groups, 4 and 8. These can be handled using the same method outlined in equation 1.

2.3.9. Group 9 (uranium & transuranics)

LENA group 9 correlates to MAAP group 12.

2.4. Source term parameters

The rest of the source term parameters are also included in the *.SCT files

These parameters mostly affect the shape and trajectory of the Gaussian plume. The parameters include the plume heat content, release height, delay from the start of the accident to release, and the duration of the release.

These parameters are conveniently provided explicitly in the UK EPR PCSR for each of the Release Categories.

3. Post processing (LENA result handling)

3.1. Dose calculations

The results of LENA calculations are dose, dose rate, and contamination levels at distances downwind of the release location. These data are calculated for each of the weather inputs (hourly data collected over 2 years). The maximum doses are experienced along the centerline of the plume. Doses off of the centerline are calculated by assuming the dose and contamination levels follow a Gaussian distribution away from the centerline.

3.1.1. Maximum individual dose at 1km & 5 km

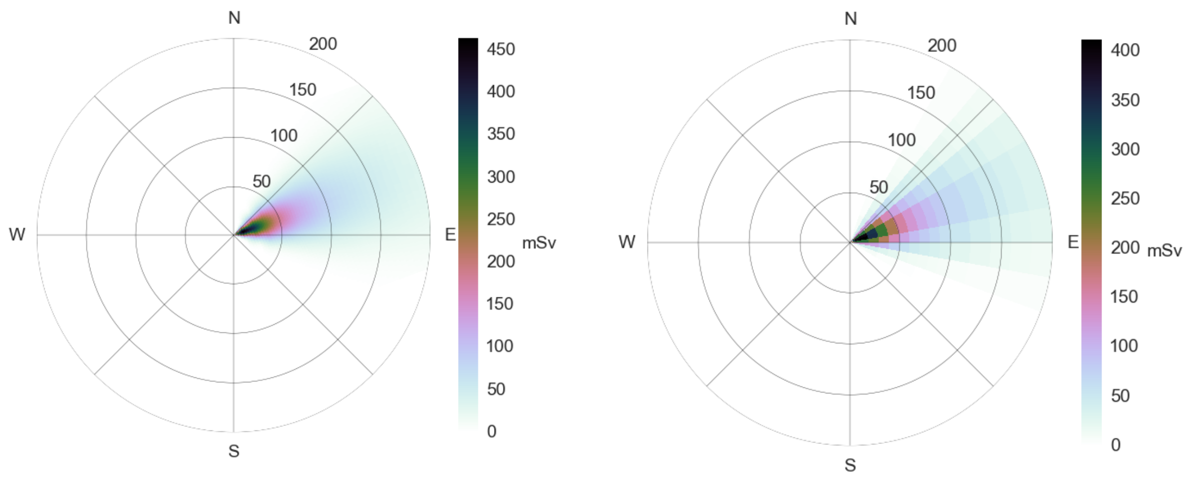
Dose calculations are calculated based on LENA outputs for each ground shine, cloud shine and inhalation dose. These are summed to determine the total whole-body dose. LENA then shows the

3.1.2. Collective doses

Collective doses are a way of quantifying the dose received to the public.

These are often used because it is a fairly simple metric, which is both easy to calculate and provides some perspective on the risks of the exposed population. Interpretation of collective doses needs to be approached with some caution as is further explained in Section 5.2.2.

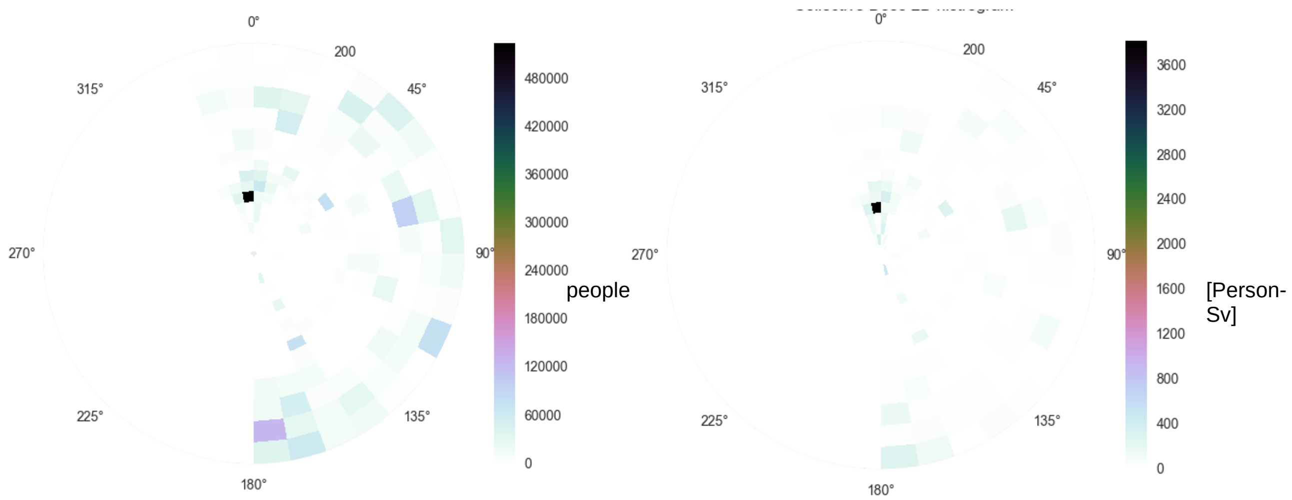
For a given plume LENA provides a 1 dimensional calculation of dose levels for distances that are defined by the user. These values define the centerline levels in the direction of the wind at the release point. Accompanying these centerline values is the standard deviation of the dose levels at each distance perpendicular to the wind direction. From these standard deviations one can calculate the width of the plume for each downwind distance. The result of these calculations for whole body dose of a single plume is shown in Figure 1a. In order to determine collective doses these data need to be combined with the discrete data provided for population. Therefore the calculated values were then discretized by using the average value in each of the sectors where the population data was defined. The discretization of the plume shown in Figure 1a is shown in Figure 1b.



(a) (b)

Figure 2. Calculation of doses for a single plume needed to be made from the 1D results provided by LENA. 1a, shows the plume calculation for a single weather case for release category 205. 1b shows the discretized values for the same plume into polar sectors.

The generic Nordic population used for the analysis as described in the input specification report is shown in Figure 3a [9]. This population is simply multiplied by the dose from the discretized plume calculation (Figure 2b) and weighted by the probability of the weather conditions (24 hours * 365 days * 2 years). The result is a weighted sum of the collective doses, shown for RC205 in Figure 3b.



(a) (b)

Figure 3. (a) shows the population distribution with the postulated release point centered at the origin. (b) Shows the weighted sums of collective doses for RC205 for all weather scenarios.

Collective doses are also collected for each plume individually. These data will also be analyzed and compared with the 2 dimensional weighted sums and statistically.

3.2. Health effects

Health effects are extrapolated from the dose calculations that are described in the previous section. The Health effects methods applied in this project are extremely simplified and are merely to provide some relatable scale to the doses calculated. Health effects studied in this report are confined to early health effects calculations for the individual doses, and latent health effects (cancers) based on the collective doses. Genetic health effects were not considered in this study.

3.2.1. Early health effects

Early health effects are very coarsely handled in the Swedish Pilot study.

The general methodology that is used is consistent with that developed in NUREG/CR-4214 [6].

The basic method is applied to the individual dose results described in section 3.1.1. The method defines cumulative hazard function, H, which is calculated from a two parameter Weibull function, shown in equation 2. The risk, r, can then be calculated by simply taking the complement of the exponential, shown in equation 3.

$$H = \ln 2 \cdot \left(\frac{D}{D_{LD_{50}}}\right)^{S} \qquad (2)$$

$$r = 1 - \exp(-H) \qquad (3)$$

D	Average absorbed dose
$D_{LD_{50}}$	Dose which causes 50% mortality of exposed population
S	Shape parameter

Sensitivity to shape parameter, S, is shown in this value depends on exposure rates and details of the exposed person, for example age. A comparison between how the above formulas depend on shape factor is shown in Figure 2. The dose calculations in LENA are based on those for an adult. Therefore, the analysis the 30 day lethal dose to 50% of the population was set at 3.0 Sv, which is a fairly conservative estimation for an adult. The shape factor, S, of 10 was used which is consistent with the discussion provided in NUREG/CR-4214.

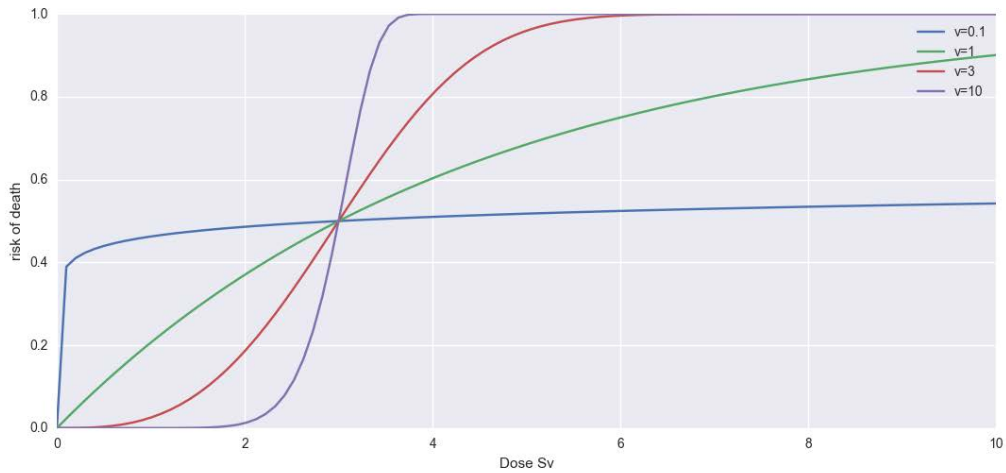


Figure 4. Sensitivity to shape parameter, S, on risk of early death.

3.2.2. Latent cancers

Ideally latent cancers would be calculated based on the exposure to individuals and the risk of those individuals to experience latent effects would be calculated based on the specific exposure levels they experience. Historically, such an analysis has not been performed due to computational limitations. Therefore more simplified analyses based on collective doses were performed. Due to resource limitations in this study a simplified approach based on collective doses is used.

Using collective doses to calculate latent health effects needs to be approached with some caution as they can be difficult to interpret in extreme cases with very high populations receiving low doses or small populations receiving low doses. For very high populations receiving low doses a high collective dose may result. The health effects, namely cancers, to humans from very low radioactive doses are difficult to differentiate from “naturally” occurring cancers. Likewise, small populations receiving very high doses will likely under predict the cancer rates or event early health effects that will occur. Ideally, collective doses are used when a population with similar risk profiles is exposed to similar doses.

In this study a very simplified method is used based on qualitative findings from Chernobyl survivors. Roughly 100 mSv exposure lead to a 0.5% increase in developing cancer [7]. Therefore, for this analysis a slightly more conservative figure is used of 1% increased risk, which is consistent with the simplified method described in [6].

It must be noted that double counting of the population that experiences early health effects is possible in this analysis. Since most of the release categories do not expose significant populations to large doses this should have a very minor effect on over estimating the number of cancers.

3.3. Environmental

Environmental effects from a nuclear accident can be numerous from health effects to plants and animals, ground and surface water, or agricultural land. The Environmental impact assessed in this study is greatly simplified. The impact is assessed by quantifying the land areas that exceed a cersium-137 contamination of 100 kBq/m^2 and 1000 kBq/m^2.

Much like dose, LENA provides a calculation of the activity of Cs-137 along the centerline of the plume along with the standard deviation of the contamination in the cross-wind direction. The width of the contamination is calculated in much the same way that the width of the plume is calculated for dose (see Figure 2a). The data are discretized in the same way that dose was calculated and each of the plumes is summed together, weighted by the probability of occurrence. Since the discretization is not bound in terms of fidelity like the collective dose calculations which were limited by the population data the discretization could be significantly more detailed. The results are compiled and compared between release categories.

(a) (b)

Figure 5. (a) Weighted sum Cs-137 contamination of all weather conditions for RC404. (b) Weighted sum of Cs-137 contamination for all weather condition for RC205.

3.4. Economic

Similar to the other consequences analyzed, the scope of an economic analysis can vary greatly based on the consequences of interest. In this study economics were scaled based on persons displaced and the duration of displacement. The projected costs of the Fukushima disaster were the basis for the values calculated.

The approximation used was that as of 2016-01-08 Tepco had paid 5.814e+13 JPY in evacuation compensation to approximately 100000 individuals [12]. This corresponds to roughly 4E+06 SEK per person. Therefore the economic approximation used in this analysis was 1e+06 SEK per displaced individual, per year of displacement.

The correlation of Cs-137 deposition that would correspond to x-years of required displacement until the Cs-137 and Cs-134 activity levels fall to 1μSv/h was derived based on dose conversion factor of 1.48e-15 (Sv/s)/(Bq/m^2) for Cs-134 and 5.51e-16 (Sv/s)/(Bq/m^2) for Cs-137. The Ratio of Cs-134/Cs-137 for each of the EPR releases is approximately 1.5. The correlation is shown in Figure 6.

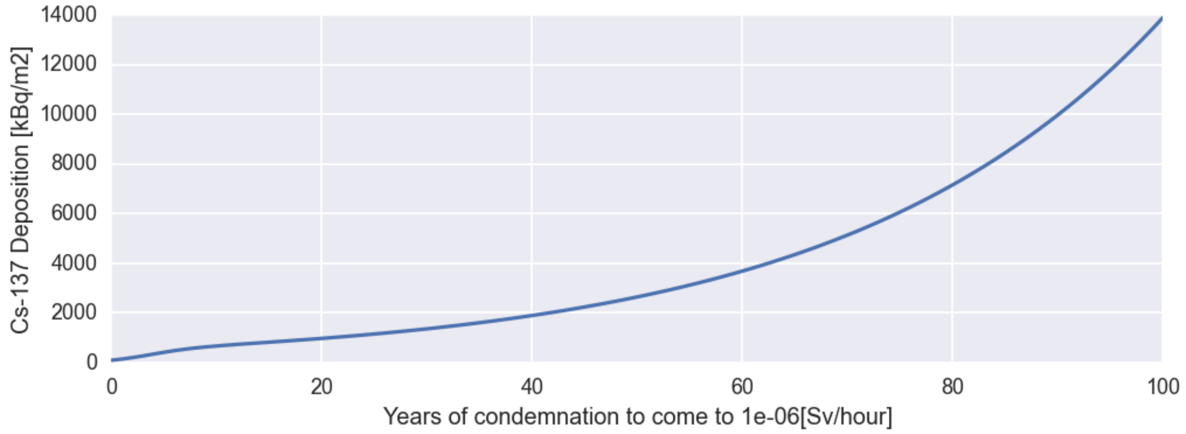


Figure 6. Correlation between Condemnation time and Cesium-137 deposition.

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Level 3 PSA – Swedish Pilot Study

Application and Result Interpretation

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1. Introduction

The pilot project is separated into two parallel activities: the *Swedish* Pilot Project, and *Finnish* Pilot Project. The contents of this report are related to the Swedish pilot project.

This technical report represents the final in the series of reports that describe the Swedish Level 3 PSA Pilot Project. This report describes 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.

Preceding this report are four additional report which outline the input, scope and methods used in the Swedish pilot study:

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

1.1. Purpose

The purpose of this report is to summarize the results and findings of the Swedish Pilot study.

1.2. 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.

1.3. 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. Analysis scope and source term descriptions

The Scope of Analysis report explicitly defines the high level elements of project scope [1]. A brief overview of the source terms as related to the results and conclusions is provided in this section.

The scope of analysis is reviewed in section 2.1. The specifics of the source terms and some of the implications of the source term selection are discussed in section 2.2.

2.1. Scope of analysis

In the scope of analysis report the source terms to be analyzed are defined, as shown in Table 1 and the extent of the analysis is shown in Table 3 along with significant discussion and justification for the definition of the scope. Many of the study limitations are expressed in greater detail the Input Specification [2] and Methodology Specification Reports [3]. The assumptions and details regarding calculations timing and result presentation are developed in this report.

Table 1: Selection matrix for severe accident scenarios for further Level 3 PSA analysis. RC numbers and Cs release fraction in parenthesis are from [1].

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.

2.1.1. Exposure times for each analysis

Essentially, there are four different metrics included in the Swedish pilot study, individual health effects, latent health effects, environmental effects and economic effects. The timing used for each of these analyses is a little peculiar and perhaps is an element that could be further improved in future analyses. In LENA one specifies the number of hours after the start of the release each calculation is performed. The results from LENA are doses due to continuous exposure during this time or the deposition of Iodine and Cesium on the ground up to this time. For each of the metrics described, a separate calculation time was considered. These times are shown in Table 2.

Table 2. Definition of exposure lengths used for each of the analyses.

Analysis	**Calculation time**
Individual health effects	Doses calculated with continuous (unshielded exposure) for 2 days (48 hours) after start of release
Latent health effects	Doses calculated with continuous (unshielded exposure) for 7 days after start of release
Environmental effects	30 days after start of release
Economic effects	# of evacuated persons 30 days after start of release

Individual health effects are calculated 48 hours after the start of each of the releases without shielding or sheltering. This implies that in the worst case it is postulated that individuals will be exposed for 2 full days. A more realistic consideration of warning time could probably be considered in future analyses. These assumptions were developed with advice from the Swedish working group members in the NKS “NORCON” project [5].

Latent health effects health effects are calculated 7 days after the start of the release without shielding or sheltering. This time interval is consistent with the dose criteria for evacuation of 20 mSv per week, according to the “Nordic Flagbook” [11].

For environmental and economic effects the cesium deposition levels calculated 30 days after the start of release were used to perform the analysis. In a few test cases, which were analyses it was found that the peak deposition values were consistent with the values calculated after 30 days.

Table 3: Scope of analysis 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 maximum 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	10 km evacuation zone	X	X	-	-	-	-
Cs‡ ground contamination threshold	1000† kBq/m^2	-	-	-	-	X	X
	100† kBq/m^2	-	-	-	-	X	X

†: Cs ground contamination thresholds.

‡: Combined activity of ^{137}Cs.

2.1.2. Presentation of results

In this analysis, separate results are calculated for each of the release categories, for each of the weather cases used (hourly data over 2 years, 17520 data points). Therefore, a statistical set could be analyzed for a given source term.

Results are presented for each of the risk metrics in several different ways.

- Tables - the mean values or median values across weather cases are presented as simple tables for each of the investigated cases.
- Box plots –box plots illustrate the ranges of statistical data.
- Simple bar charts – in cases where the frequency weighted sums were analyzed there was just a single value result for a given metric.
- Exceedance curves – In order to incorporate the frequency for a given consequence, exceedance curves were used to compare source terms.

2.1.2.1. Mean and median

The dose to the most exposed individuals and the acute individual health effects are presented in terms of median values for each release category. Median values were chosen to investigate individual risk metrics to better understand the maximum possible dose for an “average” case. If the mean values were investigated it can be postulated that a significant impact could be made by the extreme cases. (Another possibility would have been to remove the outliers and look at the averages).

The collective dose calculations, however, are presented and discussed in terms of the mean values of all of the analyzed weather cases. This seemed reasonable since the collective doses are handled stochastically across a population.

2.1.2.2. Box plots

Box plots are used to show the range of the data across all of the weather cases for a given release category. The red line in the middle of the box for each plot shows the median values, the edges of the box refer to the 25 and 75% percentiles, the whiskers of the box plot are calculated by extending the values of the 25 and 75 percentile values by 1.5 times the range of difference of the 75 and 25 percentiles as shown in the following equation:

$$whisker_{top} = q_{75\%} + 1.5\ (q_{75\%} - q_{25\%})$$

2.1.2.3. Frequency weighted sums

In order to preserve the directional component and to develop an understanding of the risks in terms of direction, probability weighted sums of each of the weather cases were also calculated for several of the metrics. These are calculated by performing the calculation for each of the cases and calculating the values for each polar sector. These values calculated for each of the sectors is then divided by the total number of cases (which was 17520, number of hours in 2 years, for all of the analyses performed in this study). The values for each of the sectors are then summed over all of the cases. These results are shown graphically and compared against the mean and median values for each of the individually calculations. It should be noted, that for collective doses this yields the same results as taking the mean, however, for threshold calculations (such as the contamination calculation), these yield significantly different results.

2.1.2.4. Exceedance curves

Exceedance curves, also referred to as Cumulative Complementary Distribution Function (CCDF), or Farmer curves, are used to incorporate the relative frequency of each of the release categories as well as the magnitude for a given metric. An exceedance curve is a plot showing the probability or frequency that a value is met or exceeded. Therefore these plots are monotonically decreasing.

2.2. Source Term

Source terms are limited to the information provided by the Pre Construction Safety Report (PCSR) for the UK EPR [4]. Single phase calculations with the exception of the late failure releases which are split into two release phases were performed.

A full description of the internal initiating events considered in the PSA analysis is provided in the Level 1 PSA chapter in the PCSR [6]. A description of the screening process and associated frequencies for external hazards is provided in [7]. Initiating events considered in the analysis include both internal and external events. The associated frequencies are developed for a Generic EPR built in the UK. Since these values are ultimately what are used to derive the release category frequencies these are the instating event frequencies that were applied to this study, even though these frequencies may not be representative of a generic Nordic site.

Internal hazards considered include the following:

- pipe leaks and breaks,
- failures of vessels, tanks, pumps and valves,
- missiles,
- dropped loads,
- internal explosions,
- fire,
- internal flooding.

External hazards considered include the following:

- earthquake,
- aircraft crash,
- extreme weather conditions: extreme snow and strong wind,
- Organic material (algae, fish, etc) and hydrocarbon-based pollution.

The release categories in the EPR study are divided into 29 separate groups based on seven attributes, which are listed below. These attributes largely correlate to the different possible outcomes from the containment event trees (CET).

1. Containment bypass versus no bypass
2. Time frame in which containment failure occurs
3. Containment failure category
4. Melt retained in-vessel
5. MCCI occurs
6. Melt flooded ex-vessel (covered by water)
7. Source term mitigated by sprays or (scrubbing for bypass sequences)

A description of source terms as they relate to the attributes is provided in Table 4. The release parameters and timing for each of the release categories considered are provided in

Table 5. The release category nuclide group release fractions are shown in Figure 1 and Figure 2

Table 4. Description of the Release Categories used in the Swedish pilot study.

Release Category	Description
RC202	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex-vessel, with containment sprays
RC205	Containment failures before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex-vessel without containment sprays
RC404	Containment failures after breach and up to melt transfer to the spreading area due to containment rupture, without MCCI, with debris flooding, without containment spray
RC501	Long term containment failure during and after debris quench due to rupture, with MCCI, without debris flooding, with containment sprays
RC502	Long term containment failure during and after debris quench due to rupture, with MCCI, without debris flooding, without containment sprays
RC503	Long term containment failure during and after debris quench due to rupture, without MCCI, with debris flooding, with containment sprays
RC504	Long term containment failure during and after debris quench due to rupture, without MCCI, with debris flooding, without containment sprays
RC802b	Small or Large Interfacing System LOCA, without Fission Product Scrubbing, fission product filtration in annulus and fuel/safeguards building ventilation systems

Table 5. Release timing parameters

Release Category	Frequency (/y)	Core uncover time (h)	Release Start time (h)	Release end time (h)	Release Duration (h)
RC202	2.60E-12	2.4	4.6	8.3	3.7
RC205	4.51E-10	2.4	3.5	10	6.5
RC404	1.09E-09	2.4	7.8	20.5	12.7
RC50x	NA	2.4	3.8	9	5.2
RC501	6.51E-13	2.4	60	70	10
RC502	3.96E-11	2.4	60	70	10
RC503	1.27E-09	2.4	85	125	40
RC504	3.29E-08	2.4	85	125	40
RC802b1	3.83E-09	1.3	1.8	2.9	1.1
RC802b2	3.83E-09	6.4	7.4	8.9	1.5

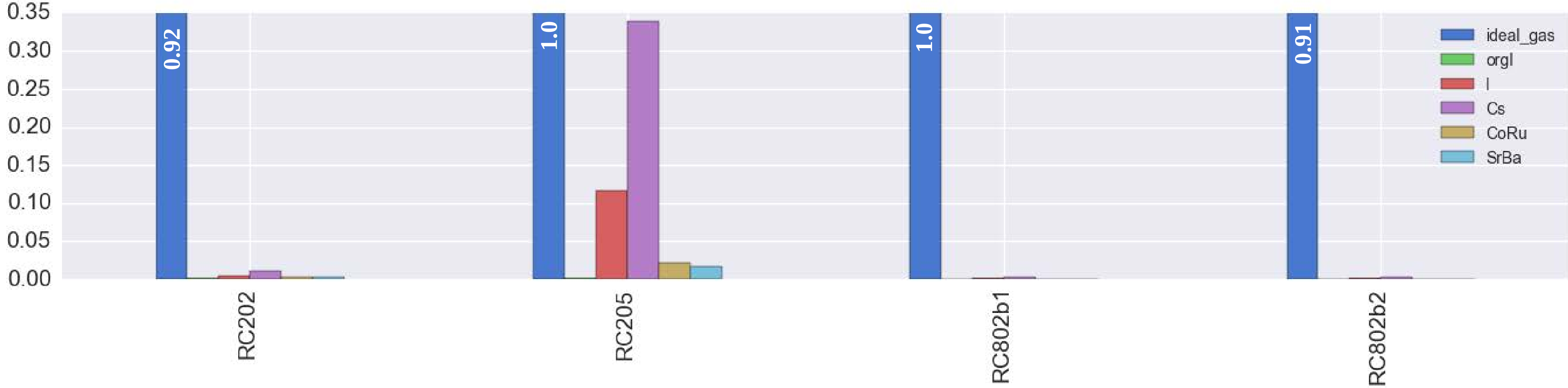


Figure 1. Early release category release fractions, as a fraction of core inventory, for each of the nuclide groups.

