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COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

Status of Practice for Level 3 Probabilistic Safety Assessments

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The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally sound and economical use of nuclear energy for peaceful purposes;
- to provide authoritative assessments and to forge common understandings on key issues as input to government decisions on nuclear energy policy and to broader OECD analyses in areas such as energy and the sustainable development of low-carbon economies.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management and decommissioning, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

Committee on the Safety of Nuclear Installations

The Committee on the Safety of Nuclear Installations (CSNI) is responsible for the Nuclear Energy Agency (NEA) programmes and activities that support maintaining and advancing the scientific and technical knowledge base of the safety of nuclear installations.

The Committee constitutes a forum for the exchange of technical information and for collaboration between organisations, which can contribute, from their respective backgrounds in research, development and engineering, to its activities. It has regard to the exchange of information between member countries and safety R&D programmes of various sizes in order to keep all member countries involved in and abreast of developments in technical safety matters.

The Committee reviews the state of knowledge on important topics of nuclear safety science and techniques and of safety assessments, and ensures that operating experience is appropriately accounted for in its activities. It initiates and conducts programmes identified by these reviews and assessments in order to confirm safety, overcome discrepancies, develop improvements and reach consensus on technical issues of common interest. It promotes the co-ordination of work in different member countries that serve to maintain and enhance competence in nuclear safety matters, including the establishment of joint undertakings (e.g. joint research and data projects), and assists in the feedback of the results to participating organisations. The Committee ensures that valuable end-products of the technical reviews and analyses are provided to members in a timely manner, and made publicly available when appropriate, to support broader nuclear safety.

The Committee focuses primarily on the safety aspects of existing power reactors, other nuclear installations and new power reactors; it also considers the safety implications of scientific and technical developments of future reactor technologies and designs. Further, the scope for the Committee includes human and organisational research activities and technical developments that affect nuclear safety.

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List of abbreviations and acronyms

ANVS	Authority for Nuclear Safety and Radiation Protection (Netherlands)
APR	Advanced power reactor
ARANO	Probabilistic consequence analysis computer code
ATD	Atmospheric transport and dispersion
BDBA	Beyond design-basis accident
CAPS	CSNI activity proposal sheet (NEA)
CCDF	Complementary cumulative distribution function
CDF	Core damage frequency OR cumulative distribution function
CFF	Containment failure frequency
CNRA	Committee on Nuclear Regulatory Activities
CNSC	Canadian Nuclear Safety Commission
COSYMA	Probabilistic consequence analysis code
CRPPH	Committee on Radiation Protection and Public Health (NEA)
CSNI	Committee on the Safety of Nuclear Installations (NEA)
DBA	Design-basis accident
DRA	Division of Risk Analysis (United States)
EG-COSTNA	Expert Group on Costs of Nuclear Accidents, Liability Issues and their Impact on Electricity Costs
FN curve	Curve that illustrates frequency of N or more consequences (F) as a function of total number of consequences (N) to express societal risk
HYSPLIT	HYbrid Single Particle Lagrangian Integrated Trajectory
IAEA	International Atomic Energy Agency
IRSN	Institut de radioprotection et de sûreté nucléaire (France)
JAEA	Japan Atomic Energy Agency
JNRA	Nuclear Regulation Authority (Japan)
KAERI	Korea Atomic Energy Research Institute

KI	Potassium iodide
KINS	Korea Institute of Nuclear Safety
KOSCA-OUTPUT	Korea off-site consequence analysis software package – Output post-processing module
LENA_P	Probabilistic version of LENA consequence analysis computer code
LERF	Large early release frequency
LNT	Linear no threshold
LRF	Large release frequency
MAAP	Modular accident analysis program
MACCS	MELCOR accident consequence code system
MELCOR	Severe accident analysis computer code
MTA EK	Hungarian Academy of Sciences, Centre for Energy Research
NDC	Nuclear Development Committee (NEA)
NEA	Nuclear Energy Agency
NPP	Nuclear power plant
NRC	Nuclear Regulatory Commission (United States)
NRG	Nuclear Research and Consultancy Group (Netherlands)
NSSC	Nuclear Safety and Security Commission (Korea)
NUBIKI	Nuclear Safety Research Institute (Hungary)
NUDOS2	NUclide DOSes version 2 code
OECD	Organisation for Economic Co-operation and Development
OSCAAR	Off-site consequence analysis code for atmospheric release in reactor accident
PDF	Probability density function
PSA	Probabilistic safety assessment
QHO	Quantitative health objective
RES	Office of Nuclear Regulatory Research (United States)
SILAM	System for Integrated modeLling of Atmospheric coMposition
SINAC	Simulator software for interactive modelling of environmental consequences of nuclear accidents
SSM	Swedish Radiation Safety Authority
STUK	Radiation and Nuclear Safety Authority (Finland)
TSO	Technical support organisation