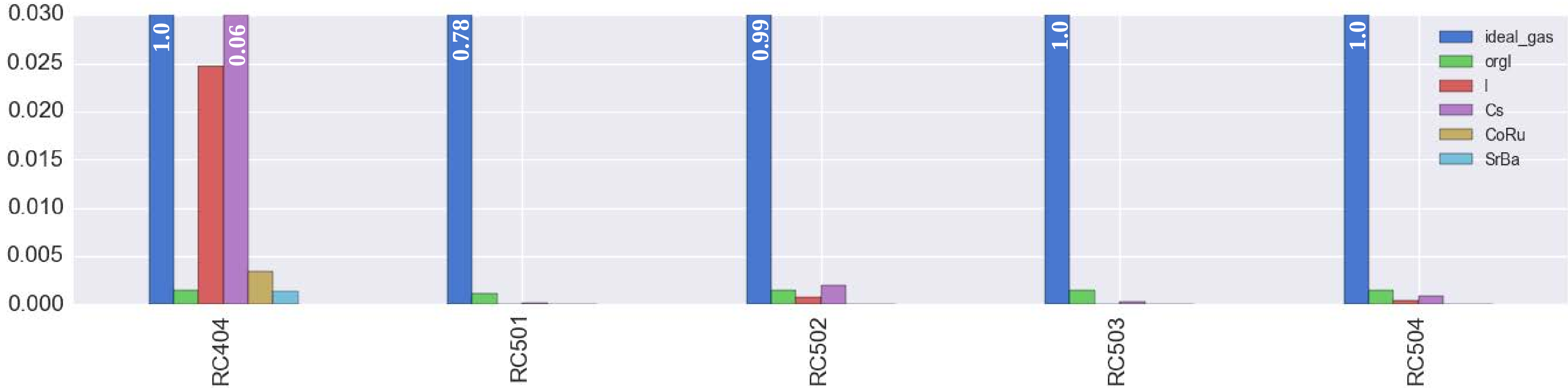


Figure 2. Late release category release fractions, as a fraction of core inventory, for each of the nuclide groups.

3. Results & Analysis

This section provides an overview of the results for individual health effects, societal health effects, environmental effects, and economic effects as performed in the Swedish pilot study.

3.1. Individual health effects

The results of LENA calculations are dose, dose rate, and contamination levels at distances downwind of the release location. The included exposure pathways are groundshine, cloudshine and inhalation doses which are summed to determine a total whole-body dose. The integration time of the dose calculations are 48 hours, starting from the time the release starts. Note, the delay time mentioned in Table 5, i.e. the time between the end of chain reaction in the core and the release time is before prior to this 48 hour integration time. The dose, dose rate and contamination levels are calculated for each of the weather inputs (hourly data collected over 2 years).

The maximum doses, which are assessed in the case of individual health effects, are experienced along the centerline of the plume. In order to estimate the collective dose and the contamination for each weather case, doses and deposition off of the centerline are calculated by assuming a Gaussian distribution away from the centerline.

The maximum individual exposures are determined within 10 km and outside of 10 km. This distinction allows for a coarse understanding of how the maximum individual doses could be reduced given perfect evacuation.

3.1.1. Maximum individual dose at inside of 10 km and outside of 10 km

The maximum individual dose is extracted from doses calculated on the centerline of the plume, for each weather condition, at 48 hours after beginning of release.

Figure 3 and Figure 4 show boxplots of the maximum exposures calculated for each of the weather conditions used, for early releases and late releases respectively.

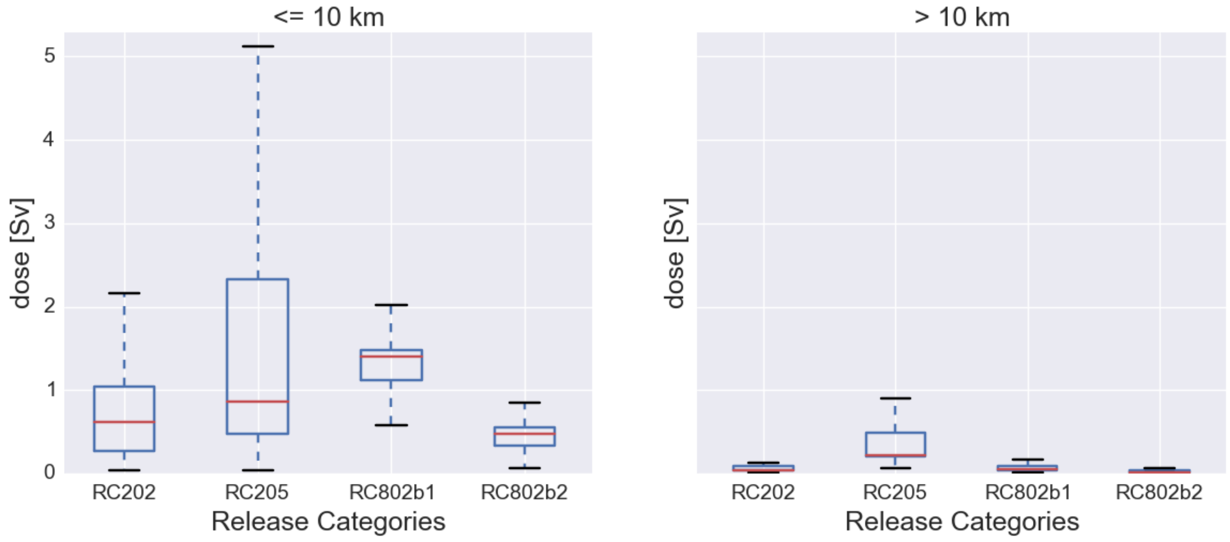


Figure 3. Boxplot showing the maximum 48 hour exposures within 10 km and outside of 10 km for early releases (less than 10 hours)

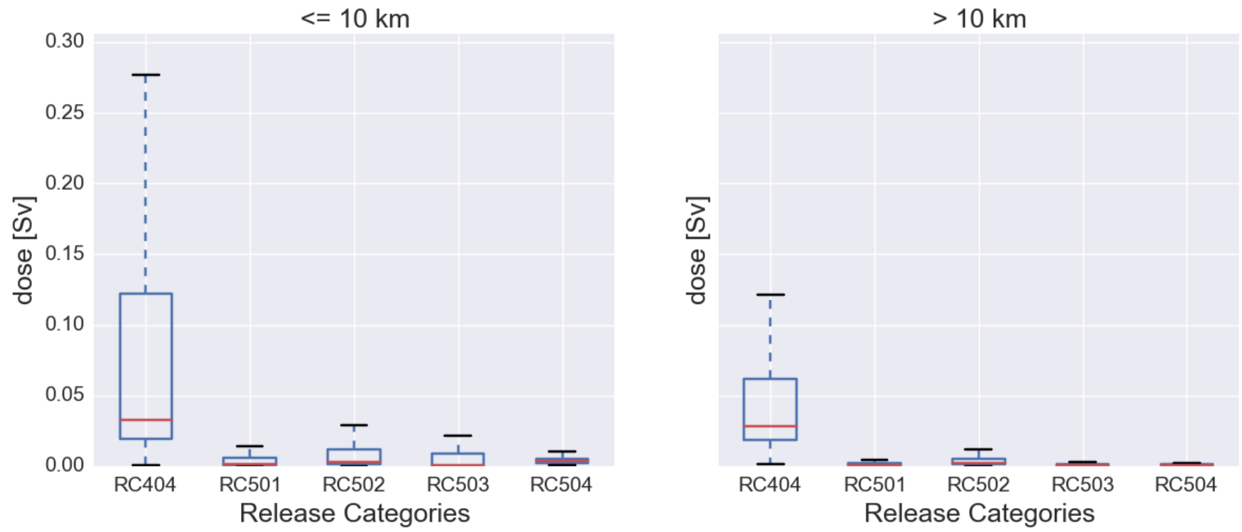


Figure 4. Boxplot showing the maximum 48 hour exposures within 10 km and outside of 10 km for late releases (greater than 10 hours).

Figure 3 and Figure 4 provide a graphical idea of the comparison of values inside and outside of 10 km. Table 6 provides a numerical comparison of the median values (shown in red in each of the boxplots). The delay in hours is provided for each of the release categories to provide perspective on the feasibility of "perfect" evacuation.

Table 6. The maximum dose was calculated for each weather case. This table shows the median values overall of the weather conditions (which was represented with the red line in the boxplots shown in Figure 3 and Figure 4).

Release Category	Median Whole Body Dose <=10 km [Sv]	Median Whole Body Dose > 10 km [Sv]	Difference [Sv]	Difference [%]	Delay Hours
RC202	6.14E-01	4.17E-02	5.72E-01	93%	4.6
RC205	8.63E-01	2.19E-01	6.44E-01	75%	3.5
RC404	3.24E-02	2.83E-02	4.13E-03	13%	7.8
RC501	1.37E-03	9.82E-04	3.83E-04	28%	60
RC502	2.67E-03	2.21E-03	4.56E-04	17%	60
RC503	8.22E-04	4.75E-04	3.47E-04	42%	85
RC504	3.30E-03	8.75E-04	2.43E-03	74%	85
RC802b1	1.40E+00	4.86E-02	1.35E+00	96%	1.8
RC802b2	4.69E-01	1.65E-02	4.52E-01	96%	7.4

It is of interest to also look at the contributions to the dose from each of the three exposure pathways, which sum to the whole body doses. These are shown in Figure 5 for early failures and Figure 6 for late failures.

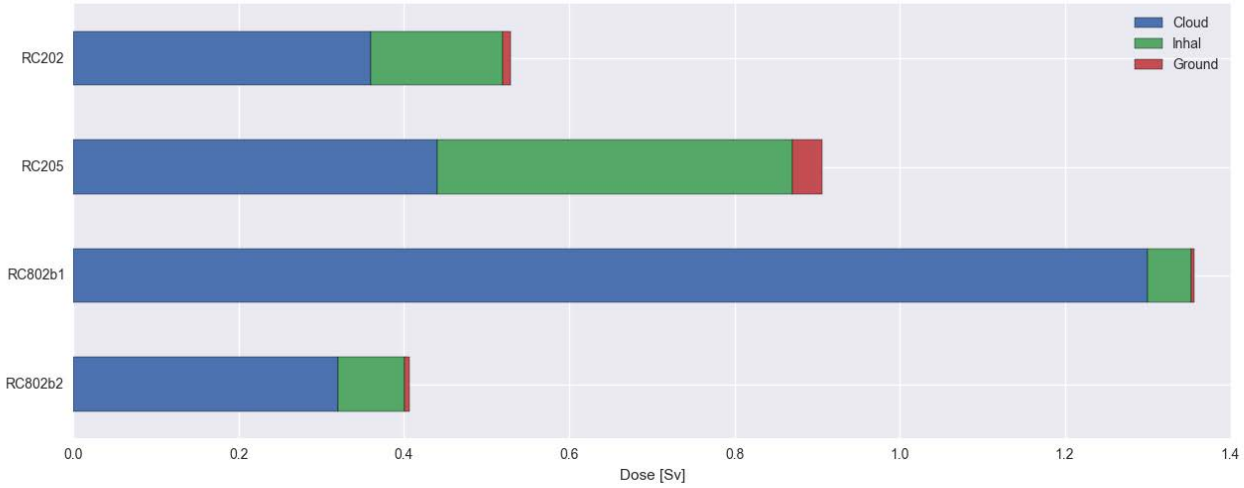


Figure 5. Median values for dose from cloudshine, inhalation, and groundshine to most exposed individual for early release categories.

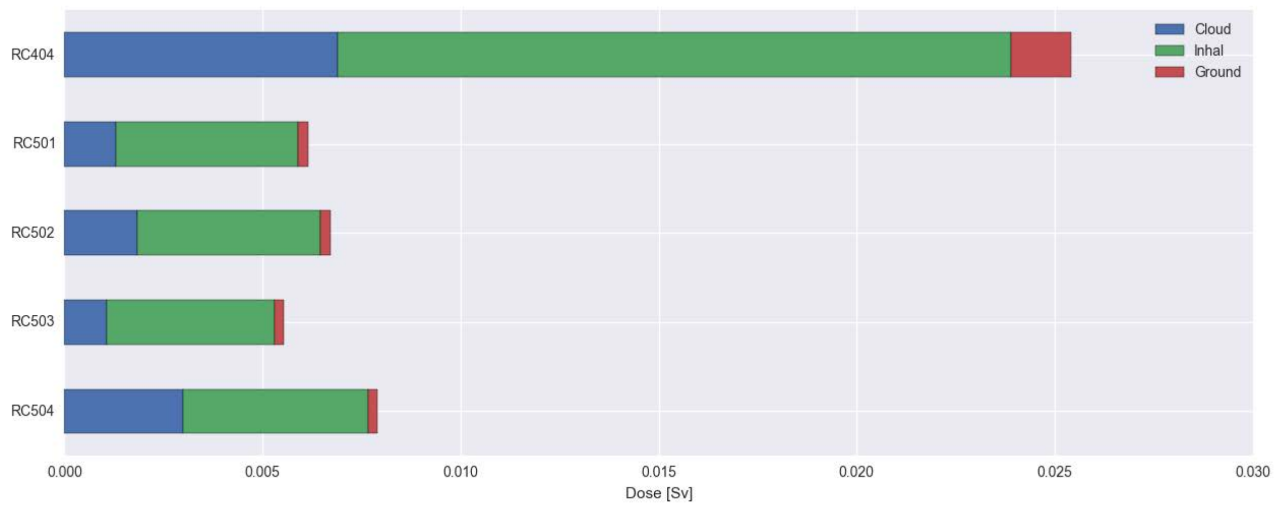


Figure 6. Median values for dose from cloudshine, inhalation, and groundshine to most exposed individual for late release categories.

Figure 3 and Figure 5 provide a description of early release risks to the maximum possible exposure. The important factors for each of the release categories are essentially (1) how much is released, (2) when the release starts, and (3) how long the release is.

First we see from Figure 1 that each of the releases has nearly all noble gases released. RC202 and RC205 have significant quantities of each of the nuclide groups released, while RC802b1 and b2 have significantly smaller releases of each of the other nuclide groups. The duration of release for each of the release categories is quite different, as shown in Table 5, where we find that RC802b1 and b2 have significantly shorter releases.

Short warning time with a large noble gas release fraction (i.e. early release) as for RC802b1, causes cloudshine from the noble gases to have a significant contribution to the dose. An early release start means that the relatively short-lived noble gases have not had time to decay. The actual release in Bq, for a 100% release fraction of noble gases will therefore depend heavily on the time the release starts.

Noble gases contribute primarily to the cloudshine exposure pathway since they are largely unaffected by rain washout, do not deposit on ground and they are easily exhaled, ie no significant contribution to the inhalation dose. Note, this makes the dose from noble gases less dependent on the different weather cases. As a result RC802 has a significantly tighter dose distribution than RC202 and RC205 which have an assortment of released nuclides in their release.

A take-away from these results is how important release timing is. RC205 is a much larger release than any of the other releases, but the very short time spans for RC802 lead to very high doses from ideal gases near the plant. Further away from the plant is a different story as cloudshine from ideal gases is less of a factor away from the release point because of the diffusion of the plume.

Turning our attention to the late releases, we see from Figure 6 that inhalation doses play a more important role for the maximum exposures. Since a large amount of the noble gas inventory has decayed, other nuclide groups will be more important. The late releases in this analysis are very long, which means that a person would be in the plume, ie inhalation pathway important, during a major part of the dose integration time of 48 hours.

Groundshine does not play a major role in the first 48 hours of exposure for any of the release categories.

3.1.2. Risk of death for maximum exposed individual

From the exposures described in the previous section, the risk of death for the median weather cases is calculated using the methodology described in the Methodology Specification report [3]. Since the risk of early death requires significantly high doses, at least approaching 3 Sv, the median weather of the maximum individual doses for all weather cases are unlikely to produce significant risk of early death for any of the release categories aside from those that produced the highest doses, as shown in Table 11. The only release category that has appreciable risk outside of 10km is RC205. RC802b1 and b2 have sizeable exposure within 10km because of release of noble gasses. These drop off significantly after a short distance because the cloudshine dissipates, and thus are very unlikely to cause acute health effects outside a few km of the release.

Table 7. The risk of early death for an individual exposed to a dose corresponding to the median of the maximum individual doses after 48 hours from start of release.

Release Category	Median risk of early death after 48 hours of release		Difference		RC Delay
	<= 10 km	> 10 km			
	[fraction]	[fraction]	[Fraction]	[%]	Hours
RC202	8.94E-08	0.00E+00	8.94E-08	100%	4.6
RC205	2.69E-06	2.98E-12	2.69E-06	99.9999%	3.5
RC404	0.00E+00	0.00E+00	0.00E+00	-	7.8
RC501	0.00E+00	0.00E+00	0.00E+00	-	60
RC502	0.00E+00	0.00E+00	0.00E+00	-	60
RC503	0.00E+00	0.00E+00	0.00E+00	-	85
RC504	0.00E+00	0.00E+00	0.00E+00	-	85
RC802b1	3.37E-04	0.00E+00	3.37E-04	100%	1.8
RC802b2	6.03E-09	0.00E+00	6.03E-09	100%	7.4

3.2. Collective doses

Collective doses were calculated with 7 days of continuous exposure with no mitigative measures and no shielding. Unlike the individual health effects the median values were not the basis of the analysis, rather the mean values were assessed. For each weather case a collective dose values is calculated in units of person-Sv. The average value over all weather cases for each of the release categories is provided in Table 11.

Table 8. Mean collective doses for population within 50 km and for population within 100 km.

Release Category	Mean whole body collective dose (person-Sv)	
	<= 50 km	<= 100 km
	[person-Sv]	[person-Sv]
RC202	3.18E+02	1.09E+03
RC205	2.51E+03	9.52E+03
RC404	4.78E+02	1.96E+03
RC501	1.63E+01	6.64E+01
RC502	3.68E+01	1.52E+02
RC503	1.46E+01	6.03E+01
RC504	2.24E+01	8.59E+01
RC802b1	1.91E+02	5.41E+02
RC802b2	8.13E+01	2.60E+02

The collective doses calculated for all of the weather cases are shown in Figure 7 and Figure 8 for the population within 50 km and the population within 100km. Since an additional parameter, population density, is introduced in the calculations the variability in the results is greater for the collective dose measurements compared to the individual health risks quantified previously. The collective dose calculations are not direction independent. Instead, if the wind carries the plume away from populated areas (e.g. seaward), then the collective doses can be significantly lower than if the wind direction is toward a heavily populated area.

The additional 50 km radius has a notable impact on the collective doses because of the large populations just beyond the 50 km radius. These populations typically don't receive nearly the doses that those within 50 km can potentially receive, however under the right circumstances (e.g. if the wind direction is blowing toward the area of highest population and the plume is relatively narrow) the upper bounds of the collective doses can be quite high as shown in the figures.

Unlike the individual doses the 7 day exposure time allows for even the late release plumes to mostly pass through the 100 km area. The ideal gas releases which were a major contributor to the maximum exposures in section 3.1 do not have as significant effect in the collective dose calculations because these areas are quite sparsely populated.

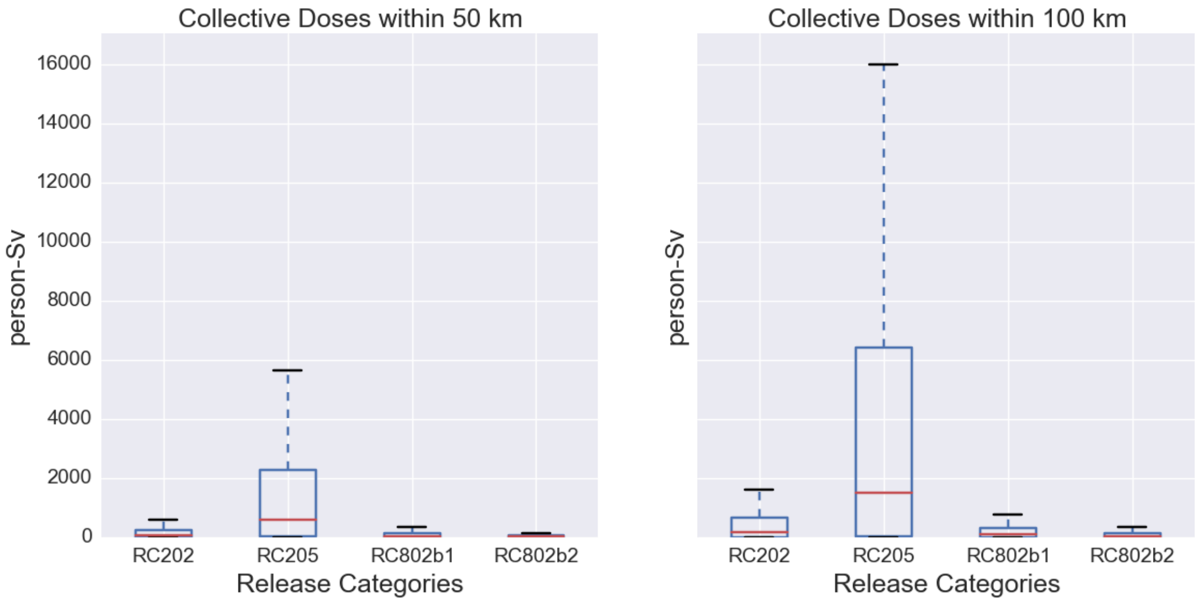


Figure 7. Collective doses for early releases.

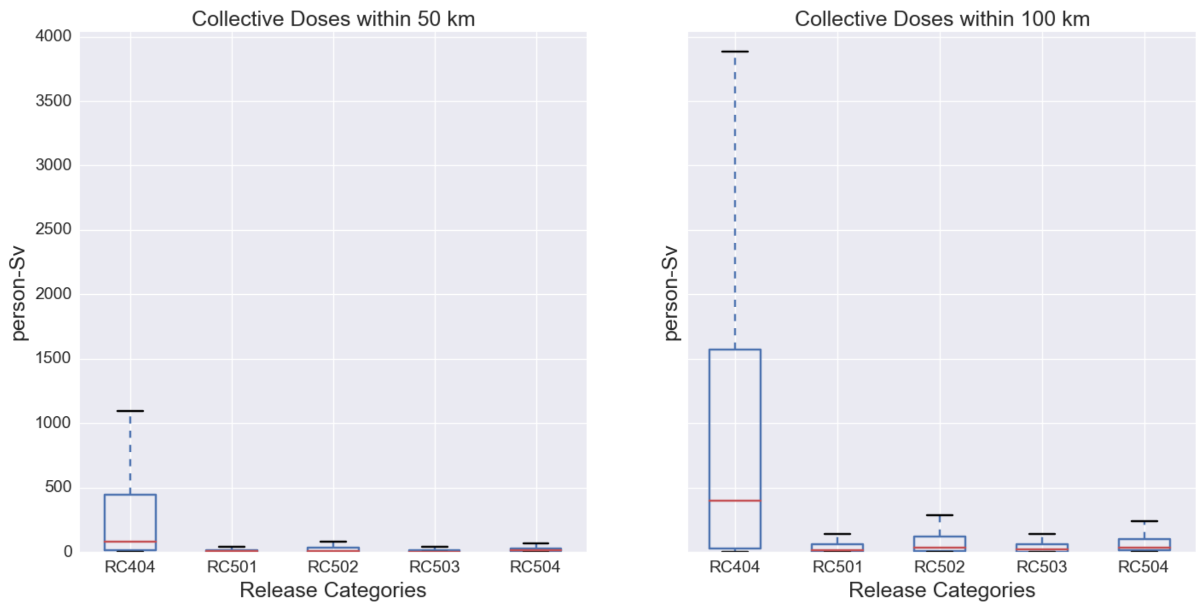


Figure 8. Collective doses for late releases.

The boxplot representations that have been used thus far provide a good picture of the magnitude and the impact of the possible weather cases. These figures do not provide perspective on the potential impact of release and the relative frequency of a release. Exceedance frequencies are often used in consequence analysis. The exceedance curves were calculated for the three largest release categories with respect to collective dose values, shown in Figure 9. In this figure we see that the maximum impact from RC205 is significantly greater than RC404, however the probability of RC404 is significantly higher. Therefore, it is more likely that RC404 causes “low” collective doses as compared to RC205.

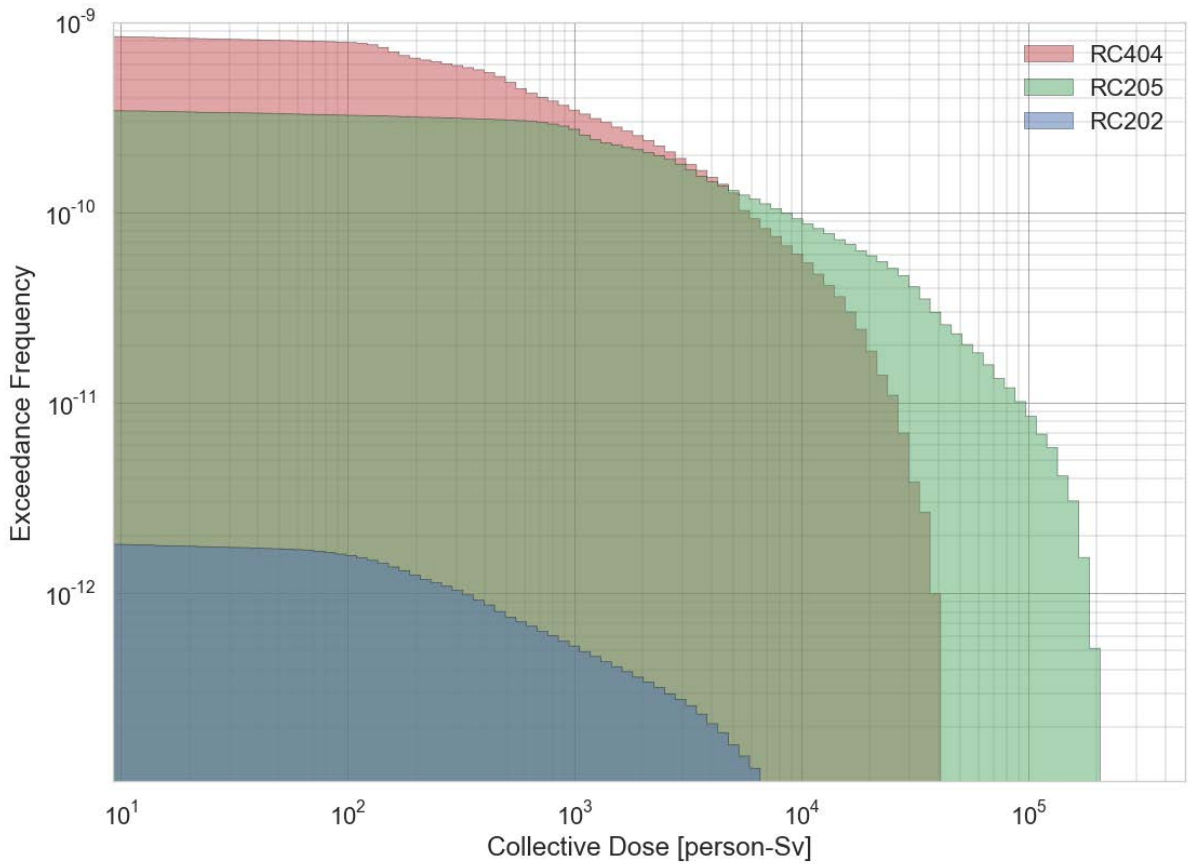


Figure 9. Collective dose exceedance curves for RC404, RC205, and RC202.

3.3. Late health effects

In this study a very simplified method is used based on qualitative findings from Chernobyl survivors. Roughly 100 mSv exposure lead to a 0.5% increase in developing cancer, as described in the methodology specification report [3]. For this analysis a slightly more conservative figure is used of 1% increased risk.

It must be noted that the late health effects analysis is performed independently of the early health effects analysis. Therefore, the possibility for “double counting” the population that experiences early health effects is possible in this analysis. Since most of the release categories do not expose significant populations to large doses this should have a very minor effect on over estimating the number of cancers.

The average lethal cancers for each release category for all of the weather cases are shown in Table 9.

Table 9. Mean lethal cancers after 7 days exposure without shielding or evacuation.

Release Category	Mean lethal cancers after 7 days of exposure without shielding or evacuation	
	<= 50 km	<= 100 km
	Lethal Cancers	Lethal Cancers
RC202	31.8	109.4
RC205	251.5	952.0
RC404	47.8	195.9
RC501	1.6	6.6
RC502	3.7	15.2
RC503	1.5	6.0
RC504	2.2	8.6
RC802b1	19.1	54.1
RC802b2	8.1	26.0

Similar to how the latent cancer risk results are presented in the SOARCA study, Table 10 presents the probability for cancer death, which is calculated from the mean collective dose values normalized by the population exposed [8]. The columns to the right of the table show the risk including the frequency of a given release category. SOARCA chose to present results in this way because the values could be easily compared to the general cancer risk of approximately 2.0e-03 per year in the United States [8].

Table 10. The mean collective dose for each release category is assumed to translate to a 10% cancer risk per person-Sv. Given the population these are translated into cancer risks within 50 km and within 100 km. In the two columns on the far right these risks a

	Frequency	Mean conditional probability of latent cancer fatality due to release exposure		Release category risk of latent cancer fatality due to release exposure	
RC	[/y]	50km	100km	50km	100km
RC202	2.60E-12	1.93E-04	7.90E-05	5.02E-16	2.05E-16
RC205	4.51E-10	1.53E-03	6.86E-04	6.88E-13	3.09E-13
RC404	1.09E-09	2.90E-04	1.41E-04	3.16E-13	1.54E-13
RC501	6.51E-13	1.00E-05	5.00E-06	6.51E-18	3.26E-18
RC502	3.96E-11	2.20E-05	1.10E-05	8.71E-16	4.36E-16
RC503	1.27E-09	9.00E-06	4.00E-06	1.14E-14	5.08E-15
RC504	3.29E-08	1.40E-05	6.00E-06	4.61E-13	1.97E-13
RC802b1	3.83E-09	1.16E-04	3.90E-05	4.44E-13	1.49E-13
RC802b2	3.83E-09	4.90E-05	1.90E-05	1.88E-13	7.28E-14

In the SOARCA study results were that a majority of the dose that was received was very low doses from populations upon return to low-contamination areas. This type of consideration cannot be easily handled with the methods in the study and was determined as beyond the scope of this study.

3.4. Environmental

Environmental effects from a nuclear accident can be numerous from health effects to plants and animals, ground and surface water contaminations or condemnation, or affects to agricultural land. The environmental impact assessed in this study is greatly simplified. The impact is assessed by quantifying the land areas that exceed a Cesium-137 contamination of 100 kBq/m^2 and 1000 kBq/m^2.

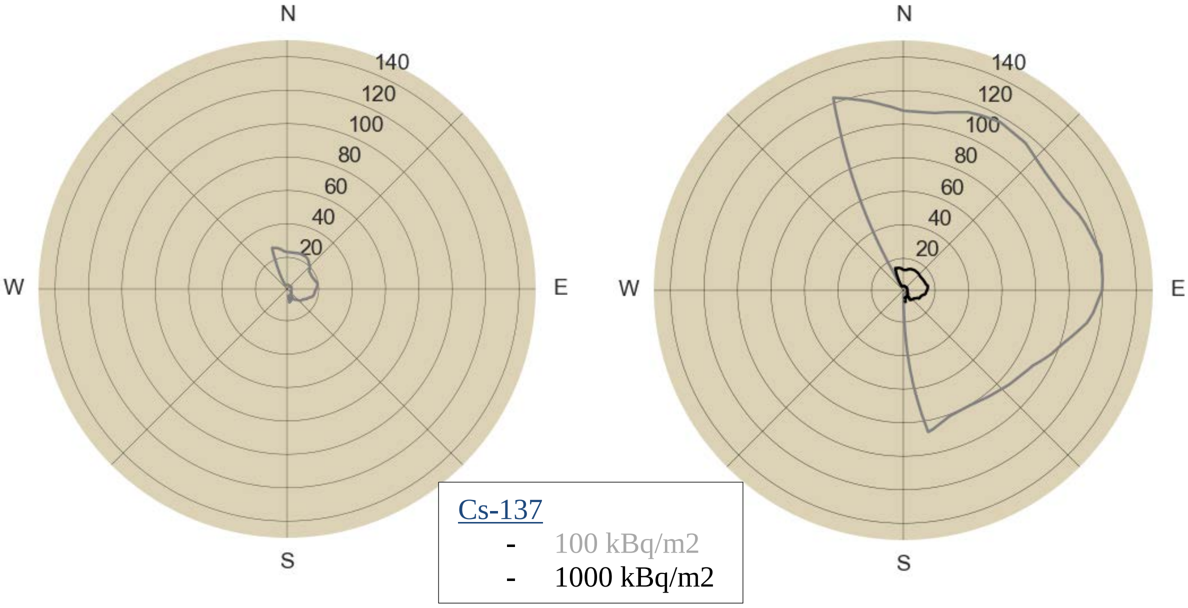
These threshold calculations are only performed for the Cs-137 deposition on land. So all water deposition is ignored, and water subsequently contaminating land is not included in this study.

The calculations were performed for each of the weather conditions and a subsequent value is derived. The left two columns of Table 11 show the average land areas that are over the stated threshold values across all of the weather conditions. The values in the right two columns represent the land areas exceeding the threshold values if each weather case is combined in a weighted sum, where each weather case is equally probable. The weighted sums provide lower values for the cases where little land area is highly contaminated and larger values for the release categories that have significant amount of land contaminated above the lower threshold of 100 kBq/m^2.

Table 11. Summary table of environmental impact for each release category.

	Mean km^2 of land area exceeding cesium contamination levels calculated for each weather case		Probability weighted sum of all weather cases - km^2 of land area exceeding cesium contamination levels	
RC	100 kBq/m^2	1000 kBq/m^2	100 kBq/m^2	1000 kBq/m^2
RC202	4.14E+02	6.36E+00	1.29E+01	0.00E+00
RC205	8.79E+03	1.81E+03	1.89E+04	2.46E+02
RC404	2.96E+03	4.64E+01	5.06E+02	0.00E+00
RC501	2.70E-05	0.00E+00	0.00E+00	0.00E+00
RC502	5.39E+00	0.00E+00	0.00E+00	0.00E+00
RC503	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RC504	1.58E-01	0.00E+00	0.00E+00	0.00E+00
RC802b1	7.34E+01	2.74E+00	0.00E+00	0.00E+00
RC802b2	6.75E+01	3.44E+00	0.00E+00	0.00E+00

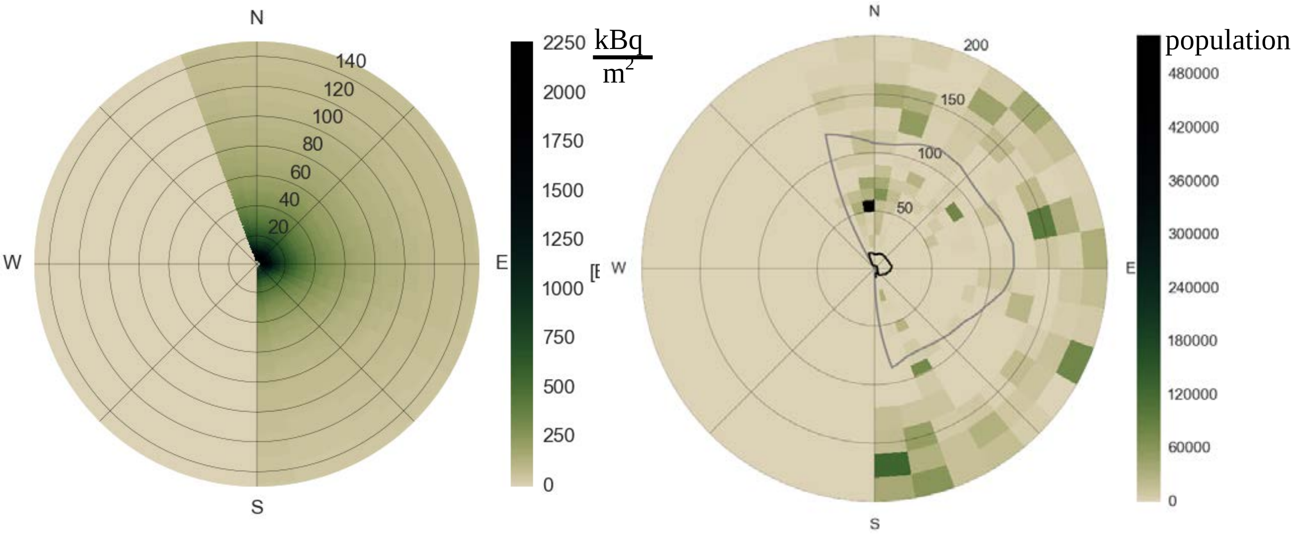
The weighted sum calculations can also provide an indication of the expected direction of contamination. The only two release categories that had appreciable weighted sums were RC404 and RC205, which are shown in Figure 10. This figure illustrates an important point. Both of these releases greatly exceed the 100 TBq threshold that was used to develop the scope of analysis, however, RC205, the early RC category, has significantly greater potential to cause wide-scale contamination.



(a) (b)

Figure 10. (a) Weighted sum Cs-137 contamination of all weather conditions for RC404. (b) Weighted sum of Cs-137 contamination for all weather condition for RC205.

Figure 11 (a) further shows the weighted sum of contamination for RC205. Figure 11 (b) shows the same iso-lines as Figure 10 (b), but with the added context of the population. The population center to the north of the release point is clearly inside of the expected 100 kBq/m^2 zone. This type of simple analysis of contamination and population is the basis of the economic analysis described in section 3.5.



(a) (b)

Figure 11. (a) Weighted sum Cs-137 contamination of all weather conditions for RC 205. (b) Population distribution with isolines for 100 kBq/m^2 (grey) and 1000 kBq/m^2 (black)

3.5. Economic

The methodology used in the simplified economics analysis is described in [3]. This methodology is based on the Cs-137 contamination that was performed for the environmental assessment.

The time period it takes for Cs-137 and Cs-134 to decay to 1μSv/h was derived based on the amount of Cs-137 deposition. The ratio of Cs-134 to Cs-137 was calculated based on the UK-EPR Pre-construction safety report release categories to be a value of approximately 1.5 for all release categories. The correlation is shown in Figure 12. The compensation cost for a release is then estimated by assuming that 1.0E+06 SEK per displaced individual per year of displacement.

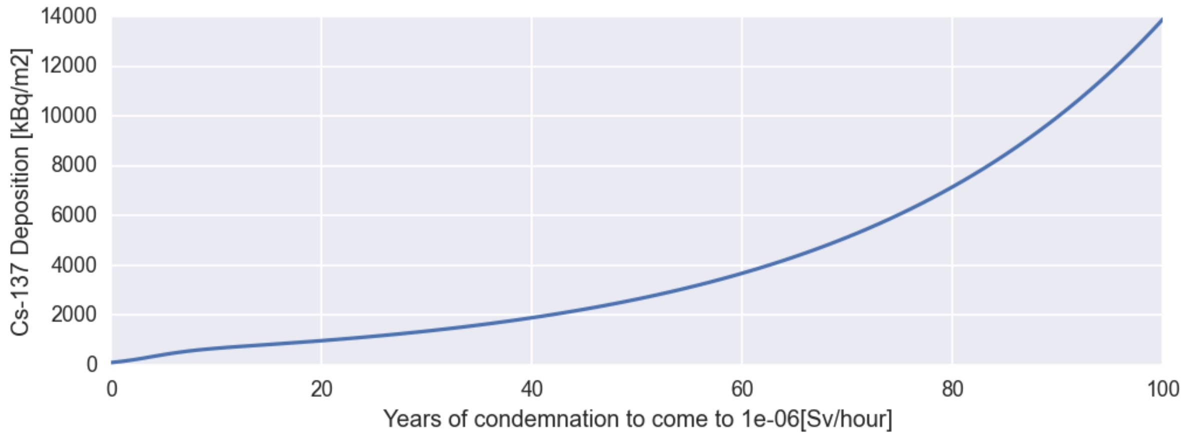


Figure 12. Correlation between years of condemnation and Cs-137 deposition.

When these assumptions are applied to the cesium deposition and populations in this project, economic displacement compensation results are evaluated for each of the weather cases. The results are shown in a boxplot in Figure 13.

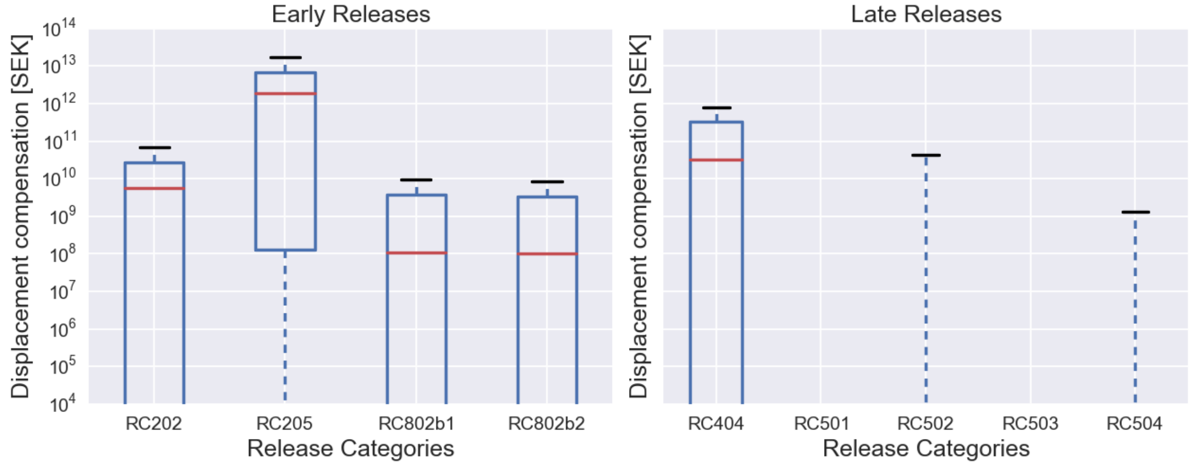


Figure 13. Boxplot comparing release categories displacement compensation for all weather cases. Note that this boxplot has a logarithmic y-axis.

Some interesting trends are shown in this economics analysis. The displacement costs paid by TEPCO as of January 8, 2016 totaled approximately 4.0e+12 SEK [9], which is within the same range as the median value for RC205, but significantly larger than the rest of the other Release Categories. The late releases tend to have very little impact with the exception of edge-cases for the releases that do not have containment spray. The late releases where containment spray is functioning (RC501 and RC503) do not cause substantial contamination of populated areas.

The frequency component of a release is not provided in the boxplot representation of results. The exceedance frequency for the range of displacement compensation costs is provided for RC404 and RC205 in Figure 14.

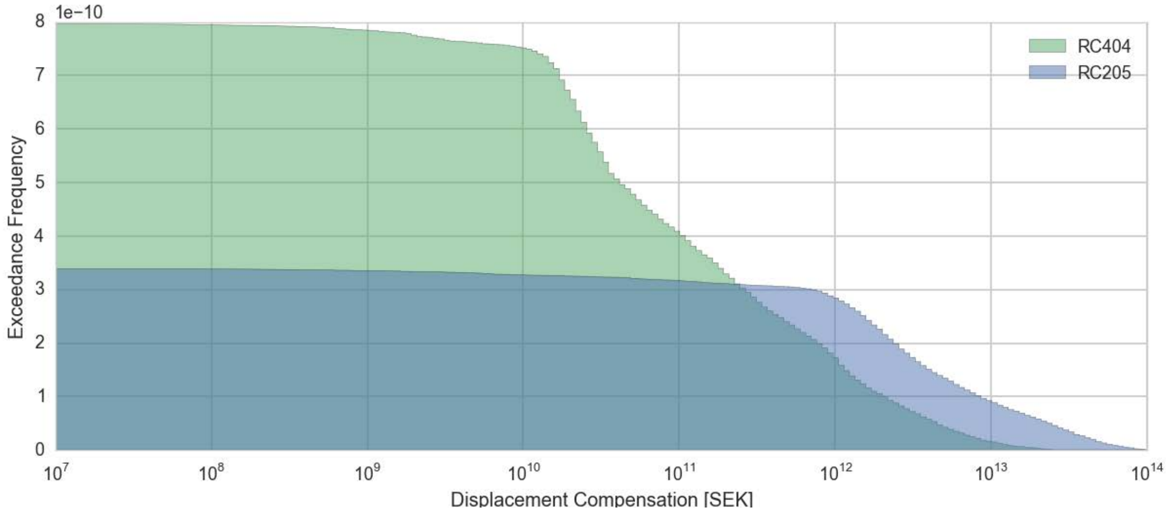


Figure 14. Displacement compensation costs for RC404 and RC205. Please note that the y-axis is linear while the x-axis is logarithmic

Consistent with the contamination analysis where results were calculated for each individual weather case and the weighted sums of all of the weather cases, economic impacts were determined from the weighted sums of the contamination values. Unlike, the environmental analysis these are very consistent with the values shown in Figure 13 for RC 202, 205, and 404.

Table 12. Estimated displacement compensation costs from the contamination values from the weighted sums for each release category.

Release Category	Estimated costs from the weighted sums of all weather cases	
	Estimated Cost (SEK)	Scientific Notation (SEK)
RC202	93 000 000 kr	9.30E+07
RC205	4 330 000 000 000 kr	4.33E+12
RC404	25 900 000 000 kr	2.59E+10
RC501	- kr	0.00E+00
RC502	- kr	0.00E+00
RC503	- kr	0.00E+00
RC504	- kr	0.00E+00
RC802b1	- kr	0.00E+00
RC802b2	- kr	0.00E+00

4. Uncertainty and sensitivity

One of the major shortcomings in this study was that a robust uncertainty and sensitivity analysis was not performed. Source term uncertainty would have required better knowledge of the Level 2 PSA uncertainties. A review of the UK-EPR PCSR uncertainty analysis of Level 2 PSA is discussed in 4.1. Since uncertainty and sensitivity was a major goal of the project, a review of the uncertainty analysis performed in the SOARCA study is discussed in section 4.2.

4.1. UK-EPR PCSR uncertainty analysis

An analysis of the frequency uncertainties was performed in the UK-EPR PCSR [4], however, this was performed for the combined Large Release Frequency (LRF) and Large Early Release Frequency (LERF). In the UK-EPR study there was no uncertainty analysis performed for release parameters (e.g. nuclide release fractions, timings, etc.). It should be noted that this is not a critique of the analysis. This is a common limitation in Level 2 PSA studies.

The uncertainty analysis presented in [4] does show some interesting relationships. Point estimates were routinely above mean, and significantly above median frequency reported. This indicates some level of conservatism was used when developing the analysis, and there is likely conservatism in the release category frequencies used in this study. The uncertainty analysis shows that both LERF and LRF span approximately 1 Order of magnitude frequency impact as shown in

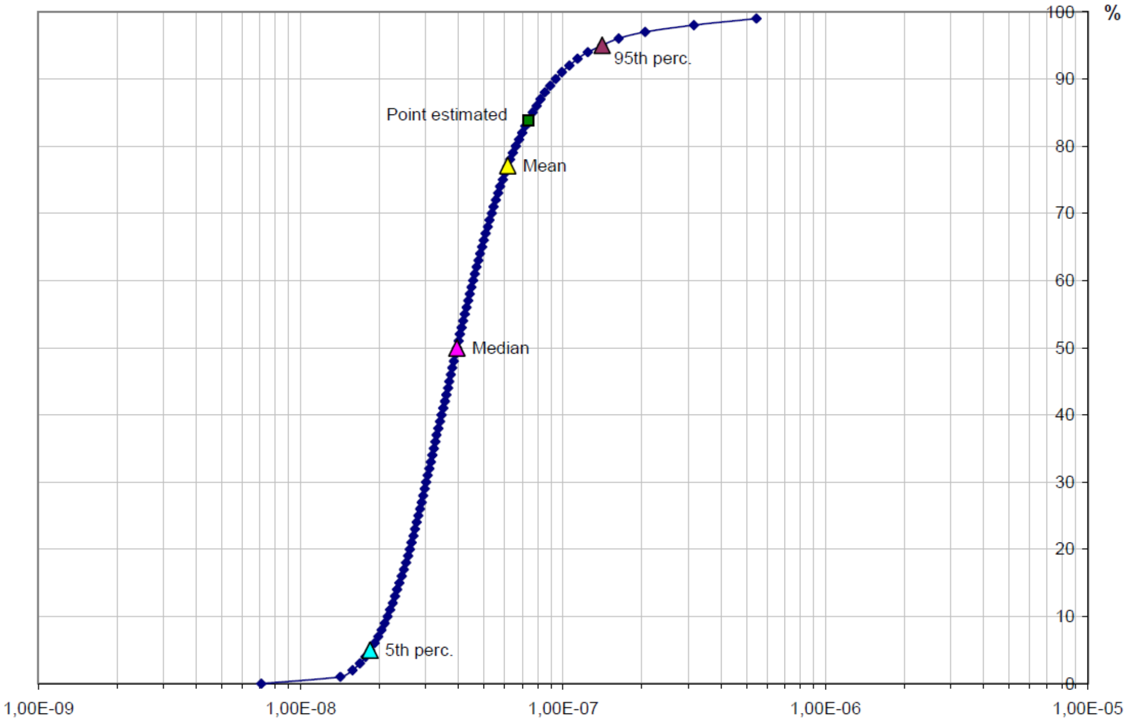


Figure 15. Uncertainty analysis performed in UK-EPR PCSR on Large Release Frequency [3].

4.2. SOARCA uncertainty study

A very large study of the uncertainty and sensitivity is performed for the long term station blackout case in the SOARCA study. The study and its findings have been published in a document NUREG 7155, which is still in the draft state at the time of writing this report [10]. The study is perhaps the most recent, complete uncertainty study for Level 3 PSA-type analysis. The study is extremely extensive and includes the parametric uncertainty analysis of many Level 2 and Level 3 PSA models and phenomena. The study does not investigate uncertainty in the frequency of source terms, nor the differences in atmospheric dispersion models. The study does investigate the sensitivity to things such as safety relief valve (SRV) closure times, health effects models, deposition velocities, etc.

The analysis showed very many interesting results. Among the findings were the following results: Regression analysis shows Level 2 phenomena account for up to1/3 of variance in magnitude of acute and latent health effects. Among the Level 3 PSA parameters analyzed, wet deposition had very significant impact for early health effects, while, dry deposition had very significant for latent health effects.