VALMA	Atmospheric dispersion and dose assessment code
VTT	VTT Technical Research Centre of Finland (Finland)
WGAMA	Working Group on Analysis and Management of Accidents (NEA)
WGPC	Working Group on Public Communication (NEA)
WGRISK	Working Group on Risk Assessment (NEA)

Executive summary

Introduction

A Level 3 probabilistic safety assessment (PSA) is an assessment of the off-site public risks attributable to a spectrum of possible accident scenarios involving a nuclear installation. In the traditional PSA framework for commercial nuclear power plants, a Level 3 PSA includes three progressive levels of analysis:

1) **Level 1 PSA**: Fuel damage accident or plant damage state frequency analysis.
2) **Level 2 PSA**: Level 1 analysis plus accident progression, containment performance, and radiological release frequency analysis.
3) **Level 3 PSA**: Level 2 analysis plus off-site radiological consequence analysis.

An increasing number of organisations in many countries are pursuing the development and application of Level 3 PSAs for nuclear installations. However, there are several modelling issues and other technical challenges that would benefit from increased information exchange and sharing of methods and practices in this area. For this reason, the Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) approved a proposed Working Group on Risk Assessment (WGRISK) Level 3 PSA activity in June 2015. The objectives of this activity were to: 1) survey member and partner countries to determine current methodological practices in Level 3 PSA; 2) identify common challenges and typical practices; and 3) summarise the results.

Process and methods

A survey instrument was developed to elicit and characterise international practices with respect to selected modelling issues and other technical challenges. The final survey instrument – which was administered by e-mail – consisted of three parts:

1) **Part 1: Respondent information**. Part 1 was designed to obtain respondent information that would allow the task group to determine whether responses differ across different countries or types of organisations.
2) **Part 2: Application of Level 3 PSA**. Part 2 was designed to obtain information about whether and how respondents or organisations use or intend to use Level 3 PSA. Since the purpose or intended use of any PSA can influence multiple aspects of a PSA model, it was judged that responses to Part 2 would provide important contextual information for understanding and evaluating responses to Part 3.
3) **Part 3: Level 3 PSA Modelling issues and technical challenges**. Part 3 was designed to elicit information about whether and how respondents or organisations are addressing specific modelling issues and other technical challenges that practitioners encounter in performing off-site radiological consequence analyses in support of Level 3 PSA applications for nuclear installations.

Survey responses

Representatives from ten WGRISK member countries submitted their responses to the survey. Representatives from diverse organisation types within each country often collaborated to submit a joint response to the survey, thereby resulting in multiple organisation types being represented in a single survey response. Fifty per cent of survey respondents represented regulatory authorities, with the other 50% representing a mix of academic/research, industry, and technical support organisations (TSOs).

Two out of the ten of surveyed countries indicated that there is a requirement to perform Level 3 PSAs for nuclear installations in their country, while 7 out of the 10 surveyed countries indicated that they are currently performing, planning to perform, or considering performing Level 3 PSAs for nuclear installations: 1) Finland (performing); 2) Hungary (planning to perform); 3) Japan (performing); 4) Korea (required and performing); 5) Netherlands (required and performing); 6) Sweden (considering performing); and 7) United States (performing).

Analysis of submitted survey responses produced many insights and resulted in the identification of typical practices and common technical challenges or limitations across several aspects of Level 3 PSA. These typical practices and common technical challenges or limitations are summarised in the sections that follow.

Typical practices

Typical practices identified through analysis of submitted survey responses can be grouped into four broad categories: 1) Level 3 PSA applications; 2) Level 3 PSA scope considerations; 3) modelling practices; and 4) presentation of risk results and uncertainties in risk results.