Weather variability was also analyzed and seemed quite consistent with Swedish pilot study. This is possibly due to the similar fidelity in input weather data, range of distances used, and simplified dispersion modelling.

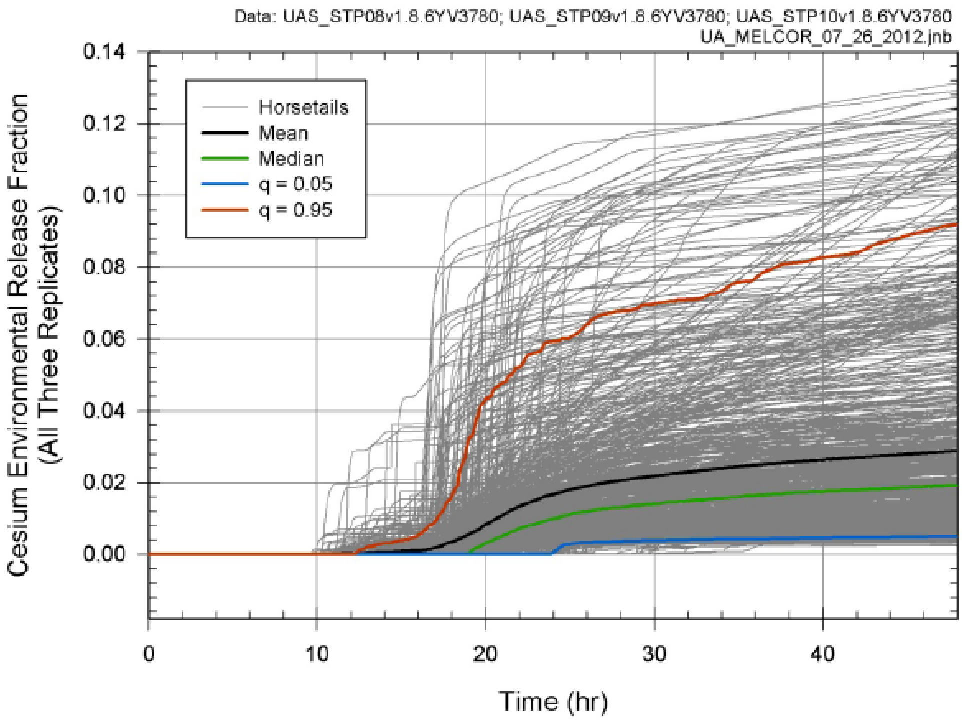


Figure 16. Figure from USNRC SOARCA uncertainty study [10] .

The range of results for latent cancer fatalities found in the SOARCA uncertainty analysis is shown in Figure 17. This shows that as the radius is increased the average risk of cancer decreases, which is logical because as one includes more distant populations they are likely to receive smaller doses reducing the average likelihood of latent effects. The range of values from the Monte Carlo analyses also tend to decrease at greater distance intervals for much the same reason.

Table 6.2-2 Conditional, mean, individual LCF risk (per event) average statistics for the MACCS2 Uncertainty Analysis for five circular areas (using all three source term replicates)

	0-10 miles	0-20 miles	0-30 miles	0-40 miles	0-50 miles
Mean	$1.7\text{x}10^{-4}$	$2.8\text{x}10^{-4}$	$2.0\text{x}10\text{-}^{4}$	$1.3\text{x}10^{-4}$	$1.0\text{x}10^{-4}$
Median	$1.3\text{x}10^{-4}$	$1.9\text{x}10^{-4}$	$1.3\text{x}10^{-4}$	$8.7\text{x}10^{-5}$	$7.1\text{x}10^{-5}$
5^{th} percentile	$3.1\text{x}10^{-5}$	$4.9\text{x}10^{-5}$	$3.4\text{x}10^{-5}$	$2.2\text{x}10^{-5}$	$1.9\text{x}10^{-5}$
95^{th} percentile	$4.2\text{x}10^{-4}$	$7.7\text{x}10^{-4}$	$5.3\text{x}10^{-4}$	$3.4\text{x}10^{-4}$	$2.7\text{x}10^{-4}$

Figure 17. Table from SOARCA uncertainty analysis [10].

5. Conclusions

The Swedish Pilot study did look at a very wide range of Level 3 PSA metrics for health effects, environmental effects, and even economic effects. Looking at different metrics highlighted how different elements of the Level 2 PSA analysis or weather input can be important for different metrics. Some of the notable findings are the following:

- A 100 TBq release criteria did provide a reasonably good screening of which release categories were likely to cause health effects. Release categories below 100 TBq were unlikely to cause health effects, while those exceeding 100 TBq had a notable risk of causing health effects when applying very conservative assumptions.
- One of the goals of the study was to investigate what can be said of release categories that fall above or below the threshold. One clear finding, for several of the risk metrics investigated the differences between a release exceeding 100 TBq and the largest releases category (greatly exceeding the 100 TBq thresholds) was very significant. The contamination metrics were unlikely to cause significant effects unless the threshold was greatly exceeded.
- The study performed acute health effects and latent health effects in a very simplified manner. Even with refined models the uncertainties for health effect quantification can be quite large as is shown in the SOARCA uncertainty analysis. For this reason it may be recommended to focus Level 3 PSA studies on dose and contamination, especially in simple studies.

The study was limited compared to some of the expectations at the beginning of the project. Many of the input, modelling and methodology limitations have been expressed in the previous reports. Despite these limitations many new and interesting insights were made as a result of this work. Due to the fact that it was a general study, and that quite simple and limited tools were used a lot of insight was made in the methods and logistics of performing a Level 3 PSA and the calculations that are required.

When developing the results it became apparent that a central limitation of the analysis was due to the generalized input data. These generalizations simplified the methods and in fact made the study manageable despite the limited resources; however, it was difficult to make real-world assertions that would have helped in assessing the utility of the analysis. The use of general source terms from the EPR report provided insight into the organization of the UK EPR Level 2 PSA. It did not allow for the project to develop much needed experience in using or potentially developing release categories based on a Nordic Level 2 PSA.

A complete uncertainty analysis was not performed in the Swedish pilot study. An interesting uncertainty study was performed in the UK-EPR Level 2 PSA, however, this did not provide the necessary information in order to perform a Level 3 PSA uncertainty study. This is due, in part, because the Level 2 PSA uncertainty was presented for the Large Release Frequency and Large Early Release Frequency, but not for individual release categories. As an alternative, the uncertainty analysis form the USNRC SOARCA study was investigated. The SOARCA study was an investigation of a set of pre-determined source terms. Therefore most of the interesting findings are related to parameter magnitudes and phenomenological uncertainties and the impact of various coefficients in the process of calculating core degradation/migration, release, and off-site consequences. The USNRC full scope Level 3 PSA study which should be finalized in the coming years should provide further insight into Level 3 PSA uncertainties, particularly with respect to the frequency uncertainty.

Performing and discussing this study demonstrated some under presented benefits of Level 3 PSA. First, simply by performing a Level 3 PSA study necessitates additional investigation and scrutiny of the Level 2 PSA study. By performing a Level 3 PSA one must take a structured view of the Level 2 PSA study and its results. All-too-often the interest in Level 2 PSA studies lies in the frequency assessment of “Large Releases” or “Large Early Releases”. In this study it was apparent that large releases could have limited or substantial off-site effects where elements such as release timing, release composition, and external conditions can have a substantial impact. Level 3 PSA also provides an interface for the radiological and PSA communities. These groups are addressing similar issues concurrently, both with separate skill-set and insights. Level 3 PSA can serve as a bridge between the radiological analysis and PSA communities which can likely provide other mutual benefits.

There are many places where this study can be expanded. Sensitivity analysis, and the impacts of shielding, and evacuation are essentially fundamental in a Level 3 PSA study, but lacking here due to analysis and resource constraints. Ultimately, this is a “generic” study, and therefore it would be difficult to further develop it, and it is perhaps more useful to develop a Level 3 PSA for an actual application. Many of the questions that still linger would be better answered by a site-specific, reactor specific study where actual Level 2 PSA data is available and must be applied to a Level 3 PSA. The true impact and benefits to a utility, emergency personnel, and the surrounding population are difficult to realize in highly general assessments.

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Level 3 PSA Guidance for Nordic Conditions

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Jan-Erik Holmberg[3]

Ilkka Karanta[4]

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[1]Lloyd's Register

[2]ÅF

[3]Risk Pilot

[4]VTT

[5]Vattenfall

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Table 1 Acronyms and abbreviations used in the document.

ANS	American Nuclear Society
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
CCDF	Complementary Cumulative Distribution function
CDF	Core Damage Frequency
CFD	computational fluid dynamics
DBA	Design basis accident
DID	Defense-in-Depth
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IDPSA	Integrated Deterministic and Probabilistic Safety Analysis
INES	International Nuclear Event Scale
LCA	life cycle analysis
LERF	Large early release frequency
LNT	linear no-threshold
MWth	Megawatts of thermal power
NEA	Nuclear energy agency of OECD
NKS	Nordic Nuclear Safety Research
NPSAG	Nordic PSA Group
NRC	United States Nuclear Regulatory Commission
OECD	Organization for Economic Co-operation and Development
OIL	Operational Intervention Level
PCA	Probabilistic consequence analysis
PSA	Probabilistic safety assessment
RC	Release Category

SAMA	Severe Accident Mitigation Alternative
SOARCA	State-of-the-Art Reactor Consequence Analyses
SSM	Statens Strålsäkerhetsmyndighet, Swedish Radiation Safety Authority
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
WGRISK	OECD/NEA Working Group RISK
YVL	Ydinturvallisuusohjeisto, Finnish guide for nuclear safety

1 Introduction

This report outlines general guidance on the application and element of a Level 3 PSA (Probabilistic Safety Assessment) study. At the time of writing, Level 3 PSA studies are not required in the Nordic countries, therefore, specific procedural Nordic guidance is not provided. Furthermore, it can be concluded that Level 3 PSA is performed relatively infrequently, at least compared to Level 1 and 2 PSA, and how a Level 3 PSA is performed is not as standardized as Level 1 and 2 PSA. This guide gives a brief overview of the state of Level 3 PSA, a description of a spectrum of analyses dependent on the consequences of interest and a discussion of the relevant considerations and recommendations for such analyses.

The project has enabled targeted discussions between consultancies, utilities, regulators, and insurance companies on the subject of Level 3 PSA.

1.1 Background, scope and objectives

Activity in the field of probabilistic off-site consequence analysis has been unsteady over the years. Internationally and within the Nordic countries, there was a large effort in the field of Level 3 PSA in the late 1980s, which included significant Probabilistic Consequence Analysis (PCA) methods work, large scope studies, and IAEA meetings and publications. Generally speaking, there was a significant drop-off in the work performed on Level 3 PSA methods and number of studies performed since the work of the late 1980s and early 1990s. However, several countries have been performing Level 3 PSA consistently for many years (e.g. the Netherlands, South Africa and UK).

Interest in Probabilistic Consequence Analysis (PCA), also referred to as Level 3 Probabilistic Safety Assessment (PSA) has been sparked in recent years by two major factors: the multi-unit severe accidents at Fukushima Daiichi and safety considerations for new nuclear power plant construction. Areas where Level 3 PSA may be useful are risk-informed decision-making, risk assessment and communication to the public. New studies are emerging in the field of probabilistic consequence analysis and many international meetings and workshops regarding public health effects of severe accidents are being arranged. At the time of writing of this Guidance Document standards for both Level 2 PSA and Level 3 PSA are being developed.

As a response to this increased interest, a Nordic research project called "Addressing off-site consequence criteria using Level 3 PSA" (L3PSA) has been conducted since 2013. The project has included the following activities:

- The development of an industrial survey completed by Nordic utilities, Nordic Nuclear Safety Authorities, and Nuclear PSA experts.
- A study of Risk Metrics
- Involvement in IAEA and ASME/ANS Level 3 PSA activities.
- Two parallel Level 3 PSA pilot studies (conducted using Swedish and Finnish probabilistic consequence analysis tools).
- Three project seminars/workshops which have provided valuable conclusions and discussions.

A significant part of the project was research performed in Finland, funded within the SAFIR Framework. Within SAFIR, a unique and significant step in Level 3 PSA has been made by performing a Level 3 PSA pilot study using the IDPSA methodology (Integrated Deterministic and Probabilistic Safety Analysis). The Finnish Pilot study and the Swedish Pilot study have been used when considerations and practical recommendations outlined in Chapter 4 of this report have been formulated.

The objective of the project has been to further develop understanding within the Nordic countries in the field of Level 3 PSA, the scope of its application, its limitations, appropriate risk metrics, and the overall need and requirements for performing a Level 3 PSA.

The final reporting of the project is this Guidance Document that aims to provide clear and applied guidance on Level 3 PSA toward regulators, utilities, and PSA practitioners based on the conclusions made over the course of the work, see Figure 1.

The scope of this guide is limited to radiation effects due to an accident where health consequences due to radiation are accounted for as well as lost land due to contamination. Other aspects, such as psychological health consequences, have been deemed outside the scope of this document.

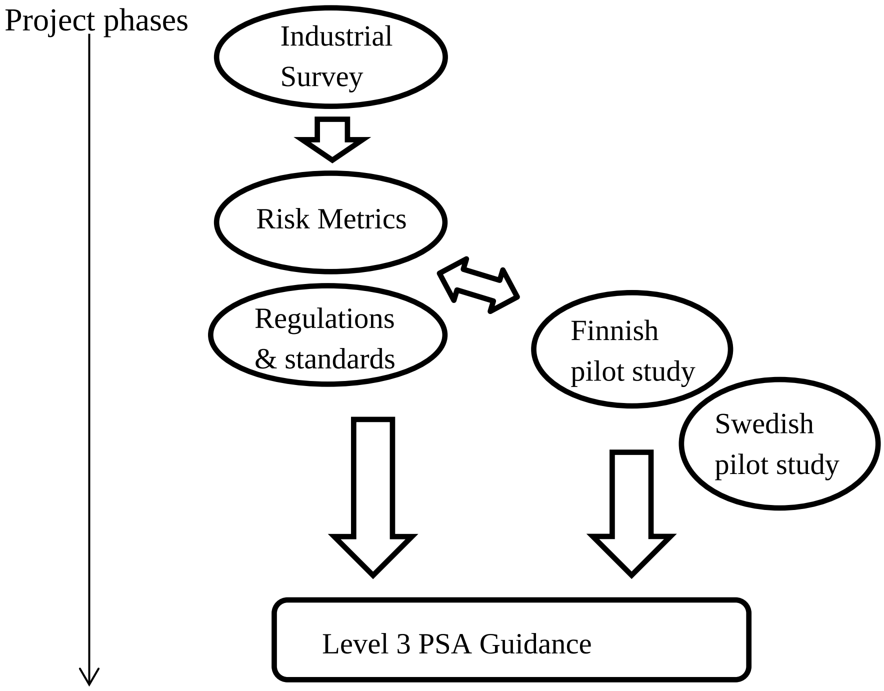

Figure 1 Level 3 PSA project overview.

1.2 The context for PSA Level 3

The Defense-in-Depth (DiD) principle is fundamental for nuclear safety defining above all the principle of multiple barriers to prevent and mitigate nuclear accidents. Today's Level 1 and 2 PSAs cover DiD Levels 1 to 4. In order to fully assess a plant's DiD concept, PSA Level 3 makes the final step by analyzing the fifth level of DiD as illustrated by an event tree in Figure 2.

Success of DiD Level 5 does not necessarily mean that the release will not result in any fatalities or cancer cases, instead it means that the consequence of release to the environment in terms of fatalities and cancers are minimized [13]. The latter sentence touches the objective of traditional Level 3 PSA, namely to illustrate the fatalities and cancers due to a radioactive release to the environment including optimization of the countermeasure strategies to make the radiological consequences as low as reasonable.

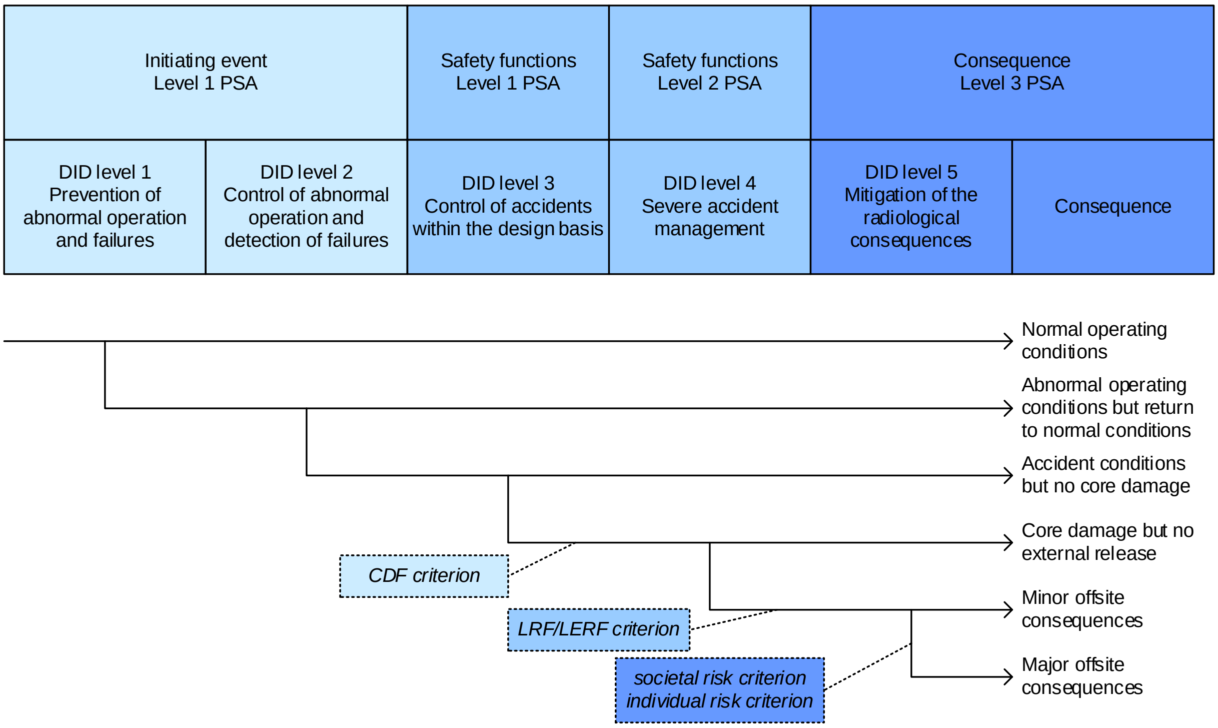

Figure 2 Levels of PSA and Defense-in-Depth (DiD) [12] [13].

1.3 Need for Level 3 PSA guidance

The importance of introducing Nordic Level 3 PSA guidance stems from a number of factors. The knowledge about the analysis as such is relatively immature and the overall purpose, methodology, risk metrics and use of results from a Nordic Level 3 PSA study should ideally be comparable given that the inputs to the Level 3 PSA (i.e. Level 1 and 2 PSA) are of comparable status in terms of level of detail, conservatism, assumptions etc. Without a common understanding of the usability of a consequence analysis, its benefits and limitations there is a risk of divergent Level 3 PSA requirements and studies, making comparison between sites and safety criteria less meaningful. [3]

In addition, since there are so many ways to perform the analyses and evaluate the results guidelines are needed to ensure that scenarios from one plant can be compared with scenarios from another plant.

1.4 Objectives of Guidance Document

The objectives of this Guidance Document are to create preconditions for a common approach on Level 3 PSA, aid the harmonization of Level 3 PSA methodology in the Nordic countries, highlight key considerations, provide understanding of the risks/consequences presented in a Level 3 PSA, and describe the:

- Regulatory framework, Nordic and international
- Value/benefits for Level 3 PSA
- Limitations of Level 3 PSA
- Level 3 PSA common elements and recommendations for Nordic countries

It should be noted that this guide represents general guidance of the elements and considerations for Level 3 PSA.

The intended audience of this guidance is the nuclear industry and authorities in the Nordic countries.

1.5 Outline of Guidance Document

Chapter 1 provides an introductory discussion on the purpose and need for having a guidance document for Level 3 PSA.

Chapter 2 gives an outlook on the regulatory framework, guides and standards in the Nordic countries and internationally.

Chapter 3 discusses expected challenges, limitations and benefits with Level 3 PSA.

Chapter 4 describes the main elements in a Level 3 PSA. In order to achieve a practical guide, the discussion in chapter 4 is based on three cases, which focus on different types of consequences of a nuclear accident. The aim is to cover a spectrum of Level 3 PSA consequences and the pertinent considerations one should be mindful of when performing or reviewing an analysis.

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). 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, SAFIR2018).

1.7 Disclaimer

The conclusions and views expressed in this document remain the responsibility of the author(s) and do not necessarily reflect those of NKS, NPSAG or other funding organizations.

2 Guidelines and regulations

2.1 Nordic regulatory framework

There are no explicit requirements on Level 3 PSA in either Swedish or Finnish regulations. In Finland, a probabilistic risk assessment is generally required in the Nuclear Energy Decree (161/1988) [22]. Guidance on Level 1 and 2 is given in YVL (Finnish guide for nuclear safety) A.7 and general requirements on consequence analysis for the public and the environment are given in YVL C.4. In Sweden Level 1 and 2 requirements are found in SSMFS 2008:1 and guidance on the requirements for deterministic consequence analysis regarding the public and the environment, is found in facility specific SSM orders.

2.1.1 Sweden

As mentioned above, no specific requirements on Level 3 PSA are found in existing Swedish legislation. New regulations are, however, being discussed at the time of writing of this report.

Requirements on radiological consequence assessments after a severe accident are outlined in a governmental decision from 1986;

- Ground deposition which prevent the use of land in the long term should be constrained
- Death due to acute radiation syndromes should not occur

The 1986 decision further states that the requirements are met if the radioactive release is restricted to 0,1 % of the core inventory for Cesium 134 and 137, in a 1800 MWth (megawatts of thermal power) reactor. Depending on core design this corresponds to approximately 150-300 TBq of Cesium. This is referred to as the FILTRA requirements.

For design basis accidents, reference values[1] for realistic off-site consequence assessments are found in site-specific authority decisions, for example in reference [39]. The resulting doses in an assessment should not exceed these reference values. In the range 0-500 m, the effective dose should not exceed 50 mSv[2]. In Table 2, reference values for distances 500-15 000 m from the release point are listed for different accident categories H2, H3 and H4.

[1] Reference value – the value of individual dose due to exposure during the accident or during one year following the accident, without countermeasures.

[2] Dose limit for workers according to SSMFS 2008:51.

Table 2 Reference values in Sweden.

Frequency range of design basis accidents	Source term	Effective dose[1]	Equivalent dose, thyroid[2]
H2 $10^{-2} < f$	0,001xFILTRA	1 mSv	1 mSv
H3 $10^{-4} < f < 10^{-2}$	0,01xFILTRA	10 mSv	10 mSv
H4 $10^{-6} < f < 10^{-4}$	0,1xFILTRA	100 mSv	100 mSv

1 Sum of external radiation from radionuclides in air, internal radiation from inhalation (50 years) and external radiation during 30 days from radionuclides on ground.

2 One-year old child, inhalation.

The authority decision gives guidance on standardized calculations; it outlines considerations for dispersion model (Gaussian), source term (total release time is set to one hour, radioactivity evenly released during this time), release height, plume rise, deposition velocities and meteorological data (depending on release height). It also states that doses should be shown not to exceed the reference values, without any countermeasures taken into account.

In the same decision it is stated that US NRC (United States Nuclear Regulatory Commission) RG 1.183 [35] should be followed altogether in conservative consequence analysis, except for dose conversion factors and treatment of weather in which case the same conditions as for realistic assessments should be used. Doses should be calculated at a distance of 200 m from the release point.

2.1.2 Finland

2.1.2.1 Requirements

Section 3 a, paragraph 22 b of the Government Decree on Nuclear Energy (161/1988) [22] defines the maximum values for the annual dose received by an individual in the population as a result of a release from accidents:

- 1 mSv for Class 1 postulated accidents (frequency less often than once in 100 usage years, but more often than once in a 1000 usage years);
- 5 mSv for Class 2 postulated accidents (frequency less often than once in a 1000 usage years); and
- 20 mSv for a design extension conditions (e.g. a Class 1 accident combined with a common cause failure of a safety system).

Section 3 a of the Government Decree on Nuclear Energy (161/1988) stipulates that at a nuclear power plant the release of radioactive substances arising from a severe accident shall not necessitate large scale protective measures for the population nor any long-term restrictions on the use of extensive areas of land and water.

In order to limit the long term effects, the limit for atmospheric releases of Cesium 137 is 100 TBq. The possibility of exceeding this limit shall be extremely small. The possibility of a release in the early stages of an accident requiring measures to protect the population shall be extremely small.

Relevant to the licensing of new nuclear facilities or significant changes to existing licenses, the Government Decree on Environmental Impact Assessment Procedure (713/2006) [23], section 10, states:

"… assessment report shall contain … an assessment of the environmental impact of the project and its alternatives, any deficiencies in the data used, and the main uncertainty factors, including an assessment of the possibility of environmental accidents and their consequences"

2.1.2.2 Guidance

The regulatory guide YVL C.4 [24] provides detailed calculation requirements concerning atmospheric and aquatic dispersion of radioactive substances and population dose estimation. For example, the models, computational methods and computer programs used have to be verified and validated. This can be based on the verification and validation of models and methods existing in scientific literature, or on the comparison of results with results from previously verified and validated models, methods and programs.

Concerning license applications of nuclear power plants, this YVL guide states that for operational occurrences and accidents, the dose assessment shall examine:

- radiation doses to the most exposed population group and to various age groups via different exposure pathways
- long-term environmental contamination

The effects of countermeasures should not be included in the assessment of radiation doses related to accident conditions.

In severe accidents, short- and long-term doses in diverse weather and dispersion situations shall be examined separately. The contribution of various exposure pathways and significant nuclides shall be specified. With regard to short-term doses, an assessment shall be carried out of the extent to which intervention levels for initiating protective measures for the population are exceeded. In the assessment of long-term doses, examination periods exceeding three months shall be used.

Generally, when assessing public exposure from accidents, calculations may statistically consider the variability of weather-dependent dispersion conditions as well as any seasonal differences in radiation doses accumulating through food chains. The results shall be given in the form of appropriate distributions as well as averages and percentiles.

In the YVL Guide C.3 [25] guidance on how to treat countermeasures is given. The specification for design of a nuclear power plant says that in the case of a severe accident resulting in radioactive release, there shall not be need for evacuation outside the protective zone (< 5 km) and no need for sheltering outside the preparedness zone (< 20 km) and in addition that the release of Cs-137 remains under the release limit of 100 TBq. Even if this is guidance on deterministic analysis it gives a hint on what is accepted by the society.

2.2 International regulatory framework

Most countries with operating nuclear reactors, including the Nordic countries, impose Level 1 and Level 2 PSA criteria in the form of binding regulations or non-binding safety goals. Ultimately, these criteria stand as surrogates for implied off-site consequences. The Level 3 PSA studies performed in the Netherlands, South Africa, and the UK are performed to fulfil national safety regulations. The safety criteria used are further described in section 3.3 and the associated references. These criteria are used in part to supplement Level 2 PSA surrogate metrics and in some cases are used in place of binding Level 1 and Level 2 PSA criteria [11].

The United States does not have binding Level 3 PSA requirements for plant licensees. In the U.S. intermittent probabilistic consequence studies are performed on behalf of the U.S. Nuclear Regulatory Commission.

2.3 International guides and standards

Activity in Level 3 PSA has been low from the early 1990s until after the Big Eastern Earthquake and the tsunami that led to the Fukushima accidents in 2011. Therefore, relatively little new guidance on Level 3 PSA is developed. The most substantial international guidance in Level 3 PSA was developed by the IAEA during the 1990s and was issued as Safety Series No. 50-P-12 in 1996 [2], following significant work performed in the US, Europe, and Japan.

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 2012. The meeting was the first activity specifically discussing Level 3 PSA since the publication of the Safety Series No. 50-P-12. Following the IAEA Technical Meeting a series of Consultant Meetings have been held in order to develop new guidance on Level 3 PSA in the form of a TECDOC.

Parallel to the IAEA work on Level 3 PSA, the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) together have been drafting a standard for Level 3 PSA. The ANS/ASME-58.25 standard provides requirements for application of risk-informed decisions related to the consequences of accidents involving release of radioactive materials to the environment. The consequences to be addressed include health effects (early and late) and longer term environmental impacts. These requirements are articulated for a range of technical Level 3 PSA areas in a structure consistent with previously published ANS/ASME PRA (PSA) standards.

Significant work has been performed and is currently ongoing in deterministic calculations of the effects of nuclear accidents. In the future, these activities may be invaluable in improving Level 3 PSA. One particular area where deterministic consequence analysis and probabilistic analysis have struggled is in the quantification of economic effects. Notable ongoing work is being performed on economic consequences by the OECD/NEA “Methodologies for Assessing the Economic Consequences of Nuclear Reactor Accidents” [19].

OECD/NEA Working Group RISK (WGRISK) continuously collects information from ongoing PSA activities at member countries. The periodically updated document “Use and Development of Probabilistic Safety Assessment in NEA Member Countries” thus contains information about countries requiring and using Level 3 PSA [20]. Level 3 PSA has been also addressed in the risk criteria and safety goals survey [21].

2.4 Available software

Several codes are suitable for supporting Level 3 analyses, either the full process, or for different parts of it. All codes that support entire Level 3 PSA analyses (and not just some phases of it) have been developed in the 1990’s or earlier. They thus reflect the theoretical and applied knowledge and the computational and software engineering capabilities available of that time.

The following codes have been created explicitly for supporting Level 3 PSA analyses:

- LENA is a simple stand-alone tool developed in Sweden during the late 1980s for emergency preparedness organizations to quickly assess ongoing or postulated accidents. The program has some limited probabilistic capabilities. The program is still used by emergency response organizations in Sweden.
- ARANO is a Finnish program which uses a Gaussian plume model for calculating population doses and health effects. It was originally developed at VTT in the 1970’s. It is very fast and still used, although it has certain limitations, e.g. not being able to change wind direction, and not being able to take area topography into account.
- MACCS (MELCOR Accident Consequence Codes Systems) is a program that is developed in the United States. The code is available to participating countries in the Cooperative Severe Accident Research Program (CSARP), and is distributed much like the severe accident analysis program MELCOR.
- COSYMA (or PC COSYMA) is a software developed during the 1980s and 1990s for probabilistic off-site consequence analysis. The software is currently used by countries such as South Africa and the Netherlands to perform required Level 3 PSA analyses. The program is however, no longer supported.
- OSCAAR is a probabilistic consequence code developed by Japan Atomic Energy Research Institute in the 1980’s. The code has been under modest, but regular development through to the present. Some of the unique capabilities of the program are Eastern countries (i.e. Japanese crops) food chain models, and simultaneous multi-source handling.

These codes are typically quite old and have had, at most, modest development over recent years. Many of the elements of Level 3 PSA have new tools that are under rigorous active development. The aforementioned Level 3 PSA tools discussed above do not represent the state-of-the-art in terms of radiological dispersion analysis. Therefore it should be noted that one is not bound to the above codes, and many codes could be used in some phases of Level 3 PSA studies, or they provide functionality useful in those analyses. For example, hundreds of atmospheric dispersion codes exist, often developed by national metrological institutes; however, most of these codes have a limited applicability to Level 3 PSA due to constraints on the kinds of releases considered, limitations on the range of reliable dispersion estimates, and probabilistic handling of data and results.

Examples of these tools are:

- SILAM /VALMA. SILAM is an atmospheric dispersion code developed by the Finnish Meteorological Institute (FMI). VALMA provides a user interface to SILAM, with population dose estimation capabilities.
- JRODOS is a decision support system for accident management. It is primarily meant for real time decision making. A statistical analysis tool for countermeasure planning (weather statistics) has recently been implemented in the software making it more suitable also for probabilistic consequence analysis, however, the program does not handle probabilistic input nor post-processing of the probabilistic output.

3 Benefits, challenges and limitations

There are a number of benefits with a Level 3 PSA. One of the (unique) advantages that a Level 3 PSA can provide is the possibility to directly assess the risk to the public and environment. However, there are also several expected challenges when conducting a Level 3 PSA. One example of such challenge is related to handling of uncertainties, e.g. uncertainties in the analysis method, general uncertainties when working with probabilities and uncertainties from ingoing parameters.

To be able to give guidance on how to perform a Level 3 PSA, the use of results and suitable risk metrics must be defined as well as the safety criteria to use. The benefits and overall challenges and limitations of Level 3 PSA are briefly described below.

3.1 Use of results

3.1.1 Tool for decision making

The main expected motivations for performing a Level 3 PSA are to use the analysis as guidance for decision making, e.g., regarding emergency preparedness and costs for plant modifications. The increased knowledge can in its turn be of help for assessing new build projects (e.g. site location) or emergency preparedness response planning (e.g. decision making). [6]

3.1.2 Tool for risk comparison

Level 3 PSA makes risk comparisons possible, at least in theory. When comparing the risk with a nuclear power plant to the risks related to other energy production forms, the whole life cycle must be taken into account (by making a life cycle analysis, LCA). One possible metric for comparing the risk between a nuclear power plant and the risks from other types of energy sources is number of deaths (e.g. per produced TWh or per operating year) [6]. Likewise, the latent cancer risks from a nuclear power plant determined by a Level 3 PSA analysis can be compared to the general risk of cancer fatalities [9].

3.1.3 Tool for public communication

The results from a Level 3 PSA study can be used for communication to the public. The most important communication path consists of two parts. The first is related to communication from experts to authorities and the other is related to communication from authorities to the community (e.g. public, non-governmental organizations, and media). However, the regulatory bodies (e.g. STUK and SSM) are both experts and authorities. Communication by authorities is more important than communication by experts. [3]

Level 3 PSA can also provide support in the definition and interpretation of safety criteria applied in the regulatory framework, such as explain what thresholds mean in terms of consequences.

For the nuclear industry Level 3 PSA could help to:

- Communicate with insurance companies; the analysis could lead to better insurance possibilities.
- Communicate with the society in large and thereby create higher acceptance for nuclear power.
- Better understanding of societal risks from commercial nuclear power and thereby improve the emergency preparedness.
- Provide better design and siting considerations for new construction projects.
- Provide support in cost benefit assessments for plant retrofits.
- Improve and extend Level 1 and 2 PSAs, improving thus their completeness and quality.

3.2 Risk metrics

A risk metric has two components: 1) probability (frequency) metric and 2) consequence (or impact) metric.

The main probability unit used in the nuclear regulatory decision making is “probability per year per reactor”. Probability units “per lifetime” and “per produced energy (or electricity) over the complete fuel life cycle” can be considered in risk comparisons.

Table 3 summarizes the main consequence metric categories (health effects, environmental impact and economic impact), their advantages, disadvantages and associated uncertainties as well as uses.

Table 3 Metric, advantages, disadvantages, uncertainties and use for different consequence categories [6].

	Consequence category		
	Health effects	**Environmental impact**	**Economic impact**
Metric	Dose [Sv] or [person-Sv] Fatalities (#) Short- and long-term effect	Contamination level [kBq/area] or [mSv/year] Restricted land and sea area or “non-usable” land and sea areal (area)	Monetary units (e.g. [EUR] or [SEK]) Different costs are to be included depending on stakeholder (owner or insurance company)
Advantage	Relatively easy to estimate dose and connect dose to fatalities.	Relatively easy to estimate contamination of land and sea. Complements well the health metric.	Most complete metric, everything can be accounted for.
Disadvantage	Does not consider the total impact of a nuclear accident.	Contaminated area as a single metric does not characterize a site well in site studies, because the value of land and the consequences of its contamination may vary significantly. Use of multiple metric requires conversion factors between different environmental impacts.	Laborious to assess comprehensively and the impact is stakeholder dependent. May be difficult to agree on conversion factors for non-monetary costs.
Uncertainties	Risk models applied (convert dose to health effects) include large uncertainties	Conversion factors which are needed to compare different environmental impacts	Large uncertainties in the estimation of cost. Which cost are to be included. How to estimate the cost of different factors. Political factors can affect the results.
Use	Improve plant design and emergency preparedness Regulatory requirement [3]	Improve plant design and emergency preparedness In some cases regulatory requirement [3]	Improve plant design and emergency preparedness Communication with insurance company Optimization of safety improvements

[3] In those countries where Level 3 PSA is performed as part of fulfilling regulatory requirements.

Health effects and environmental impact are rather similar metrics from the estimation and purpose point of view. The assessment of these metrics should be of interest for all stakeholders. It could be expected that even internationally the stakeholders could agree on which metric to use and risk criteria to be applied. At least for health effects, there are references for safety goals and associated numerical criteria. For the environmental impact, numerical criteria may not be necessary.

Economic impact is an ideal metric from decision making point of view and it would allow cost-benefit studies to be performed. In practice, it can be difficult to agree on what to include in the quantification of economic impact and how to convert different impacts into a monetary scale. Despite the difficulties to evaluate economic impact, one possibility could be to apply some simplified categorization of economic impacts in terms of order of magnitude. It should be sufficient to estimate whether the cost is $\sim 10^9$ € or $\sim 10^{10}$ €. Main use of economic impact as a risk metric may be in cost-benefit assessments instead of being used in connection with numerical risk criteria.

3.3 Safety criteria

Level 1 and Level 2 PSA results are often quoted in terms of Core Damage Frequency (CDF) and Large Early Release Frequencies (LERF). Unlike Level 1 and 2 PSA, specific risk metrics and associated criteria have not consolidated to specific metrics/values for Level 3 PSA. This section explores some of the Level 3 PSA criteria that are used in countries that require Level 3 PSA along with references to these criteria. Further discussion on relevant risk metrics and considerations for specific consequences can be found in section 4.

Currently, there are very few countries that require Level 3 PSA for licensing of nuclear reactors. Countries that require Level 3 PSA include the Netherlands, South Africa, and the United Kingdom. For each of these countries Level 3 PSA criteria focus on the health effects risk metric category as described in section 3.2.

Criteria for the Netherlands are cited in this section. Similar criteria for the UK and South Africa can be found in [15] and [14] respectively.

<u>Dose criteria</u>

Projected or averted dose criteria can be used to assess the effectiveness of emergency operating procedures, off-site countermeasures, or plant design alternatives. One reason for focusing on dose is due to the fact that off-site health effects are largely influenced by changes to the population surrounding a site. Unlike Level 1 and Level 2 PSA, this is outside of the control of the plant. The U.S.NRC has implemented dose aversion criteria in Severe Accident Mitigation Alternative (SAMA) analyses based on cost-benefit criteria [40]. The Nordic radiation safety authorities recently published guidelines and recommendations on dose criteria for countermeasures in the early and intermediate phase of an emergency exposure situation [8]. The dose criteria stem from relevant reference levels set out in ICRP (International Commission on Radiological Protection) publication 103 [17] and 109 and, in contrast to the averted dose criteria used in US, refer to the projected dose[4] to an individual, see discussion in 4.2 for more information about countermeasures.

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[4] Projected dose – the dose that would be expected to be uncured if a specified countermeasure or set of countermeasures – or in particular, no countermeasures – were to be taken [8].

Health effect criteria

Health effects criteria are usually subdivided into individual health effects and societal health effects. Individual health effects are often used to demonstrate that acute health effects occur at very low frequencies, if at all, as a result of a nuclear accident.

In the Netherlands the criterion for individual risk states:

- The maximum allowable individual risk of death as a consequence of the operation of a certain installation is 10^{-6} per year [11].

Societal effects usually quantify the risk of latent cancers and latent cancer fatalities to a population. In the Netherlands the Level 3 PSA requirement is defined:

- The societal risk is defined as the risk of 10 or more casualties, which are directly attributable to the accident, and this risk shall be lower than 10^{-5} per year for 10 deaths, 10^{-7} per year for 100 deaths, 10^{-9} per year for 1,000 deaths, etc. [11]

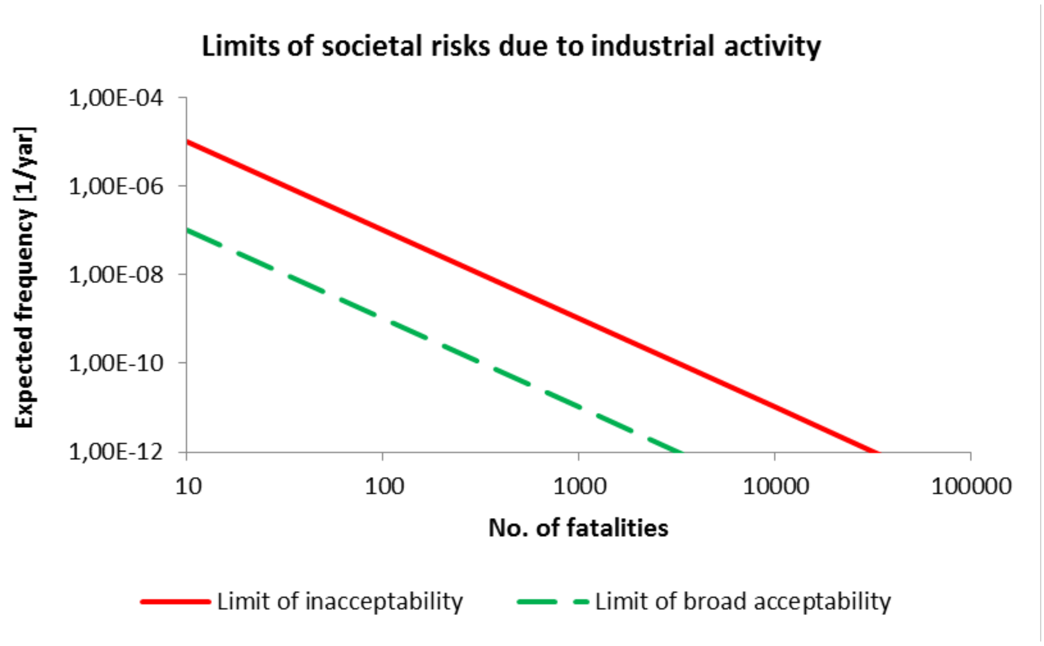


Figure 3 Visual representation of Level 3 PSA criteria in the Netherlands [11].

Environmental criteria

Few formalized criteria currently exist for environmental consequences; however, this may change in the relative short term. Criteria that could exist for assessing environmental impact could be calculations of the land area that experiences contamination above certain thresholds, such as specified by ICRP [10].

In Swedish pilot study the thresholds quantified were: Land area greater than 100 kBq/m^2, and 1000 kBq/m^2 Cs-137 contamination [30].

Economic criteria

Economic regulatory criteria have not been historically defined by the nuclear regulators and have not been regularly placed on Level 3 PSA studies. One exception to this is the criteria imposed by the United States Environmental Protection Agency which uses a 200,000 USD/person-Sv dose aversion metric. [16]

3.4 Resources for Level 3 PSA

The required resources for performing a study depend significantly on a number of considerations, for example:

- The consequences to be assessed,
- Availability of input data and analysis tools,
- Experience of those performing the study,
- How often a Level 3 PSA study has been performed,
- Level of detail of the study and the approximations made,
- Scope (e.g. geographic and temporal extent) of the study.

In countries that regularly perform Level 3 PSA studies the resources required to update the Level 3 PSA can be quite modest. In one expert's opinion it takes approximately 10%-20% of the resources required for update of the Level 2 PSA in order to update a large scope Level 3 PSA [36]. This resource estimation implies that many difficulties that would be experienced the first time performing a study have already been addressed in the past, such as collection and formatting of input data.

For organizations that are looking to start performing Level 3 PSA studies there are a few important aspects that should be considered. It is important to understand that Level 3 PSA is largely a radiological analysis which interfaces with Level 1 and Level 2 PSA. Therefore, background knowledge in atmospheric transport and radiological analysis is at least as important as knowledge in PSA. Furthermore, radiological background knowledge for deterministic analysis already exists in most organizations studying or regulating nuclear safety.

Collection of input data can be a difficult step in the development of a Level 3 PSA. There are several factors that need to be considered when starting the input data collection phase of a Level 3 PSA project. The credibility and level of detail of data of interest will depend on the Level 3 PSA analysis methods to be used. Weather data can often be procured from national meteorological organizations and sometimes are available from weather stations located on site. A first step when collecting data may be interfacing with emergency preparedness organizations and radiological analysis groups within one's own organization which may already collect relevant data or may have appropriate contacts at meteorological organizations and statistical collection bureaus.

Once data is collected it does not necessarily fulfil the input requirements for the methods used to perform a Level 3 PSA study. Time must be allocated to analyze, understand, and format data to perform the study.

Scope of the study (e.g. which release categories to be analyzed, single unit or entire site with several units, etc.) and the level of detail (e.g. complexity of dispersion models, aquatic pathways considered, etc.) will have a significant impact on the resources required to perform a Level 3 PSA. Two fairly simple pilot studies (see the project reports listed in section 6) have been performed in conjunction with this project. In these studies, some findings were made with very simple tools and limited input data, see section 5.3.

3.5 Understanding uncertainties

When the result from a Level 3 PSA study is presented it is important to include the uncertainty of the result. The uncertainty stems from data and modelling uncertainties in all three levels of PSA. The uncertainties could be presented added together or separately, since this information is of value in establishing priorities for further work and providing insights into the results of the probabilistic risk assessment.

There are modelling and parameter uncertainties. Parameter uncertainty is usually categorized as either aleatory or epistemic uncertainty. The distinction is that aleatory uncertainties characterize stochastic variability, for example weather variability. In Level 3 PSA, weather variability is the major form of aleatory uncertainty considered in most consequence analyses and it is generally quantified even when other input parameters are chosen as point estimates. Epistemic parameter uncertainty is associated with lack of knowledge and can be reduced by further study. An example in Level 3 analysis is the uncertainty associated with health effects.

In addition, there are incompleteness uncertainties, i.e. lack of completeness in treating all aspects of an accident scenario. This is usually difficult and resource intensive to study, it requires development of a model describing a previously not included phenomena. A best estimate approach is generally recommended because of the difficulty in determining uncertainties and difficulty in determining “conservative” assumptions. This arises because seemingly “conservative” assumptions may lead to non-conservative results in downstream considerations. For example, in terms of Level 3 PSA, if countermeasures are controlled via threshold values, if one conservatively over-estimates a release one might evacuate as opposed to shelter. Such an assumption would potentially provide additional exposure while the public is evacuating when if the release was less they would have remained in shelter.

Uncertainty is the major point of discussion in all levels of PSA analysis. Much like Level 1 and Level 2 PSA, in Level 3 PSA it is common to overly focus on the precise numbers produced in the analysis, rather than to view the relative importance of various results. The values calculated from a PSA study are best served as tools for comparing between the Level 3 PSA results *within* a given study (i.e. where the relative importance of a given release category can be determined, and which initiating events and features of safety systems designed for accident prevention and mitigation mostly contribute to the risk).

3.5.1 Level 2 uncertainties

Level 2 PSA risk can be defined as the product of the magnitude of the radioactive release and the frequency for the release. To represent the parametric uncertainties from the Level 1 and 2 PSA the uncertainty distribution can be evaluated by combining both the uncertainty from Level 1 and the uncertainty from Level 2 using a Monte Carlo approach within the quantification calculations. The resulting uncertainty distribution for each Release Category (RC) can be derived and brought into the Level 3 assessments as part of the input data.

The Level 2 PSA uncertainty analyses are often determined through the lens of sensitivity calculations. Such studies can be used to mark the range of source terms for a particular release category to represent modelling uncertainties. These source term sensitivities can be compared with the results of the parametric uncertainty analysis.

In the SOARCA (State-of-the-Art Reactor Consequence Analyses) study uncertain parameters were selected to capture the following [9]:

- accident sequence issues,
- accident progression issues within the reactor vessel,
- accident progression issues outside the reactor vessel,
- containment behavior issues, and
- fission product release, transport, and deposition within plant structures.

It is recommended that Level 2 PSA uncertainties are sufficiently characterized in contemporary analyses [29], [34], especially in order to assess eventual Level 3 PSA uncertainties. It is recommended that the Level 2 PSA uncertainty assessment should include the following:

- <u>Incompleteness uncertainties</u>: These uncertainties are difficult to address or quantify but should be discussed as far as possible.
- <u>Modelling uncertainties</u>: These uncertainties should be discussed as far as possible and should be addressed or quantified by sensitivity assessments.
- <u>Parametric uncertainties</u>: A rigorous propagation of uncertainties (e.g. Monte Carlo analysis) may be warranted to properly characterize the quantitative results, which are in turn fed to the Level 3 PSA [29].

3.5.2 Level 3 uncertainties

Often, Level 3 PSA studies do not include rigorous uncertainty analysis; rather a less complete sensitivity study of various parameters or modelling assumptions is performed in place of a full uncertainty analysis. The possibility of defining the epistemic uncertainty requires the availability of various different consequence analysis models and the ability to isolate various assumptions. Due to resource intensity and general scarcity of an analyst having an array of several probabilistic consequence analysis tools, such analyses are uncommon. Instead, the uncertainties related to input parameter are more readily quantifiable and therefore studied. When performed, parametric uncertainties are typically quantified through Monte Carlo simulations.

One of the most significant uncertainty/sensitivity studies performed for a probabilistic off-site consequence analysis to date is the uncertainty analysis performed in the SOARCA study for the Peach Bottom boiling water reactor. The SOARCA study, like the Swedish Pilot study only assessed a subset of source terms and not the entire spectrum of source terms as is often performed in a Level 3 PSA. Therefore, in the uncertainty analysis that was performed, no assessment of release category grouping was performed. In the SOARCA [28] study, uncertainties are connected to the following parameters:

- deposition data,
- shielding factors,
- input data to calculate health effects (dose coefficients, mortality risk coefficient, etc.),
- dispersion parameters,
- relocation parameters, and
- evacuation parameters.

The SOARCA study found that a significant set of the most influential parameters, given their defined uncertainty bounds, on acute radiation and latent cancer risks were associated with Level 2 PSA parameters (e.g. safety relief valve closure time, fuel failure criteria, etc.). The most influential severe accident progression parameters did not vary substantially between acute effects and latent effects. The most important Level 3 PSA parameters for latent cancer fatalities were the dry deposition velocity variability along with the mortality risk coefficients and dose rate effectiveness factors. For acute radiation, the most important parameters tended to by wet-deposition and acute radiation modelling parameters The SOARCA study, did not however, test the impact of modifying the dispersion model to a more sophisticated model or determine the impact of aqueous dispersion. [28]

4 Elements of the analysis

A probabilistic safety analysis Level 3, also called probabilistic consequence analysis, consists of several main elements such as input data, the input data converted into dose evaluation, possible countermeasures, estimation of possible effects converted into risk factors (risk metrics) and the effects expressed in some consequence metric [2]. This chapter aims to describe each main element, i.e. the inputs and outputs in the analysis as illustrated in Figure 4.