Level 3 PSA applications

- Among countries that are currently performing, planning to perform, or considering performing Level 3 PSAs for nuclear installations, five broad types of Level 3 PSA applications were identified: 1) applied research; 2) comparison with risk acceptance criteria, safety goals, or other quantitative objectives; 3) evaluation of protective action effectiveness to inform emergency preparedness and response guidelines; 4) information for the development of severe accident management plans; and 5) environmental assessments.

- Although survey respondents generally acknowledged the potential benefit of using results from Level 3 PSAs to inform nuclear installation siting decisions, most indicated that Level 3 PSAs are not being used when considering such applications. Results from deterministic design-basis accident (DBA) dose consequence assessments are typically used to demonstrate that doses following postulated fission product releases are within prescribed limits.

- Results from deterministic analyses provide the primary analytical basis for establishing emergency planning or protective action zones. If Level 3 PSA insights are used in developing requirements for the sizes and boundaries of emergency planning or protective action zones, such risk insights are typically considered as a complement to results from deterministic analyses as part of a risk-informed approach.

Level 3 PSA scope considerations

- Since the objective of Level 3 PSA is to assess the risks to the off-site public that are attributable to a spectrum of possible accident scenarios involving a nuclear installation, countries that perform Level 3 PSA typically exclude the on-site worker population.

- Radiological releases to aqueous pathways are typically screened out and excluded from Level 3 PSAs for nuclear installations that have the potential for airborne radiological releases, since airborne releases are expected to dominate radiological health risks.

- Among most countries that perform Level 3 PSAs, economic models are generally used to estimate the costs attributed to the implementation of modelled protective actions to reduce the radiological dose to the off-site public from the accidental release of radiological materials.

Modelling practices

- Level 2 PSA analysts typically work closely with Level 3 PSA analysts to: 1) define radiological release categories; 2) select a representative accident sequence for each radiological release category to estimate source term characteristics; and 3) perform sensitivity or uncertainty analyses on release categories or release fractions to evaluate the impact on risk results.

- The size of the modelled region and spatial intervals around a site are typically application- and site-specific. In general, a larger number of spatial intervals with finer resolution grid elements are used for the region close to the site, with the number of spatial intervals and grid resolution decreasing as the distance from the site increases.

- Most countries that perform Level 3 PSA: 1) use site-specific data that includes hourly observations for an entire year; 2) implement code-specific weather binning and sampling strategies to account for the impact of variability in weather conditions over time; and 3) use site-specific information sources to model various other site characteristics.

- A range of protective actions is typically modelled for the early (emergency), intermediate, and late (recovery) phases of accident response. Most countries that model protective actions typically do not: 1) model population groups with different protective action behaviours; or 2) use models to account for the probabilities of success or failure of protective actions.

- In general, a range of early radiological health effects are estimated using deterministic models. Fatal and non-fatal cancers involving multiple organs generally represent the latent radiological health effects that are estimated using stochastic models. Respondents from all countries that perform Level 3 PSA indicated they use a default dose-response model based on the linear no-threshold (LNT) hypothesis to estimate the excess cancer risk attributable to radiological doses caused by accidental releases from nuclear installations.

Presentation of risk results and uncertainties in risk results

- In general, risk results and uncertainties in risk results are presented using a variety of formats and graphical displays.

Common technical challenges or limitations

Analysis of submitted survey responses also resulted in identification of common technical challenges or limitations:

- Among countries that are not currently performing Level 3 PSAs for nuclear installations, four potential barriers to performing Level 3 PSA were identified: 1) absence of a technical or legal framework to perform Level 3 PSA; 2) large uncertainties in Level 3 PSA results, especially when combined with uncertainties propagated from Level 1 and Level 2 analyses; 3) limited expected benefit in terms of potential back fits or safety improvements; and 4) additional resources required to perform detailed off-site radiological consequence analyses.

- Most computer codes for performing probabilistic analyses of off-site radiological consequences: 1) use relatively simple atmospheric transport and dispersion (ATD) models; and 2) either do not support terrain modelling or rely on the use of simple terrain models.

- None of the available probabilistic consequence analysis computer codes can directly perform risk characterisation by mathematically combining the radiological release category frequencies with the corresponding conditional consequences.

- Most countries that perform Level 3 PSA do not have a communication plan or guidelines for risk communication. A notable exception is the US Nuclear Regulatory Commission (NRC) that has developed guidance documents aiming to enhance risk communication with internal and external stakeholders.

Recommendation for future activities

A key finding from analysis of submitted survey responses is that many typical practices or common technical challenges identified as part this WGRISK activity are driven by the limitations or capabilities of available probabilistic consequence analysis codes. This finding underscores the potential need for future research and leads to the following recommendation stemming from this WGRISK activity.

Recommendation: Consider performing a study to benchmark available probabilistic consequence analysis computer codes used for Level 3 PSA applications.

The NEA previously performed similar studies to benchmark probabilistic consequence analysis codes. Yet the most recent of these studies was completed more than 20 years ago. Since probabilistic consequence analysis codes have evolved considerably during the past two decades, it seems reasonable to perform a follow-on benchmarking study to compare more recent versions of the computer codes. However, many of these probabilistic consequence analysis codes are currently being enhanced to address various modelling issues and technical challenges, some of which were brought into focus by the 2011 accident at the Fukushima Daiichi nuclear power plant in Japan. Based on these ongoing code enhancements, a logical path forward may to be to delay any future benchmarking study (if one is to be undertaken) until after these enhancements have been implemented, with verification and validation testing completed, as appropriate.

Acknowledgements

This project was carried out by the Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) Working Group on Risk Assessment (WGRISK). Technical staff in the US Nuclear Regulatory Commission (USNRC)/Office of Nuclear Regulatory Research (RES)/Division of Risk Analysis (DRA) led this project, with support from technical staff assigned to other USNRC offices, as well as from individuals from other member countries who volunteered to serve on the core task group and expanded task group for this WGRISK activity.

As the lead organisation for this project, the USNRC established an internal working group that: 1) identified and prioritised specific modelling issues and other technical challenges in performing Level 3 PSA off-site radiological consequence analyses that could benefit from information exchange and sharing of methods and practices across the international PSA community; 2) developed an initial draft of the survey instrument that would be used to elicit and characterise international practices with respect to selected modelling issues and other technical challenges; and 3) developed an example response that would be distributed with the final survey instrument to provide survey respondents with guidance by way of an example on the scope and level of detail that would be expected for responses to each of the survey items. The following USNRC staff members served on or supported this working group: Keith Compton, Kevin Coyne, Elijah Dickson, Daniel Hudson, Tony Nakanishi, Donald Palmrose, Jason Schaperow, and Nathan Siu.

The following individuals represented their respective organisations and countries as members of the core task group that was responsible for project planning, survey design, survey administration, analysis of submitted survey responses, and development of this final report:

Name	**Organisation**	**Country**
Usha Menon Chantal Morin	Canadian Nuclear Safety Commission (CNSC)	Canada
Tams Pázmándi	Hungarian Academy of Sciences, Centre for Energy Research (MTA EK)	Hungary
Yu Maruyama	Japan Atomic Energy Agency (JAEA)	Japan
Kwang-Il Ahn Seok-Jung Han	Korea Atomic Energy Research Institute (KAERI)	Korea
J.L. (Hans) Brinkman J.B. (Jacques) Grupa	Nuclear Research and Consultancy Group (NRG)	Netherlands
Daniel Hudson (Project lead)	US Nuclear Regulatory Commission (USNRC)	United States

The following individuals represented their respective organisations or countries as members of the expanded task group and added valuable contributions to this project by reviewing and providing constructive feedback on the draft survey instrument or draft final report:

Name	Organisation	Country
Shahen Poghosyan	International Atomic Energy Agency (IAEA)	N/A
Aurélie Lorin	NEA/Committee on Nuclear Regulatory Activities (CNRA)/Working Group on Public Communication (WGPC)	
Neil Higgins	NEA/Nuclear Development Committee (NDC)/Expert Group on Costs of Nuclear Accidents, Liability Issues and their Impact on Electricity Costs (EG-COSTNA)	
Ilkka Karanta Markus Porthin	VTT Technical Research Centre of Finland	Finland
Attila Bareith	NUBIKI Nuclear Safety Research Institute	Hungary
Per Hellström	Swedish Radiation Safety Authority (SSM)	Sweden

The members of the core task group wish to acknowledge and extend their sincere gratitude to the countries, organisations, and individuals who supported this project by providing high-quality responses to the survey and by responding to subsequent correspondence to obtain additional information, as needed. These countries, organisations, and individuals are identified in each of the submitted survey responses provided in Appendix C.