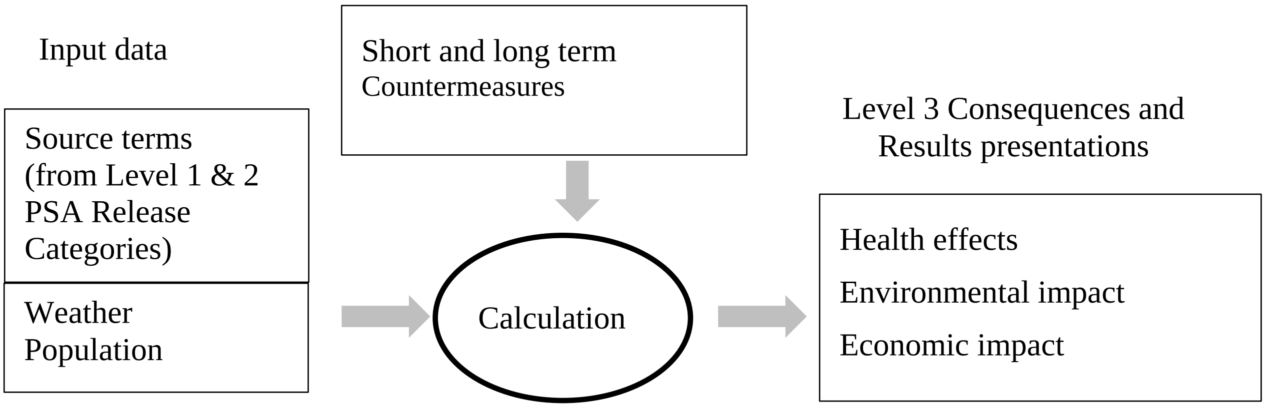


Figure 4 Illustration of main elements in Level 3 PSA.

In order for this guide to give more practical advice, the discussion in this chapter is based on three cases which aim at covering a spectrum of Level 3 PSA consequences. The cases are:

- Case A: Size of land area with significant Cs contamination (environmental risk)
- Case B: Risk of (early) death to maximum exposed individual (individual risk)
- Case C: Number of lethal cancers (late effects) (population risk)

For each case, attributes (elements) needed from Level 2 PSA and other sources when performing Level 3 PSA are defined in Table 3.

For each element of the analysis described in this chapter (the input data, applied countermeasures, resulting consequences and possible presentations of the results) considerations, recommendations and examples from the pilots are presented.

Date and time of the release may have significant effects on consequences. Frequency of external events (floods, storms, hail, deep frost etc.) varies seasonally. Consequences also have seasonal dependence as exemplified below:

- Amount of people in the affected area may depend on the season if there are summer cottages in the emergency preparation zone.
- Time of day of the release affects e.g. evacuation computations because at daytime people may be at work whereas in the night time they are generally in their homes.

Seasonal effects should be considered in modelling, computation, interpretation, and sensitivity analyses to such extent that is reasonable and practical. An important aspect to this is of course what the Level 1 and 2 PSA feed into the Level 3 PSA in terms of seasonal variations (e.g. differences in initiating event frequencies for external hazards). Since the Level 1 and 2 PSA normally generates average source terms (and their frequencies) over the year short time variations like daytime and night time may be difficult to capture in a realistic manner and it can be questioned if this would add any value in a practical application of Level 3 PSA. However, different type of seasonal and time variations can typically be taken into account in sensitivity ("what-if") analysis.

Table 4 Investigated attributes (elements) for a Level 3 PSA (per case).

Cases / **Attributes (elements)**		**Case A:** Size of land area with significant Cs contamination	**Case B:** Risk of (early) death to maximum exposed individual	**Case C:** Number of lethal Cancers (late effects)
Objectives	Type of output / Maximum impact (best estimate) & uncertainties	Area of lost land	Number of fatalities (short term)	Number of evacuated and number of fatalities (long term)
	Type of risk metric	Environmental	Health	Health
Input data / prerequisites	Source term	RCs (release categories) including Cs	Contributing RCs from large early releases	All RCs
	Atmospheric & metrological data	Results depend on how these weather input data interact with source terms.		
	Population, land-use & economic data	Population and land-use not needed.	Ingestion pathway not contributing.	
Counter-measures	Short or long term [8]	Decontamination	Prompt evacuation or sheltering	Depending on situation, all countermeasures influence the results
Level 3 Consequences, presentation of results	For discussion on recommendations for Level 3 PSA results presentation see section 4.4			

4.1 Input data and prerequisites

The input data required for a Level 3 PSA study depends on several factors. These factors include the consideration of the consequences to be assessed, input requirements of the calculation tools to be used, the level of detail of the modelling, and the underlying assumptions that are made for the study.

Even relatively simple Level 3 PSA studies require a significant amount of input data. At a minimum the following types of data are usually considered:

- Source term data,
- Atmospheric and meteorological data,
- Population and countermeasure data.

Each of the above listed input data will be briefly discussed in the following sub-sections. References to further discussion are provided for more detailed discussion on these and other potentially important input considerations.

4.1.1 Source term / Level 2 PSA interface

From the standpoint of a full-scope PSA the natural starting point for the consequence analysis is the postulated radioactive release to the atmosphere provided by Level 2 PSA for a number of release categories [2].

Considerations

The source term is often seen as the connection between the Level 1 & 2 PSA and the Level 3 PSA. In some cases, Level 3 PSAs are performed without significant input to upstream Level 1 and 2 PSA, in which case the source term may provide the sole link between the plant response and severe accident progression and the Level 3 input parameters. A 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 [1]. These practices are still largely the basis for current Level 3 PSAs. There has been some expansion in the level of input which Level 3 PSA codes can accommodate, albeit modest.

When looking at Level 2 PSA output it is important to focus on the aspects that affect how radiation is transported. As is discussed in section 4.1.2, atmospheric transport has been the primary mechanism considered in Level 3 PSA. For this reason, Level 2 PSA generally provides information regarding:

- **Magnitude** of release of radioactive material to the atmosphere. The magnitude is given for each radionuclide considered or group of nuclides, and are usually expressed as percentage of core inventory or in the activity of each of the nuclides released (e.g. in Becquerel).
- **Release timing**: the timing from initiating event to plant shut down to start of release and end of release.
- **Energy** of the release (i.e. heat).
- **Release location** (e.g. release height, coordinates).
- **Release frequency**.

Other transport mechanisms (e.g. due to basemat melttrough or leakage of radioactive water) may also be deemed important, in which case other relevant parameters may need to be considered.

The true values of radionuclide inventory depend on the fuel loading, the cycle burn-up, and if/how the plant was shut down. 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 condition.

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 in the rather simple methods employed for most probabilistic off-site consequence 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 the 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.

The end-states of most Level 2 PSAs are the definitions of releases categories. These release categories may be derived using various methods. For example, release categories may be developed by grouping similar accident sequences in a containment event tree. It is important for the Level 3 PSA analysis that the categories that are developed are composed of releases with similar source term characteristics (i.e. the bulleted list mentioned above).

Recommendations

A common concern in all levels of PSA is a general completeness of source term groups / release categories. The current state of practice for Level 3 PSA is to include on the order of tens of release categories, each with one representative source term. What is important to capture in the source term is that each of the grouped sequences would result in releases with roughly the same timing, and approximately the same release fractions for each of the isotopic groups.

The assessment of overall completeness ultimately depends on the consequences of interest and the level of detail that is intended. Health effects typically arise from very significant releases. Furthermore, acute health effects can usually only occur in the most severe accidents with early releases. When developing release categories or characteristic source terms for a given category one should be mindful of the release properties that most effect radiation transport. Through the lens of the cases outlined in Table 4, one can outline certain valid assumptions for each as follows:

> **Case A**: Size of land area with significant Cesium contamination
> In case A one is interested in the contaminated land values. This means that smaller releases, which still have the potential for significant low-level contamination, may be relevant. Depending on the risk criteria applied, it may be important to include large and small releases. However, if one is interested in specific contamination, for example Cesium contamination, releases limited to noble gases may be irrelevant (e.g. filtered releases).

Case B: Risk of (early) death to maximum exposed individual
Very significant releases are needed to cause acute health effects. It is likely that most of the accident sequences for release categories in a Level 2 PSA study will allow for evacuation and sheltering which will limit, if not eliminate, acute health effects. Therefore, it may be sufficient to focus efforts on very large early releases even if they are substantially less probable.

Case C: Number of lethal cancers (late effects)
Latent cancer fatalities are the focus of Case C. Since lower doses than those considered for acute radiation effects can cause cancers, smaller and later releases may be important. The extent of relevant modelling considerations depends on the modelling assumptions, i.e. if a linear no-threshold model is used then very small doses can still be postulated to cause cancer cases, however a model with a threshold may imply that very small releases are unlikely to cause latent cancer fatalities.

It may be of interest to even identify issues from Level 1 PSA that may have an impact on the Level 3 PSA application, e.g. to assess the impacts of external events on off-site consequences. In order to capture such correlations these types of data must also be contained in the Level 2 / Level 3 interface.

Examples

It is difficult to provide specific recommendations of the quantity and make-up of release categories and the method for which one should develop representative source terms for these categories. Broadly, the release categories should represent the spectrum of source terms given the scope of the analysis (e.g. internal events, external events, etc.). As an example, in a Level 2 PSA it may be justified and reasonable to have conservatively defined source terms for the applied release categories, especially since one release category (RC) may contain many different accident sequences. It is not unusual that the source term assigned to a release category is based on the accident sequence that gives rise to the largest release (source term). In a Level 3 PSA such conservatism in source terms assessment may however give rise to recommendation on evacuation which might have a negative impact in terms of off-site consequences. Instead, if more realistic (less conservative) source terms would be used sheltering, and/or other less harmful countermeasures would be recommended instead as a conclusion from the Level 3 PSA.

In this section it has been discussed that releases that could potentially cause early health effects are usually both large and early. This is a phenomenon that is shown in both the Finnish and Swedish pilot studies [4] [33]. Other probabilistic analyses have also used this concept as a basis for choosing particularly large releases as opposed to the more probable small releases [27]. Therefore, if one is interested in the risks of health effects, the release categories should focus on the more severe accidents.

When published, both the ANS/ASME standard, and the USNRC Level 3 PSA will provide valuable insight in the scope of release categories that represent a “state-of-the-art” analysis from the American perspective for a very large scale study.

4.1.2 Atmospheric and meteorological data

Historically, the focus of Level 3 PSA dispersion calculations has been on atmospheric dispersion since this is the dominant transport mechanism contributing to off-site health effects. The reason this is true is simply because atmospheric transport provides the quickest transport mechanism for an appreciable amount of radioactive material to reach the population. Therefore, for the determination of health effects, it is often seen as sufficient to focus on atmospheric dispersion to capture the frequencies or probabilities for health effects to the population [2]. In addition, ingestion pathways and ground water contamination as a result of the atmospheric dispersion are commonly used to determine the impact of restrictions on food and drinking water.

When performing detailed studies of contamination, impact on food stuffs or economic studies, other transport mechanisms such as aqueous transport mechanisms may have appreciable impacts on the results. Such transport mechanisms may be especially important if one is trying to capture low-level contamination.

<u>Considerations</u>

The input requirements required to perform the atmospheric dispersion calculations can vary substantially based on the complexity of the models used. At a minimum the following data are required for a basic Gaussian plume calculation:

- Wind speed
- Wind direction
- Atmospheric stability / buoyancy
- Mixing layer height
- Precipitation levels.

During the last several decades rather modest advances have been made in Level 3 PSA, however, quite significant advances have been made in the deterministic calculations of atmospheric dispersion. As these more advanced models (e.g. Gaussian-puff models, Lagrangian particle transport models, and computational fluid dynamics models (CFD)) incorporate more detailed models and are more readily influenced by site-specific phenomena a significant quantity of additional input will be required, e.g. wind fields, detailed topography, etc.

In the proximity of the nuclear site (approximately 5-100 km), and for atmospheric conditions common to Sweden, Finland and Denmark relatively simple atmospheric dispersion methodologies have been used to perform probabilistic consequence analyses. These simple methods have been assumed to be applicable due to the stability of the atmosphere, modest and consistent precipitation, and relatively flat topography. When a range of weather conditions are assessed as is done in a typical Level 3 PSA it may be appropriate to use a simple Gaussian Plume model in the range of 5-100 km. One complication that is common to the Nordic reactor sites is that they each have a coastal location. In order to better characterize coastal effects or to look at very near or very distant effects Gaussian Puff models or Lagrangian transport models would be required.

Recommendations

The current state-of-practice for atmospheric dispersion input data for Level 3 PSA is to use at least hourly meteorological data over the span of several years. The data required will be specified by the dispersion tools being used. In the ideal case the dispersion analysis software documentation will help to assist in the collection of the necessary data. These data are often collected for each of the Nordic plant sites and can be ascertained through the emergency preparedness organizations.

In Level 3 PSA analysis it is common to use Gaussian plume or Gaussian-puff dispersion. Gaussian dispersion models have been shown to have reasonable applicability, under ideal conditions, for the range of several km (1-5 km) out to approximately 50-100 km. In order to accurately analyze the on-site doses or the doses and deposition out much further than 100 km more advanced methods than a simple Gaussian model would be required [37].

Very near the site, local effects (e.g. building wash, deposition onto buildings) have a significant impact on the calculation of doses. Detailed analysis of such situations may require CFD analysis, which would greatly complicate the possibility of a fully probabilistic analysis. At extreme distances a Lagrangian particle model, is likely required. For analysis of the effects on global scale appropriate models of upper atmospheric effects may become relevant, which may be relevant for very hot releases which directly puncture the atmospheric boundary layer. When extreme distances are included it is important to keep in mind that the doses will be significantly smaller at these large distances. Therefore, it will be of greater importance to carefully handle and investigate the low-dose effects and associated large uncertainties.

Examples

LENA and ARANO, used in the Swedish and Finnish pilot studies, both use the simplified Gaussian plume dispersion model. These tools require input data consistent with the list provided in the considerations section. These programs, along with most other Level 3 PSA tools, also utilize default values for deposition velocities, plume meandering corrections, etc. It is important to take note of these assumptions and assess if they are relevant to a given application.

Reference [38] is a comparison study that was performed in 2004 comparing Gaussian plume dispersion, with a range of more complex two and three dimensional dispersion methods. The study was performed under nearly ideal conditions for the Gaussian plume model, but found that the Gaussian plume was within a factor of 2 up to 300 km from the release point.

4.1.3 Population, land-use and economic data

Consequence analysis codes require population, land-use and production data for estimation of short- and long-term health effects and economic impact. This data has to be coupled to the geography of the area being examined.

Considerations

These data are typically collected from a wide range of sources which for the first of a kind study may require significant resources to acquire. This, however, is a burden that is lessening significantly in future updates. Depending on the scope of the study, cross-border effects may be important. These considerations can also influence the complexity of input collection of population and economic data.

Health effects have traditionally been the fundamental consequence analysis assessed in Level 3 PSA. Logically, this would require, at a minimum, the population distribution for the area surrounding the plant site. In simple studies without highly detailed dispersion calculations, significant detail of the population distribution may not be necessary. When using more refined methods, division of the population distribution into separate cohorts (e.g. age groups) may be advised.

Date and time of the release may have effects on the subsequent consequences. The frequencies of external events (floods, storms, hail, deep frost etc.) vary seasonally and consequences can also have seasonal dependence: e.g. amount of people in the affected area may depend on the season if there are summer cottages in the emergency preparation zone. Time of day of the release affects e.g. evacuation computations because at daytime people may be at work whereas in the night time they are generally in their homes.

For estimation of economic impacts information on costs/resources in the area studied is needed. Data on the materials, equipment and labor costs of countermeasures (e.g. evacuation and decontamination) is also needed. In some cases, costs of population relocation, such as renting or construction of housing to the relocated, and e.g. their loss of salaries, need to be estimated using relevant data. If the cost of radiation-induced health effects is also taken into account, information on medical costs and the costs of loss of life expectancy are also needed. For a more thorough treatment of these, see [19].

Recommendations

Ultimately, the required input data and relevant level of detail for input data will be specified by the analysis tools used to perform an analysis. The range of distances analyzed in a particular analysis will be strongly tied to the dispersion models used. For Gaussian plume dispersion models the range of distances for which the model is applicable is typically from 1-5 km through to 50-100 km, as discussed in section 4.1.2. When more advanced dispersion methods are applied, population data out to greater distances may be relevant. If detailed near-site consequences are assessed, a complex dispersion method would be required (e.g. CFD) and highly detailed population distributions near the site may be relevant.

In Nordic countries, nuclear facilities are typically far from large cities and close to large bodies of water. Therefore, detailed input data about e.g. commuter traffic may not be required as they may be for highly populated urban centers. It is also important to note that population and land use data and related assumptions will also influence countermeasure considerations which are further discussed in section 4.2.

Related to the seasonal variations the general recommendation is that they should be considered in the analysis to such extent that is reasonable and practical. An important aspect to this is to assess what the Level 1 and 2 PSA feed into the Level 3 PSA in terms of seasonal variations (e.g. differences in initiating event frequencies for external hazards). Since the Level 1 and 2 PSA normally generate average source terms (and their frequencies) over the year, short time variations like daytime and night time may be difficult to capture in a realistic manner and it can be questioned if this would add any value in a practical application of Level 3 PSA. However, different type of seasonal and time variations can typically be taken into account in sensitivity (“what-if”) analysis.

Examples

In Finland, the National Land Survey of Finland has land property registers that may help in assessing economic consequences. In Finland, population data is available at the accuracy of square-kilometer.

In the Swedish pilot study, 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 520000 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. [31]

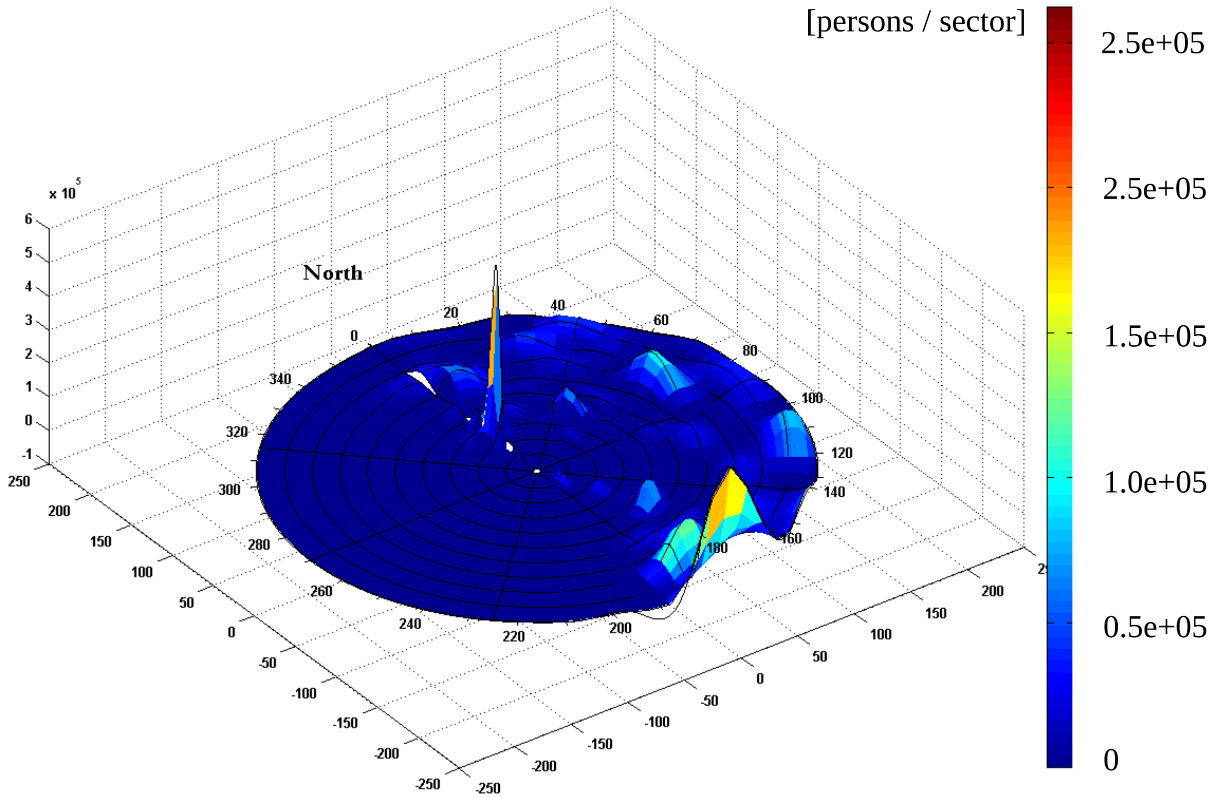


Figure 5 Surface plot representation of population distribution used in this analysis, representative of southern Sweden with maximum distance stretching 200 km and a coastal location. The colour bar to the right of the graphic shows the correspondence between the colours of the surface plot and the population in persons per sector [31].

4.2 Countermeasures

In case of an emergency there are pre-planned strategies for countermeasures. Common Nordic recommendations are found in a newly published report [8]. Countermeasures for the public are listed in Table 5.

The consideration of countermeasures can dramatically impact the results of a Level 3 PSA study, especially on the health effect risk metrics. Countermeasures can also have an impact on the economic impacts of an accident. Economic considerations are however not discussed in the recommendations provided in this section. A Level 3 PSA analyst's ability to incorporate and study various countermeasures is highly dependent on the tools and models available. Advanced Level 3 PSA tools (e.g. MACCS and COSYMA presented in section 2.4) provide many customizable models for various countermeasures.

<u>Considerations</u>

In probabilistic off-site consequence assessments, countermeasures can be considered as follows, depending on the scope of the assessment:

a) **Not included**: The effect of countermeasures to limit public exposure is not included as stated in the Finnish YVL guide C.4. Use this approach either if only contamination of land is of interest or if worst case scenarios should be calculated (no counter-measure deemed to be successful).
b) **Rudimentarily included**: For example, assume a successful evacuation of the emergency zone before the release and no countermeasures outside of the emergency zone. Assume that food and drinking water are taken from a region not contaminated, i.e. no dose from the ingestion pathway.
c) **Probabilistically included**: Each, in the analysis included countermeasure, are assumed (or assessed) to be successful/partly successful/not successful with a given probability. An uncertainty distribution is than coupled to the probability. Note that a countermeasure only should be executed if certain criterion is fulfilled.

In a detailed Level 3 PSA, the timing of the countermeasures can be very important. For example, the dose will vary greatly if evacuation is performed before, during, or after a passing plume. Also, timing of the lifting of countermeasures, such as relocation, will affect the dose result. In the case of accidents initiated by external events, the initiating event may effect of the timing and effectiveness of countermeasures, for example if the road network is compromised due to adverse weather the evacuation might be delayed and sheltering may be more effective.

In Nordic applications, the adoption of a countermeasure should be governed by the dose criteria as presented in Table 5. In order to rapidly respond to an emergency, the dose criteria are transformed to measurable quantities like e.g. dose rates. These triggers are usually referred to as Operational Intervention Levels (OIL). OILs are used in some Level 3 PSA software for establishing set points for various countermeasures. A discussion of Nordic OILs and guidelines for protective measures is provided in reference [8].

Table 5 Dose criteria for countermeasures [8].

Phase	Countermeasure	Dose[1] criteria
Early phase	Sheltering indoors	If the total dose is estimated to exceed 10 mSv in two days
	Partial sheltering indoors	If the total dose is estimated to be 1 - 10 mSv in two days
	Iodine prophylaxis	If the dose to the thyroid gland is estimated to be over 50 mGy for adults, and over 10 mGy for children under 18.
	Evacuation	If the total dose is estimated to be over 20 mSv in one week after the accident and if it can be anticipated already in this phase that sheltering indoors will last longer than two days.
Intermediate phase	Sheltering indoors	If the total dose is estimated to exceed 10 mSv in two days.
	Partial sheltering indoors	If the total dose is higher than 10 mSv in the first month after the accident but still below 10 mSv in two days.[2]
	Lifting the sheltering indoors	Can be lifted when the total dose is below 10 mSv per month.[2]
	Evacuation	If the total dose is estimated to exceed 20 mSv in one week after the accident.
	Relocation	Should be considered if the total dose exceeds 10 mSv in one month after the decontamination of the area.
	Lifting evacuation or relocation	When the dose is less than 10 mSv in the first month after return.[2]

[1] Projected dose to an unprotected person except for Iodine prophylaxis dose criteria

[2] Requires that the exposure will decrease rapidly or can be decreased effectively by e.g. decontamination.

One important element that should be considered is that a “conservative” assumption in terms of dose criteria (or other set points) might lead to suboptimal actions (for example, evacuation when sheltering is more justified). In sophisticated Level 3 PSA models, the distances or extent countermeasures are implemented and timing (starting and lifting of a countermeasure) can often be varied, which may be part of a sensitivity analysis. In the SOARCA study another timing issue was studied in the sensitivity analysis; delayed notification leading to delayed countermeasures. [27].

Another aspect studied in the SOARCA sensitivity study was how the scope of the countermeasure, here the size of the area evacuated, affects its own success. If a larger evacuation zone is modelled, the evacuation process is slowed down due to additional traffic congestion and other delays (more people to evacuate from the area). The exposure of the people living close to the release point to those closest to the plant will increase.

When several countermeasures are considered, it is important to understand that they are not implemented independently of each other and depend heavily on the specifics of the particular scenario. For example, if the accident leads to a delayed release and the weather situation results in a large spread of contamination, the resulting doses are most likely small. Food bans or lifting of access restrictions to public areas, its implementation and timing will be very important. On the other hand, if a large early release is analyzed and evacuation and Iodine tablet intake is assumed successful, the resulting doses will also be small and the lifting of the evacuation/relocation will be important. The Finnish pilot study modelled the success or failure of a countermeasure as a probability of a branch in an event tree including countermeasure alternatives [4].