The group also wishes to acknowledge the leadership and vision for this project that Dr Kevin Coyne provided throughout the conception, development, and implementation phases of this project. Lastly, this task would not have been possible without the significant contributions, leadership, and co-ordination provided by the task project leader, Dr Daniel Hudson. The valuable insights about current methodological practices in Level 3 PSA that were developed from this project would not have been realised without Dr Hudson's substantial efforts.

1. INTRODUCTION

1.A. Background and motivation

The mission of the Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) Working Group on Risk Assessment (WGRISK) is to advance understanding of probabilistic safety assessment (PSA) and to enhance its utilisation for: 1) improving the safety of nuclear installations; 2) improving the design and operation of nuclear installations; and 3) increasing regulatory effectiveness through risk-informed approaches. In support of this mission, WGRISK routinely collects and shares information among member countries regarding PSA methods and practices used to estimate nuclear accident risks and to identify significant risk contributors in support of nuclear installation applications.

A Level 3 PSA is an assessment of the off-site public risks attributable to a spectrum of possible accident scenarios involving a nuclear installation. In the traditional PSA framework for commercial nuclear power plants, a Level 3 PSA includes three progressive levels of analysis: 1) core damage accident or plant damage state frequency analysis; 2) accident progression, containment performance, and radiological release frequency analysis; and 3) off-site radiological consequence analysis. A Level 1 PSA includes only the first level of analysis and estimates the frequency of core damage accidents or plant damage states and identifies significant contributors to these frequencies. A Level 2 PSA includes the first and second levels of analysis and - in addition to Level 1 PSA risk metrics - estimates the frequencies of defined radiological release categories and identifies significant contributors to these frequencies. Finally, a Level 3 PSA includes all three levels of analysis and - in addition to Level 2 PSA metrics - estimates the frequencies of off-site public health, environmental, and economic consequences attributable to the set of radiological release categories and corresponding source terms defined in the Level 2 analysis. Compared to Level 1 and Level 2 PSAs, a Level 3 PSA thus represents a more complete characterisation of the off-site public risks attributable to a spectrum of possible accident scenarios involving a nuclear installation and provides an important input to cost-benefit analyses used to evaluate proposed risk management options. Although the term Level 3 PSA has been used in some cases to refer only to the off-site radiological consequence analysis element, the focus of this activity is on off-site radiological consequence analyses that are performed as part of an integrated PSA that includes all three analysis levels.

An increasing number of organisations in many countries are pursuing the development and application of Level 3 PSAs for nuclear installations [1,2]. However, the experience of several WGRISK member countries indicates there are several modelling issues and other technical challenges that would benefit from increased information exchange and sharing of methods and practices in this area. Such co-operation could: 1) help harmonise international methods and practices; and 2) assist organisations in developing new Level 3 PSAs or in improving upon existing Level 3 PSAs.

1.B. Objectives

Given the increasing interest in Level 3 PSA and the potential benefits of information exchange and sharing of methods and practices in this area, the CSNI approved a proposed WGRISK Level 3 PSA

activity in June 2015. The approved CSNI Activity Proposal Sheet (CAPS) for this activity is provided in Appendix A.

The objectives of this activity were to: 1) survey member and observer countries to determine current methodological practices in Level 3 PSA; 2) identify common challenges and typical practices; and 3) summarise the results of this activity in a final task report. This report represents the final product of this CSNI-approved WGRISK Level 3 PSA activity.