Recommendations

The design of the countermeasure assessment in a Level 3 PSA study should include the following aspects:

- which countermeasures should be included
- when selected countermeasures are applied (time frame, dose criteria)
- to whom or in what area a countermeasure is adopted (countermeasure scope)
- the effectiveness or success of the countermeasures

Depending on the objective of the analysis, the countermeasures may affect the result differently and sometimes countermeasures are irrelevant. The following discussion provides recommendations for each of the Cases outlined in Table 4. It should be noted; regardless of the complexity of the countermeasure implementation in the analysis, a sensitivity analysis should be performed justifying the assumed approach.

Case A: Size of land area with significant Cesium contamination
No countermeasures are needed in the analysis.

Case B: Risk of (early) death to maximum exposed individual
Dose with and without protective actions should be analyzed. The countermeasures that should be included in the analysis are evacuation, sheltering and Iodine tablet distribution. Evacuation and sheltering should be assumed to follow the guidelines presented in Table 5.

Case C: Number of lethal cancers (late effects)
Dose with and without protective actions should be analyzed. The short term countermeasures that primarily should be included in the analysis are evacuation, sheltering and distribution of Iodine tablets. Evacuation and sheltering should be assumed to follow the guidelines presented in Table 5. In addition, countermeasures affecting the low doses in the intermediate and long-term phases need to be considered.

Table 6 presents the general timeframe when various countermeasures might be considered as effective in an analysis, the relevance of countermeasures for the different cases A-C, and a short description of the efficiency and impact of each countermeasure.

Table 6 Countermeasures for the public.

Countermeasure	Relevant timeframe	Relevant Case	Efficiency / impact
Evacuation	Short and long term[1]	B, C	Evacuation & sheltering can have substantial impact on results; these often have complex trade-off effects. If successful, results in avoidance of large doses and hence early health effects.
Sheltering indoors	Short term	B, C	
Partial sheltering	Short term	B, C	
Iodine prophylaxis • for adults • for children under 18 years old and pregnant women	Short term	C	Directly tied to latent cancer fatalities, specifically thyroid.
Protection of: • food production • commodities and products • indoor spaces of factories and production facilities	Short and long term	A, C	Can have impact on ingestion pathway, and low dose exposure. Also potentially important for eventual economics considerations.
Selection of crop varieties in agriculture	Long term	A, C	
Banning of foodstuffs, drinking water and other natural goods	Long term	C	Can have impact on ingestion pathway, and low dose exposure.
Restrictions on: • entering contaminated areas • built-up recreational areas • recreational use of natural areas	Long term	C	Impact on long term, low dose, exposure. Importance will depend on the LCF modelling considerations.
Decontamination activities: • Cleaning indoor spaces • Decontamination of vehicles, machinery and tools • Decontamination of built-up environment	Long term	C	Impact on long term, low dose, exposure. Importance will depend on the LCF modelling considerations. Will effect relocation considerations and economic impacts.
Relocation	Long term	B, C	

1 Long term in this report include the intermediate phase used in ref. [8] and the time thereafter

Examples

Swedish pilot

In the Swedish pilot study, one of the objectives was to analyze early health effects and the maximum individual dose within a specified distance was calculated. To understand the impact of evacuation, a 10 km evacuation zone was used in the analysis. Evacuation was assumed to be 100% successful in the evacuation area and implemented before the release of any activity.

Another objective of the pilot study was to analyze late health effects. Even if countermeasures would reduce the dose and hence the late health effect, these are not included in the analysis. The model used in the Swedish pilot does not include the ingestion pathway and therefore some of the countermeasures are not relevant. In addition, the analysis only covered dose uptake until one month after the accident, making long term countermeasures irrelevant. The analysis shows the conditional worst case late health effects, conditional since only a subset of accidents was analyzed and it was assumed that food and drinking water can be taken from some uncontaminated area.

Finnish pilot

In the Finnish pilot sheltering and evacuation were included in a probabilistic event tree model.

Firstly, the population dose was calculated for each end point of the event tree without evacuation and sheltering. For the end points with sheltering but without evacuation, the population doses obtained were multiplied by a ‘sheltering factor’. For the end points with successful evacuation, wind direction out towards the sea, and if wind speed below 4 m/s the population dose was assumed to be 0.

The probability of sheltering is set to 0.8. Evacuation success probability is calculated as the ratio of the time it takes for a plume to reach a city and the time it took to empty the evacuation zone in the Fukushima prefecture from people in March 2011 (approximately three days). If it takes longer time for the plume to reach the city than the evacuation to be conducted, the evacuation is considered a success with probability 1. Using this approach, the evacuation success probability turned out to be very small, <0.05.

An uncertainty analysis was performed using uniform uncertainty distributions, sheltering probabilities was assumed to between 0.6 and 1. On evacuation, the uncertainty analysis was conducting by postulating an uncertainty distribution on the evacuation time.

SOARCA [27]

In the SOARCA study, evacuation and returning home was included in the model. A sensitivity analysis on the size of the emergency zone was performed. SOARCA also include shadow evacuation, i.e. people outside the emergency preparedness zone evacuate even if they are not asked to do it. Shadow evacuation can delay the evacuation of people closer to the plant increasing their risk of exposure. Further, SOARCA included relocation for people outside the emergency zone, exceeding a given dose criterion. It should be noted that most of exposure in the SOARCA study came from low doses over long time to people returning home upon lifting evacuation.

4.3 Level 3 Consequences

As stated in section 3.2 there are three main consequence metric categories; health effects, environmental impact and economic impact.

Case A in Table 3 (Cs contamination) is an example of a simple environmental impact study. Health effect studies are exemplified in case B (early deaths) and C (lethal cancers) which may require more input than case A and are therefore, regarding needed costs/resources, examples of a more complex Level 3 PSA. Case A-C can all be transformed into monetary values, given that related costs can be estimated, with increased complexity accordingly.

4.3.1 Health effects (case B and C)

Health effects metrics considered in Level 3 PSA can either be expressed in terms of radiological risk (dose) or as radiation detriment such as acute radiation sickness in the short term and cancers in the long term. Non-radiological health effects, such as the consequences of stress to the population, are not usually addressed in Level 3 PSA studies.

<u>Considerations</u>

Dose assessment:

As mentioned in section 4.1.2, atmospheric dispersion is the most important transport mechanism resulting in exposure of humans. External exposure comes from “cloud shine” (most important noble gases and Iodine), “ground shine” (most important Cs) and radioactive material deposited on skin and/or clothes. Internal radiation comes from radioactive material inhaled (either directly from the passing plume or from re-suspension of deposition) or ingested from contaminated food and/or liquid. The important radionuclide in the case of inhalation is Iodine due its absorption in the thyroid and a large dose coefficient. For ingestion Cesium play a major role.

Depending on the time frame of the dose calculation, i.e., is it a life time dose or a two-week dose that is of interest, different exposure pathways will be important. For lifetime doses the ingestion pathway is most likely dominant, if people continuously eat and drink contaminated food/water.

Another consideration that affects the relative importance of the various exposure pathways is if/how countermeasures are included in the analysis. The approach in the SOARCA study is that there are enough water and food supplies in non-affected areas in the US that can be distributed to evacuated or relocated people from contaminated areas [27]. In this case the ingestion pathway is not relevant. Another example is if evacuation is assumed to be successful during plume passage, then cloud shine and inhalation from a passing plume are irrelevant.

A restriction may be the capability of available software to model exposure pathways. These restrictions need to be considered before the analysis.

Health effect assessments:

The calculated dose is a measure of the risk for health effects. To convert the dose to actual health effects or, as in most Level 3 PSA studies, number of deaths, a risk model needs to be quantified. There are different risk models for early and late health effects.

The probability to die due to acute radiation syndromes is usually described by a sigmoid function, characterized by its midpoint (LD_{50}[5]) and the shape of the curve which reflects the variation in individual radiosensitivity [17]. Due to installed mitigation systems (like containment filters) and applied countermeasures the doses can usually be expected to be well below this midpoint.

When using the linear no threshold model for health effects, the number of latent (fatal) cancers are normally derived by multiplying total population dose (in manSv) with a constant risk coefficient. In the case of cancer deaths, this coefficient is normally 0.05 (see, e.g., [7]). This risk coefficient is re-established in the ICRP publication 103 for the overall fatal risk for low dose and low dose rates [17].

This method is however questioned. ICRP stresses in the same publication that "the computation of cancer deaths based on collective effective doses involving trivial exposures to large populations is not reasonable and should be avoided". Further, in the IAEA report on the Fukushima accident [18] it is stated: "Since the accident occurred, several hypothetical estimates of future incidence of cancer have been reported in the media, sometimes basing predictions on calculations of collective dose or its computational equivalent. Such predictions are inappropriate (see Annex X) and may lead to anxiety and emotional distress among exposed populations."

In the IAEA and the UNSCEAR (United Nations Scientific Committee on the Effects of Atomic Radiation) Fukushima studies, the risk models are more sophisticated. The population is divided in different risk groups (age and sex-specific) and the risk coefficients for different types of cancers are used.

In the SOARCA study different approaches of estimating the risk of latent cancer death due to low exposure is presented. A linear no-threshold model is compared against several different threshold models, where below a certain threshold-dose the risk of death is assumed to be zero. Since a large number of the latent cancer deaths are attributed to low doses over long periods of time, the result of the threshold models shows a significantly reduction in the number of cancer fatalities [27].

Recommendations

The linear no-threshold (LNT) model with its risk coefficient of 0.05 for cancer deaths from population dose may be applied as a coarse estimate of late health effects. However, LNT is likely to be superseded by more sophisticated models, and developments in this field should be followed.

Due to the uncertainty and questioning of available risk models one should consider expressing the Level 3 PSA consequences in terms of doses, either as the only result or in addition to the estimated number of deaths.

[5] LD_{50} – The dose that is lethal for half of the population.

Examples

In many cases, health effects are low due to the low radiation dose to the general population. For example, The Finnish Pilot study [4] [42] postulated an imaginary accident at the Fukushima Daiichi site, where the actual source term of the March 2011 nuclear accident would have been released in a rather short time interval, and the population of the nearby major cities would have been in their homes (and having not died in the tsunami or been evacuated, as was the real case). The propagation of the radioactive cloud was analyzed under the weather conditions that statistically prevail in that part of Japan in March. It was estimated that the total expected number of cancer deaths to the Japanese population in major cities close to Fukushima Daiichi would have been 16 under rather conservative assumptions, and with a probability of more than 0.9 there would have been no cancer deaths at all. This can be explained by two facts: first, the wind speed is low for much of the time, and the winds in March blow predominantly to directions where the population density is low or non-existent (the Pacific Ocean); and second, the distance from the site to major population centers is so large that it allows evacuation well before the radioactive plume reaches them.

In the Finnish pilot study, a linear no-threshold model was used where the coefficient was set to 0.05 cancer deaths/manSv. This is the coefficient usually used for relatively low levels of radiation [7], such as typically those to the general public in the case of a nuclear accident.

4.3.2 Environmental impact (case A)

Recently there has been significant interest in the calculation of environmental effects and specifically the land areas associated with certain contamination levels without the more controversial extensions to health effects and economic effects, which may have significant uncertainties.

Considerations

A nuclear accident provides a potential risk of radiological contamination to the environment in the area surrounding the site. High levels of radioactive contamination pose threats to the health of humans and wildlife, safety of food products grown or manufactured in the area, and therefore may result in extended condemnation of land, water, and food products.

The impact to the environment of a nuclear accident can vary significantly based on factors such as specifics of the radioactive materials found at the facility that is undergoing the accident, the characteristics and timing of the accident, the weather conditions during the accident, the distribution of land and water surrounding the facility, etc. This means that it is potentially difficult to generalize relevant criteria and methodology for assessing environmental impact due to a nuclear accident. Despite these difficulties there are some commonalities and generalizations that can be analyzed using Level 3 PSA.

Short term contamination is generally concerned with Iodine and other relatively short lived radionuclides. Iodine in particular is readily taken up in the body, which makes it an important consideration for health effects as well as contamination, and specifically contamination of farm land and food products (such as dairy).

Medium and longer term contamination is generally more concerned with Cesium-134 (2 year half-life) and Cesium-137 (30 year half-life). These fission products are found in sizable quantities in an operating nuclear reactor and therefore these fission products may constitute a significant part of the release. Chemically, Cesium behaves similarly to Potassium and readily substitutes Potassium in clay minerals. This means that Cesium acts differently based on soil environments with clay and non-clay soils, and high or low Potassium levels. So for clay soils Cesium tends to be quite *immobile* and is less likely to be absorbed by plants. For non-clay and Potassium deficient soils Cesium is readily absorbed by plants.

Very long lived radioactive isotopes such as transuranic elements (Plutonium, Americium, etc.) are not typically the focus of off-site contamination because they are poorly transported through the environment and have a low volatility. Such very long lived isotopes should however be taken into consideration if the purpose with the off-site consequence analysis (Level 3 PSA) would be to study long termed effects, i.e. hundreds to thousands years after the accident.

Precipitation plays a significant role in the distribution of radiological materials following an accident. High levels of precipitation correlates to higher levels of contamination in the near vicinity of the accident site and lower at further distances. It will also affect water-runoff (land contamination that is transported by rain water to water ways) and contamination of water ways, which is either coarsely approximated or often not incorporated in current Level 3 PSA.

In general, the Level 3 PSA methods in their current state are well suited for performing short term land contamination and surface water contamination studies in the vicinity of a plant (tens of km). These methodologies are however less likely to be able to assess contamination to bodies of water and ground water due to the significant variations from site to site.

Recommendations

Determining the effects of a nuclear accident to agriculture, food production and consumption, and even health effects with significant accuracy is quite difficult. The effects may depend significantly on social actions and food production and consumption patterns which are hard to predict. For this reason, it may be less controversial to describe the severity of an accident through the area and magnitude of area contaminated above certain thresholds.

Examples

The Swedish pilot study used relatively simple methods in order to estimate the land areas exceeding 100 kBq/m^2 and 1000 Bq/m^2 of Cs-137 contamination.

4.3.3 Economic impact (case A-C)

Considerations

As concluded in section 3.2, economic impact is an ideal metric from decision making point of view and it would allow for cost-benefit studies. In practice, it can be difficult to agree on what to include in the quantification of economic impact and how to convert different impacts into a monetary scale. Liability, for example, is very different for different owners.

The main use of economic impact risk metric may be in cost-benefit assessments instead of being used in connection with numerical risk criteria.

Recommendations

Two approaches are suggested for the estimation of the economic impact, simple and advanced (respectively called top down and bottom up approach in [19]). Both approaches are related to the INES (International Nuclear Event Scale) accident classes, level 4-7.

In order to correlate the radioactive release to a level of economic impact, a scale of radioactive release is presented in [5], where each "Release level" category corresponds to an INES level. In the same report a scale of property damage is presented, where the levels describe to which extent the reactor core, reactor coolant pressure boundary (RCPB), and containment have been damaged. In Table 7 and Table 8 the cost groups related to each "R-category" respectively "D-category" are described.

Table 7 Cost groups per Release level category (R-category) [5].

The table shows how the cost groups and cost subgroups can be described for each level in the Radioactive release (R) category.

Cost Group	Sub-groups	Release category			
		R1	R2	R3	R4
		Acceptable releases and Tens of TBq ^{131}I	Unacceptable releases and Hundreds of TBq ^{131}I	Large releases and Early large releases and Thousands of TBq ^{131}I	Several tens of thousands of TBq ^{131}I
Evacuation	Transport	No evacuation needed [a]	Evacuation to first level [a]	Evacuation to second level [a]	Evacuation to third level [a]
	Accommodation	No accommodation needed	Sheltering and temporary accommodation needed.	Sheltering, temporary accommodation and permanent resettlement needed	Sheltering, temporary accommodation and permanent resettlement needed; to larger extent than in R3.
	Loss of income	Loss of income due to short-term restriction on farmland, fishing and forestry industry.	Loss of income due to temporary unavailability, of e.g. farmland, and possible loss of product production, e.g. milk production, for a period.	A loss of income due to temporary unavailability of e.g. farmland, and long-term loss of economic use of land e.g. farmland and forest. Longer periods and larger extent than in R2.	A loss of income due to temporary unavailability of e.g. farmland, and long-term loss of economic use of land e.g. farmland and forest. Longer periods and larger extent than in R3.
Health effect costs	Compensation for fatalities	Fatalities possible among rector personnel.	Fatalities possible among rector personnel.	Fatalities possible including among personnel and public.	Fatalities possible including among personnel and public.
	Compensation for early health effects	Possible early health effects among personnel. No expected early health effects among public.	Possible early health effects among personnel. No expected early health effects among public.	Early health effects possible among personnel and public.	Probable early health effects among personnel and possible public.
	Compensation for late health effects	Possible early health effects among personnel. No expected early health effects among public.	Possible early health effects among personnel. No expected early health effects among public.	Late health effects possible among personnel and public.	Late health effects probable among personnel and public.
Restoration cost	Restoration of surrounding environment	No restoration needed	Possibly some restoration needed	Restoration of surrounding environment needed; extent depending on affected area and disposition of nuclides	Substantial restoration of surrounding environment needed; extent depending on affected area and disposition of nuclides

a) In the table the evacuation cost category depend on three levels of evacuation. This assumption has been made based on literature study of the Fukushima accident, in which three evacuation zones where set up in order to protect the public. In the table it is assume that the evacuation always involves three levels, but that in R1 to R3 all levels are not needed and evacuation will only occur to the first (R2) and second (R3) level.

Table 8 Cost groups per Damage level category (D-category) [5].

The table shows how the cost groups and cost subgroups can be described for each level in the Property damage (P) category.

Cost Group	Property damage category			
	D1	D2	D3	D4
	Damage on reactor core	Severe damage on the core and significant effects on the RCPB	Severe damage on the core, the RCPB and the surrounding systems, limited damage on the containment	Severe damage on the core, the RCPB and the containment as well as the surrounding systems
Value loss on machinery and property	Partial value loss of reactor	Complete value loss of reactor	Complete value loss of reactor	Complete value loss of reactor
Reactor area clean-up	On-site in reactor building clean up. Limited to inside environment.	On-site in reactor building clean up. Limited to inside environment.	Extensive clean-up on-site. Limited to inside environment.	Extensive clean-up on-site needed. Both inside and outside environment on the plant area.
Costs related from final storage of radioactive waste	Partial core and vessel object to final storage.	Full core and vessel object to final storage.	Full core object to final storage; partly in-vessel, partly ex-vessel. Contamination to reactor containment and vessel.	Full core in-vessel and ex-vessel object to final storage. Contamination to all in-building systems.

Simple estimation of cost

A simple, robust and cost effective (in terms of required resources) way to estimate economic consequences is to look at empirical data of nuclear accidents. In Table 9 an estimated damage cost related to the severity of the nuclear accident in terms of INES level is presented. This table is based on empirical data of economic impact reference cases investigated in [5] and [26]. For better understanding, comparable costs are also provided.

Table 9 Estimated damage cost related per INES level.

Event Scale	**Damage level**	**Fatalities**	**Damage cost** ($G€=10^9€$)	**Comment** ~ Estimated or Comparable cost
INES 4	Property damage D1-D3, Release damage R1	None	<1 G€	Partial or complete loss of reactor ~ Losses 2020 due to Ringhals 1-2 phase out, ~0,5 G€
INES 5	Property damage D1-D3, Release damage R2	Possible in personnel	1-10 G€	Loss of reactor and unacceptable release. ~ Windscale 2 G€, TMI 10 G€ ~ 0,25-2,5% of Swedish GNP ~ Nuon losses 2013, ~1,5 G€ ~ Volkswagen emission scandal 2015, ~10 G€
INES 6	Property damage D2-D4, Release damage R3	Possible in personnel and public	10-100 G€	Large release ~ 2,5-25% of Swedish GNP ~ Deepwater Horizon oil spill 2010, ~50G€
INES 7	Property damage D4, Release damage R4	Probable in personnel and public	100-1000 G€	Severe release, ~ Chernobyl 300 G€, Fukushima 270 G€ (3 units) ~ 25-250% of Swedish GNP (GNP 2014 = 415 G€) ~ Swedish national depth 140G€, 37% of GNP

Advanced assessment of economic consequences

There is significant discussion and work being performed in the economics of nuclear accidents in the wake of Fukushima. For a more thorough Level 3 PSA of economic consequences it is relatively common to consider all potential economic impacts.

Examples

Cases A-C in Table 4 provide examples of three different applications. Cost estimates are needed to be able to compare the economic impact stemming from these applications. For this matter an estimation of cost per contaminated area (case A), cost of fatalities (case B and C) and cost of evacuation (case C) can be performed.

4.4 Presentation of results

The results that can be presented vary significantly for Level 3 PSA studies, the ways these results can be presented vary even more. Level 3 consequences results are usually presented through various risk metrics, which are discussed in section 3.2. In the case of health effects, these are the estimated number of acute radiation sicknesses in the short term and cancer deaths in the long term. Studies may also choose to convey results in terms of individual doses, collective doses, or environmental metrics such as land areas exceeding certain contamination thresholds. In the case of economic consequences, these may be the amount of electricity production lost at the site, the amount of production lost in the fallout area due to evacuations, the value depreciation of the real estate in the affected area, and the costs of evacuation and other countermeasures.

Considerations

Good practice in the presentation of results for a Level 3 PSA should be approached much like any other technical report. The scope of the study and risk metrics need to be presented as clearly as possible (see sections 3.2 and 3.3 for applicable risk metrics). The relation and applicability of the risk criteria should also be mentioned at the onset of the result presentation. Furthermore, assumptions and limitations should be clearly stated along with a description of their impact. Since the results and conclusions of a study are tightly coupled to the input data and methods applied these should also be adequately described. Finally, uncertainties should be quantitatively presented or at least qualitatively discussed.

Level 3 PSA results usually provide a distribution of risks and associated frequencies. It is important to represent both the magnitude of a consequence along with the frequency or associated probability. A common way of presenting Level 3 PSA consequence magnitudes and frequencies simultaneously are with so called “Farmer’s curves”. Farmer’s curves also have a list of aliases including, exceedance curves, Complementary Cumulative Distribution functions (CCDF). These curves show the probability or frequency (x-axis) that a certain event occurs or is exceeded (y-axis). Therefore, as one travels from left to right the trend of a CCDF curve is to strictly decrease. An example of a Farmer’s curve for the probability of cancer cases, as calculated in the Finnish Pilot study, is shown in Figure 6. Farmer’s curves are also presented in the Swedish pilot study as shown in Figure 7, which shows the exceedance frequencies of collective doses for three different release categories.

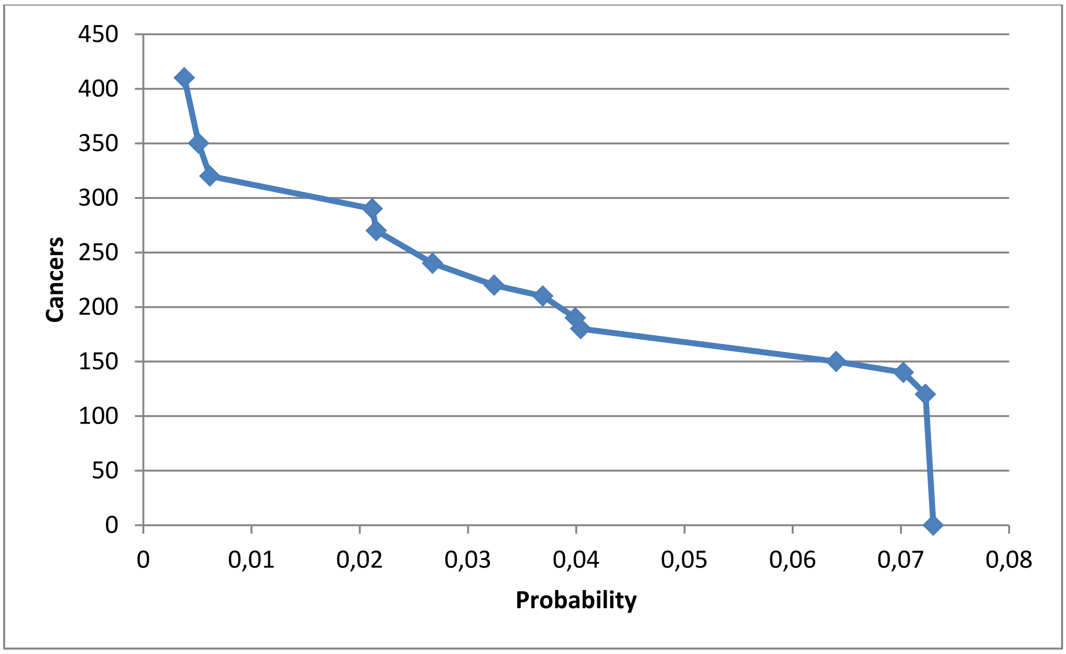


Figure 6 Farmer's curve plotting the number of cancer deaths against the conditional probability given an imaginary Fukushima-like nuclear accident (see Finnish Pilot Study [4]).