1.C. Task group composition and co-ordination

Representatives from six WGRISK member countries comprised the core task group for this activity, with the United States serving as the lead country. Countries and organisations that contributed to the core task group included:

1. Canada: Canadian Nuclear Safety Commission (CNSC)
2. Hungary: Hungarian Academy of Sciences, Centre for Energy Research (MTA EK)
3. Japan: Japan Atomic Energy Agency (JAEA)
4. Korea: Korea Atomic Energy Research Institute (KAERI)
5. Netherlands: Nuclear Research and Consultancy Group (NRG)
6. United States: US Nuclear Regulatory Commission (USNRC)

The expanded task group included core task group members and other interested WGRISK participants who are familiar with Level 3 PSA, and additional NEA organisations. In particular, since Level 3 PSA involves multiple disciplines and areas of interest to other groups, representatives from five groups within and beyond the NEA were also engaged in this activity to ensure effective co-ordination between related activities. These groups included:

1. NEA/CSNI/Working Group on Analysis and Management of Accidents (WGAMA)
2. NEA/Committee on Nuclear Regulatory Activities (CNRA)/Working Group on Public Communication (WGPC)
3. NEA/Committee on Radiation Protection and Public Health (CRPPH)
4. NEA/Nuclear Development Committee (NDC)/Expert Group on Costs of Nuclear Accidents, Liability Issues and their Impact on Electricity Costs (EG-COSTNA)
5. International Atomic Energy Agency (IAEA)

1.D. Intended audience

The intended audience for this report includes a broad spectrum of individuals and entities across the international community with an interest in the assessment of off-site public risks attributable to a spectrum of possible accident scenarios involving a nuclear installation

1.E. Report structure

The main report is organised into five chapters and is supported by three appendices. This chapter introduces: 1) the background and motivation for this CSNI-approved activity; 2) the objectives of this activity; 3) the composition of task groups assigned to this activity, along with identification of other groups within and beyond NEA that were engaged in this activity to ensure effective co-ordination; 4) the intended audience for this report; and 5) the structure of this report.

Chapter 2 summarises the process and methods used to complete this activity and includes five sections that respectively address: 1) design of a survey instrument that was used to elicit and characterise international practices with respect to selected modelling issues and other technical

challenges; 2) survey administration; 3) analysis of submitted survey responses to develop results and insights; 4) a planned expanded task group workshop; and 5) development of the final task report.

Chapter 3 summarises results and insights obtained from analysis of submitted responses to the survey that was administered by e-mail to elicit and characterise international practices with respect to selected modelling issues and other technical challenges. Results and insights are grouped by survey part and associated Level 3 PSA technical element.

Chapter 4 identifies and summarises key conclusions from this activity, including typical practices and common technical challenges or limitations that were identified through analysis of submitted survey responses. The chapter also provides recommendations for future activities that follow from these conclusions.

Chapter 5 identifies the list of reference documents cited throughout the main report.

Appendix A provides the CSNI-approved CAPS for this WGRISK activity.

Appendix B provides the final survey instrument that was distributed by e-mail to WGRISK member and observer countries to elicit and characterise international practices with respect to selected modelling issues and other technical challenges.

Appendix C provides the ten survey responses that were submitted by representatives of ten countries in support of this activity. Other than changing the formatting to be consistent with the rest of the report and correcting any obvious typographical errors, the detailed responses that survey respondents submitted have been preserved in their entirety.

2. PROCESS AND METHODS

This chapter summarises the process and methods used to complete this activity and includes five sections that respectively address: 1) design of a survey instrument that was used to elicit and characterise international practices with respect to selected modelling issues and other technical challenges; 2) survey administration; 3) analysis of submitted survey responses to develop results and insights; 4) a planned expanded task group workshop; and 5) development of the final task report.

2.A. Survey design

As the lead organisation for this activity, the US Nuclear Regulatory Commission (NRC) established an internal working group comprised of technical staff with diverse expertise pertaining to development and application of the off-site radiological consequence analysis element of a Level 3 PSA. The purposes of this working group were to: 1) identify and prioritise specific modelling issues and other technical challenges in performing Level 3 PSA off-site radiological consequence analyses that could benefit from information exchange and sharing of methods and practices across the international PSA community; 2) develop an initial draft of the survey instrument that would be used to elicit and characterise international practices with respect to selected modelling issues and other technical challenges; and 3) develop an example response that would be distributed with the final survey instrument to provide survey respondents with guidance by way of an example on the scope and level of detail that would be expected for responses to each of the survey items.