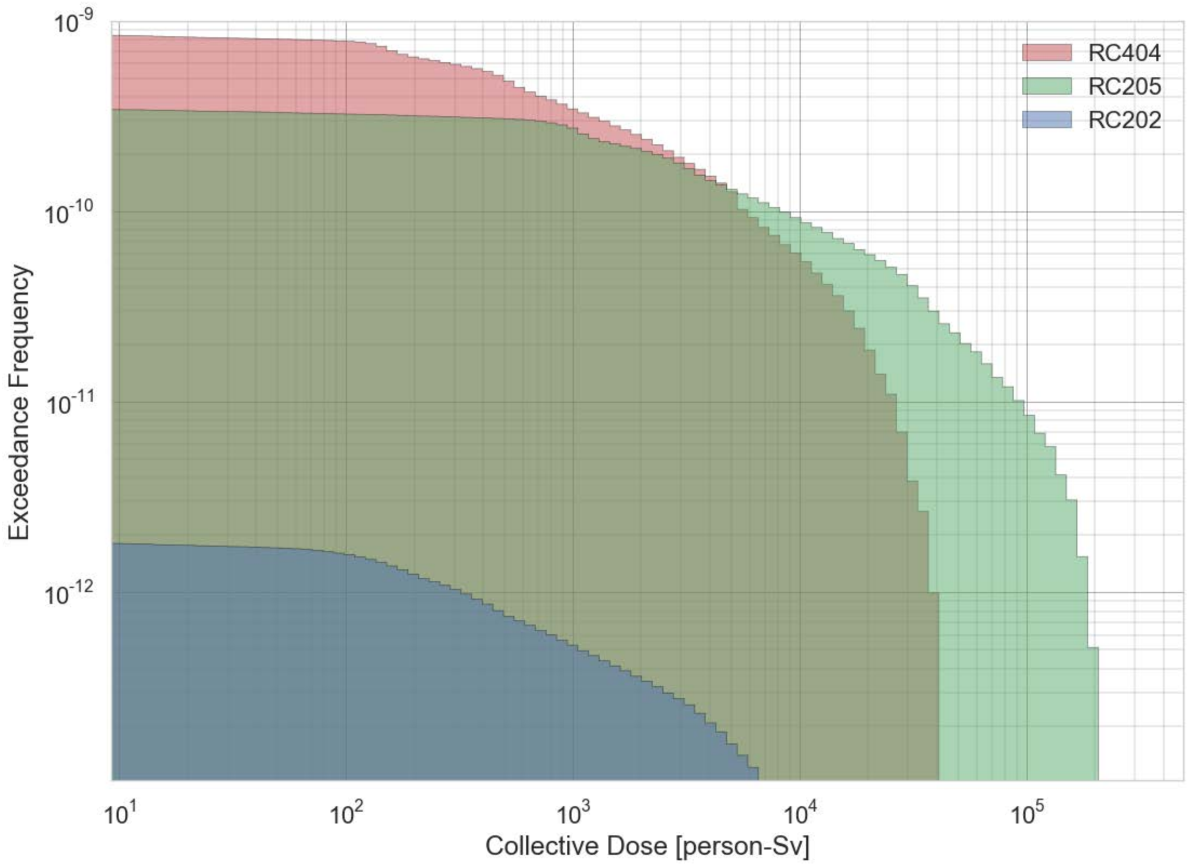


Figure 7 Another representation of a "Farmer's curve" depicting the exceedance frequency of different collective doses for three different release categories (see Swedish Pilot Study [33]).

Recommendations

Since scope and purpose Level 3 PSAs may vary significantly the important results and metrics used may not be the same in different studies. Likewise, the best way to present the data will also vary. What can be stated is that when presenting Level 3 PSA results, the emphasis should be on representing and describing the consequences (risk metrics). When countermeasures are assessed, their effectiveness should be presented through a comparison of a metric both without any countermeasures and with each countermeasure applied (or all that would be applied in each situation).

With respect to Level 3 PSA results, graphical presentation of consequence calculations is preferred, with numerical values relegated to tables (might be presented in appendices). This is relevant because significant uncertainties may exist and it is more meaningful to use Level 3 PSA results for understanding trends / high level changes rather than focusing on exact numbers.

Above all, it is important that the results are presented clearly. Results do not need to be presented in overly complicated ways. Often these studies present the end of a long line of probabilistic safety analyses (Level 1, 2 and 3 PSA). These studies have huge numbers of assumptions and considerations that one must incorporate into a results discussion. It is not likely that those reading the results of a Level 3 PSA study will be familiar with all of these assumptions and their implications. Therefore, it is important when presenting the results to be clear, concise and direct. One should point out many of these considerations to the reader, highlighting those of greatest relevance. The SOARCA study, which was a very large research project, has a modest set of simple bar charts and tables used to show the results of the study, but it also includes substantial discussion that easily breaks down the results and the implications of these results [27].

Examples

As an example of presentation of results in a well-known study, consider the summary report of NUREG-1150 study [1]. The main results presented with respect to off-site consequences are early fatalities, late cancer fatalities, population exposure in a 50-mile zone, and population exposure in the entire site region (in person-rems). These are presented in graphical form by plots of the complementary cumulative distribution functions (e.g. the number of early fatalities against the exceedance frequency per year). Also site-specific parameters (e.g. source terms and their frequencies, exclusion area radius) and assumed countermeasures are listed. The values of the following risk measures are given:

- early fatality risk,
- latent cancer fatality risk,
- population dose within 50 miles of the site,
- population dose within the entire site region,
- individual early fatality risk within 1 mile of the exclusion area boundary, and
- individual latent cancer fatality risk in the population within 10 miles of the site.

Also the most important initiating events and plant damage states (by their contribution to the risk measures) are identified, and their proportion of the total consequences is plotted in pie diagrams. The summary for each of the plants considered concludes by presenting general observations of the plant's safety in the light of the risk analysis.

In the Finnish pilot [4] [42], the central results were the (probability, cancer deaths) pairs, representing the risk. These figures were obtained from the end points of the event tree. These pairs were represented in a table, where columns represented wind directions (four in all, corresponding to the compass points of the nearby major cities relative to Fukushima Daiichi site), and rows represented triplets of wind speed, precipitation (rain/no rain) and sheltering success (sheltering/no sheltering). Evacuation was not represented in the table, because successful evacuation of a city led to zero population dose in it. The results were summarized in a figure representing Farmer's curve, or the plot of the number of cancer deaths (in decreasing order) against the corresponding probability (see Figure 6 above). The results of uncertainty analyses were represented as the cumulative probability distributions of the expected number of cancer deaths, and the probability of having any cancer deaths at all (i.e. the probability that the number of cancer deaths equals or exceeds 1).

In the Finnish pilot, the interpretation of results is easy, each (probability, cancer death) pair is a risk figure for the consequence measure used. Farmer's curve presents the probability of a number of cancer deaths occurring. Cumulative uncertainty distributions give the range within which the expected number of cancer deaths varies, and the range within which the probability of the number of cancer deaths exceeding 0 varies.

In the Swedish study [33], separate results are calculated for early health effects, latent health effects, land contamination, and each of the release categories included in the study, for each of the weather cases used (hourly data over 2 years, 17520 data points). Therefore, one of the assumptions made in the analysis was to assume each weather case as independent and equally probable. Therefore, a statistical set could be analyzed for a given source term. These data are presented in tables of mean or median values, and in box plots such as the plot provided in Figure 8.

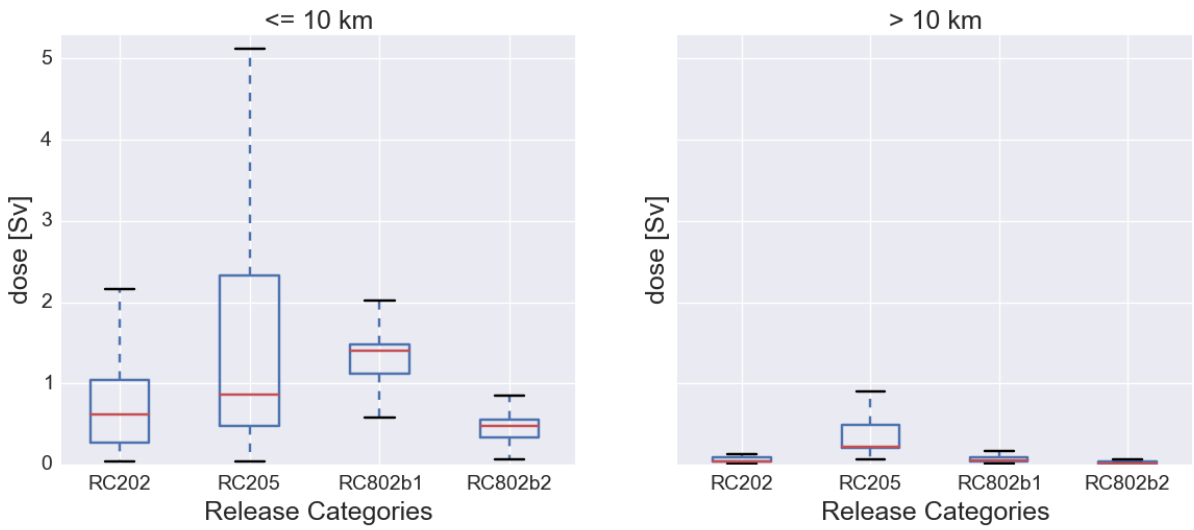
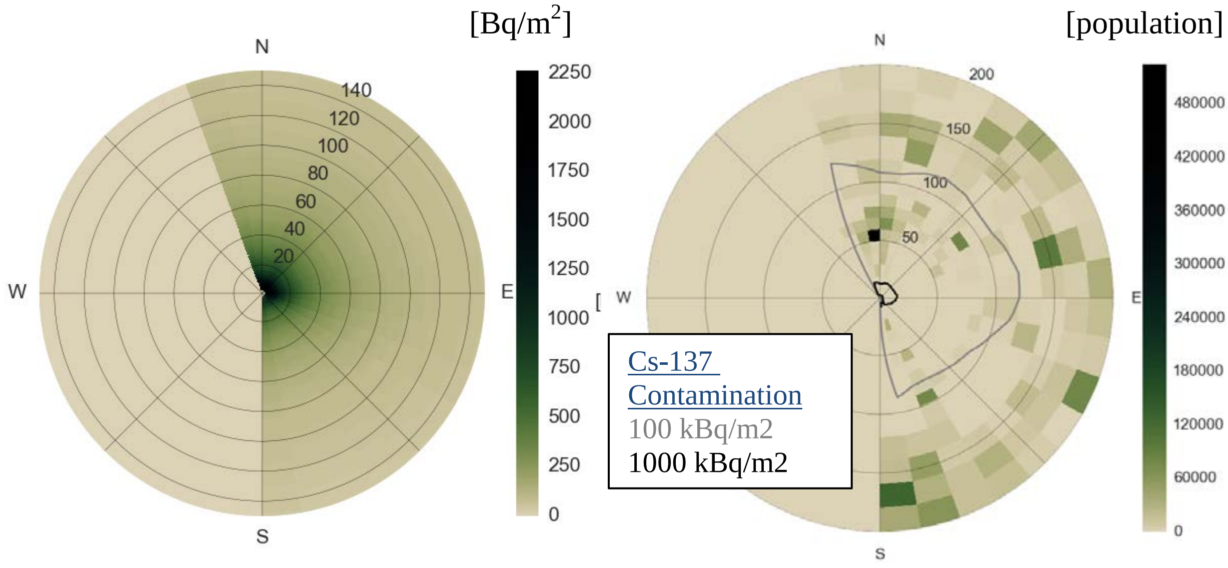


Figure 8 Boxplots from Swedish pilot study showing range of results for weather scenarios [33].

In order to preserve the directional component and to develop an understanding of the risks in terms of direction, probability weighted sums of each of the weather cases were also calculated for several of the metrics. These results are shown graphically and compared against the mean and median values for each of the individually calculations which were presented in the boxplots. An example of how these calculations could be viewed visually, and relative to a population as shown in Figure 9.



(a) (b)

Figure 9 (a) Mean Cs-137 contamination of all weather conditions for RC 205.
(b) Population distribution plotted along with isolines for weighted sum of 100 kBq/m^2 (grey) and 1000 kBq/m^2 (black) contamination.

In the Swedish study the interpretation of the results was quite complex because lots of result data were produced, however limiting assumptions were also imposed. The effects of some of the assumptions could be tested with sensitivity studies, but the methods did not allow for rigorous uncertainty analyses.

5 Summary and conclusions

The discussion and conclusions presented in this Guidance Document are based on the whole multi-year project where specific insights have been gained from several of the activities initiated during this project. This section provides a high level summary of the Guidance Document at the same time as some of the more important conclusions are highlighted.

5.1 Guidance scope and objectives

This Guidance Document focuses on the current state of Level 3 PSA and the elements of potential Level 3 PSA studies. Additional details can be found in the project reports, the referenced documents, the international guidance documents mentioned in this report, and the soon to be released ANS/ASME Level 3 PSA standard. The objective of today’s Level 3 PSA studies is to assess the off-site radiological consequences of a radioactive release caused by an accident at a nuclear facility. The consequences of interest can be separated into the following general categories:

1. Health effects
2. Environmental effects
3. Economic effects.

5.2 Level 3 PSA status

5.2.1 Not a standardized analysis

Since there are relatively few examples of Level 3 PSA, it is not currently standardized. Many different types of analyses can be performed that may be suitable for different purposes for different stakeholders; e.g. what is important in siting of a new plant may not be relevant when developing mitigation strategies for an existing plant, or when one is trying to develop financial liabilities / insurance reference levels. Different objectives will need consideration of different consequences and use of related metrics that support decision making for the expected end user of the result, i.e. stakeholders.

5.2.2 Generic analyses

In the course of the project, stakeholders demonstrated an interest in the investigation of generic Nordic Level 3 PSA analysis, and whether such a study would be satisfactory and useful. There is ample potential for genericity, since there are only few relevant factors specific to each site (such as source terms, local weather conditions, local demographics, road network). The number of atmospheric dispersion runs, the weather conditions considered, statistical analyses applied etc. can be shared between sites in a generic way. Unfortunately, this question cannot be completely addressed, since different types of potential stakeholders have different needs and interests with respect to probabilistic off-site consequence analysis. The non-trivial task of explicitly outlining the purpose of Level 3 PSA should be first decided. Only after such a decision can practical guidance be developed on to what extent Level 3 PSA can be made generic and cost-effective. Additionally, further investigation of Level 3 PSA uncertainties, and comparison of metrics of interest (e.g. assessment criteria, safety goals) should also be established when further developing guidance on the use and adequacy of Level 3 PSA studies.

5.2.3 Why Level 3 PSA?

In spite of its impediments, it is found that Level 3 PSA is useful in many regards because it provides the unique ability to investigate the range of possible consequences and give an estimate of their relative likelihood. Below some potential benefits from a Level 3 PSA are listed.

A) Complement existing deterministic consequence analyses

Deterministic studies focusing on health effects are already performed with regularity in order to fulfil Nordic nuclear safety requirements. Level 3 PSA can provide further understanding of consequences for the spectrum of PSA derived source terms.

Furthermore, as Level 3 PSA provides a framework to analyze conditions that are not specifically discussed in deterministic analyses the Level 3 PSA may be used to integrate individual deterministic consequence analyses into a systematic risk-based assessment.

B) Improvement of Level 1 and 2 PSA

In the same way as when a Level 1 PSA is extended to Level 2 it is common that some conservative assumptions, or simplifications, made in the Level 1 PSA need to be revised. The extension to Level 2 will therefore often result in an improved Level 1 PSA. A similar positive effect is likely to occur on the Level 1 and Level 2 PSA when extended to Level 3.

An example of the above mentioned positive effect is the additional attention which needs to be applied to Level 2 PSA results. The analysis of Level 2 PSA results and source terms often focuses on the frequency assessment of “Large Releases” or “Large Early Releases”. When Level 3 PSA analysis is performed, other important considerations with respect to timing and magnitude of releases must be considered in a more detailed manner, which can benefit both Level 1 and Level 2 PSA studies.

C) Justification of Level 1 and 2 PSA acceptance criteria

In the case the Level 2 PSA does not focus predominately on “Large” or “Unacceptable” releases, i.e., releases that are greater than a certain acceptance criterion (see section 2.1), but more focus on analyzing the full distribution of source terms (“source term risk”), a Level 3 PSA would still be beneficial since it is the off-site consequences that define the risk. As written in section 2.2 the Level 1 and Level 2 criteria used are surrogates for implied off-site consequences.

D) Communication of risk

Level 3 PSA assessments make it possible to discuss risk with potentially impacted external organizations and groups, e.g. Level 3 PSA could be a useful tool in the development of insurance premiums and liquidity requirements, or assessing and communicating public risk with authorities (both radiation protection authorities and other authorities).

E) Risk comparison

Off-site risk considerations are required and performed in other countries consistent with the off-site requirements of other industries ([11], [14]). Given that the risk metrics used are chosen in such a way that they are not dependent on industry it would be possible to compare risk contributions from different industries and between sites within the same industry.

It shall be noted though that even if the possibility of comparison of societal risks exist, care needs to be taken. Differences in the scope, assumptions, and end states have to be identified and their effect on the risk metrics needs to be clarified.

F) Link between different experts/communities

Performance of a Level 3 PSA should mean that an interface between the radiological and PSA communities needs to be created. These groups are addressing similar issues concurrently, both with separate skill-set and insights. Similarly, as the Level 3 PSA provides insights in terms of off-site consequences from a probabilistic point of view it may be used for recommendations related to emergency preparedness (e.g. the effect of different countermeasures like evacuation versus sheltering).

5.2.4 Challenges, and limitations

In terms of challenges, the major challenge for using Level 3 PSA to assess risk is the uncertainties in its results. This is also a major concern in Level 2 PSA and to an extent even in Level 1 PSA. In the relatively simple pilot studies, little could be directly addressed with respect to uncertainties. Fundamental questions, which remain to be addressed such as whether uncertainties prohibit the usefulness of results, require clear demonstration and presentation of uncertainties. Some referenced materials, such as the USNRC SOARCA analysis [27] [28] have started to address these questions for a subset of source term considerations. Continuation in the investigation of specific Level 3 PSA related uncertainties, and also Level 1 and Level 2 uncertainties, will support interpretation of results and decision making.

Further challenges in Level 3 PSA stem from the fact that Level 3 PSA is relatively uncommon. For this reason, tools for performing Level 3 PSA analysis are currently outdated, and competence in Level 3 PSA is difficult to maintain and currently sparse. For these reasons it is difficult to maintain continuity in the development of the analysis.

5.2.5 Analysis considerations

The discussion on the elements of Level 3 PSA, as described in section 4, is based on three cases that aim to cover a spectrum of Level 3 PSA consequences, which address environmental and health metrics as mentioned in section 5.1. These different cases are used to inform which aspects a Level 3 PSA engineer should consider when performing an analysis in order to quantify varied consequences and risk metrics. The delineation between cases A, B and C, which have different target metrics, have been made because the intended use of a study has a great impact upon the necessary input and resources. The three cases are the following:

Environmental risk – Case A: Size of land area with significant Cs contamination

- Both “large” and “small” releases may be of importance. However, if one is interested in specifically Iodine 131 or Cesium 137 contamination, release categories limited to noble gases may be irrelevant (e.g. filtered releases).
- Assessment of countermeasures is not necessarily needed in the analysis.

Individual risk – Case B: Risk of (early) death to maximum exposed individual (individual risk)

- Very significant releases are needed to cause acute health effects. So, one should focus on “large” releases, potentially even for less frequent sequences.
- Analysis with and without protective actions should be performed. The countermeasures that should be included in the analysis are evacuation, sheltering (minimizing ventilation, closing windows and doors, etc.) and Iodine tablet distribution.

Population Risk – Case C: Number of lethal cancers (late effects)

- Even low doses can cause cancer and therefore smaller (less significant) releases are also important.
- Modelling of cancer risk and associated assumptions are important for the results.
- Analysis with and without protective actions should be performed.
    - The short term countermeasures that primarily should be included in the analysis are evacuation, sheltering and distribution of Iodine tablets.
    - In addition, countermeasures affecting the low doses in the intermediate and long-term phases, such as food bans, selection of crops etc., need to be considered.

5.3 Pilot studies

Some conclusions can be drawn from the pilot studies conducted in Finland and Sweden during the Level 3 PSA project.

5.3.1 Finnish pilot

The Finnish pilot is based on a lightweight approach to Level 3 PSA modelling, where probabilistic issues and uncertainties are handled via event trees. The event tree contains weather variables and countermeasures. All uncertainty analyses are also handled on the event tree side. Deterministic calculations – in the pilot's case, population dose estimation – are handled in a separate code, ARANO. The most important benefits of the approach and tools used, as observed in the pilot study, are as follows:

- Modelling and analysis can be performed in a rather short time frame (relatively little work and short computation time), and thus the approach
    - is fast in providing a general view of the consequences of a release and
    - enables easy testing of the impact of various modelling assumptions.
- The event tree tool used – FinPSA Level 2 – is found to be suitable for this kind of modelling, especially because its embedded programming language in practice enables any computation to be conducted that can be expressed by a computer program. This was found to be useful both in event tree computations and in Monte Carlo uncertainty analyses.

No direct downsides of the approach were observed. However, some open questions and issues concerning the applicability and reliability of the approach remain:

- In the pilot study, several simplifying assumptions were made e.g. assuming static weather and simple impact of evacuation. It is not sure whether it is worth aiming at more realistic modelling, where for example weather is dynamic (wind may change direction during the time span considered etc.), or where the evacuation is handled via a simulation model giving the locations of the population as a function of time. Such improvements in modelling might lead to an unmanageably large and complex event tree.
- The accuracy of the approach needs to be studied. The approximations made in modelling cause uncertainties in the results that may be larger than those produced by more traditional Level 3 PSA methods. Comparisons could be made with other consequence analysis codes to see if they are able to produce more accurate results.

On the whole, the approach chosen proved to be suitable for the modelling and analysis to answer limited scope questions concerning Level 3 PSA, and to obtain a general view of consequences in various circumstances. It seems to be best suited to problems where high accuracy of results is not a primary concern. An example is the assessment of long-term radiological consequences, where uncertainties are inevitably high, and high accuracy modelling would therefore be superfluous.

Concerning the case studied in the Finnish pilot, an alternative version of the Fukushima accident, some conclusions can be drawn. The results indicate that the very small radiological consequences, as estimated by the UNSCEAR study on the actual Fukushima accident, are not due to context factors (most of the population had either died in the tsunami or been evacuated because of it before the nuclear accident), or good luck (favorable wind direction in the first days of the release). Instead, the results are something to be expected given general weather conditions in that part of Japan in March and the efficiency of evacuation.

5.3.2 Swedish pilot

The Swedish Pilot study looks at a range of Level 3 PSA consequences; Health, environmental, and economic effects. One of the goals of the Swedish pilot study was to investigate what can be said about release categories that fall above or below a threshold of 100 TBq. In order to be able to investigate the impact of the 100 TBq threshold the only countermeasure that was applied was an evacuation zone with 5 km radius around the plant. This review of varied metrics highlighted how different elements of the Level 2 PSA analysis or weather data are important for different metrics. Some of the notable findings are the following:

- A 100 TBq release criteria provides a reasonably good screening of which release categories are likely to cause health effects. Release categories below 100 TBq are likely to result in limited health effects only, while those exceeding 100 TBq have a notable risk of causing health effects when applying conservative assumptions.
- One clear finding is that for several of the risk metrics investigated, the differences between a release exceeding 100 TBq and those greatly exceeding the 100 TBq (>10 000 TBq) threshold is significant. The contamination metrics were unlikely to cause significant effects unless the threshold is greatly exceeded.
- The study calculates acute health effects and latent health effects in a simplified manner. Even with refined models the uncertainties for health effect quantification can be quite large as is shown in the SOARCA uncertainty analysis [9]. For this reason, it may be recommended to focus Level 3 PSA studies on dose and contamination, especially in simple studies.

It shall be noted that a complete uncertainty analysis, including source term and modelling uncertainties, is not performed in the Swedish pilot study.

5.4 Future work

One important conclusion is that different types of stakeholders have different needs and interests in probabilistic off-site consequence analysis. A consequence of this is that the guidance given in this document is more of a brief overview of the methodology with description of a spectrum of possible analyses rather than detailed specific guidance on how to perform a Nordic PSA Level 3.

The following three areas are identified as especially important to investigate in more detail:

- The integration of Level 1, Level 2, and Level 3 PSA. Neither of the pilot studies could control the development of the Level 1 and Level 2 PSAs which would feed a Level 3 PSA. Therefore, little experience or insight was gained on this important process and the feedback of potential improvements to the Level 1 and 2 PSAs is therefore limited. In particular, the impact of specific initiating events on off-site consequences, and the use of actual severe accident release data are identified as important to explore in more detail.
- Countermeasures were implemented partially in both the Finnish and Swedish pilot studies, more detailed investigation of countermeasures could significantly improve the conclusions and potentially the usefulness of the analysis. More specific, analysis of the impact and logistics of implementing radiation protection measures as outlined in the Flag book [17] is proposed.
- The final and most important future work would be to more deeply investigate uncertainties. The extent of uncertainties and difficulty of performing uncertainty analysis, covering data, model and scope uncertainties, and ultimate limitations of Level 3 PSA results due to uncertainties need further attention. This is true also for Level 1 and Level 2 uncertainties that feed into the Level 3 results and are of importance for end users in decision making.

The Swedish and Finnish pilot studies that have been performed within this research project have given invaluable insights to the guidance given in this document. However, in order to be able to give more specific guidance in general and further recommendations on the above listed aspects it is recommended to perform more site/plant specific pilot studies as a method to study the aspects listed above.

It shall also be noted that a Level 3 PSA is most likely even more beneficial when it comes to evaluating site wide consequences (i.e. multi-unit PSA) and countermeasures in the emergency preparedness. This should especially be the case for sites with different nuclear installations, e.g. power plants of different designs and nuclear fuel repositories (intermediate or final).

6 Project reports

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