The draft survey instrument that the USNRC working group developed was initially distributed by e-mail to members of both the core task group and expanded task group for review in September 2015. Two specific questions were posed to task group members for consideration as they performed their review of the draft survey instrument:

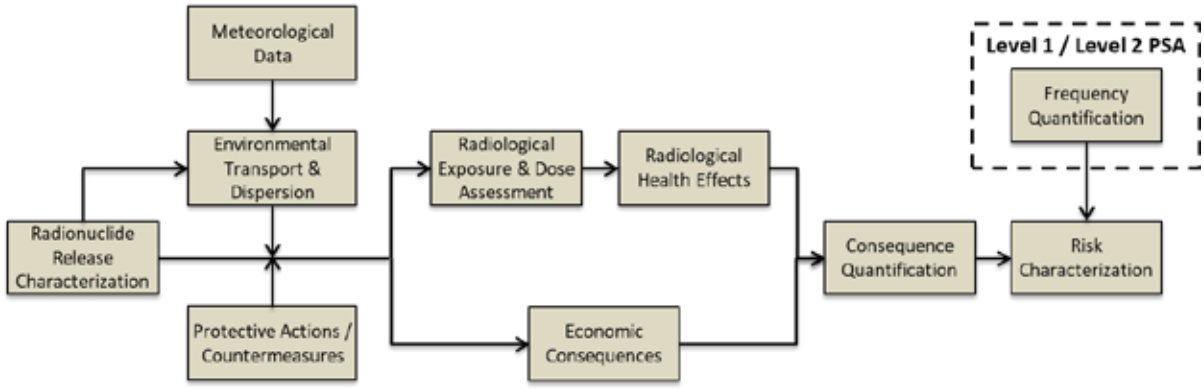
1. Does the survey fail to address any important Level 3 PSA modelling issues or technical challenges that you believe should be included?
2. Are there any questions (or groups of questions) that you believe can be removed from the survey to reduce the burden on respondents and to maximise our response rate?

From October 2015 to March 2016, the draft survey instrument was then iteratively revised to address the feedback received from members of both the core task group and expanded task group after their review. In April 2016, after completion of these revisions and before distribution to WGRISK member and observer countries, the survey instrument and example response that the USNRC working group had developed were distributed by e-mail for review to representatives of the following five groups within and beyond the NEA that were identified in Chapter 1: 1) NEA/CSNI/WGAMA; 2) NEA/CNRA/WGPC; 3) NEA/CRPPH; 4) NEA/NDC/EG-COSTNA; and 5) IAEA. No additional feedback was received after this review.

The final survey instrument, which is provided in Appendix B, included three parts. Part 1 (Respondent information) was designed to obtain respondent information that would allow the task group to determine whether responses differ across different countries or types of organisations. Part 2 (Application of Level 3 PSA) was designed to obtain information about whether and how respondents or organisations use or intend to use Level 3 PSA. Since the purpose or intended use of any PSA can

influence multiple aspects of a PSA model (e.g. scope, level of detail, assumptions, or other modelling choices), it was judged that responses to Part 2 would provide important contextual information for understanding and evaluating responses to Part 3. Part 3 (Level 3 PSA Modelling Issues and Technical Challenges) was designed to elicit information about whether and how respondents or organisations are addressing specific modelling issues and other technical challenges that practitioners encounter in performing off-site radiological consequence analyses in support of Level 3 PSA applications for nuclear installations. To facilitate identification and organisation of this information, survey items in Part 3 were grouped by the associated major technical elements or tasks that typically comprise such off-site radiological consequence analyses. These technical elements and the relationships between them are illustrated below in Figure 1.

Figure 1. Typical Level 3 PSA Technical Elements and Their Relationships



Note: This figure illustrates technical elements that typically comprise off-site radiological consequence analyses performed as part of Level 3 PSAs and the relationships between elements. Survey items pertaining to Level 3 PSA modelling issues and technical elements were grouped by associated technical element to facilitate identification and organisation of this information.

2.B. Survey administration

The WGRISK Secretariat distributed the final survey instrument and example response to WGRISK member and observer countries by e-mail in May 2016. Completed survey responses were requested to be submitted directly to the USNRC point of contact for the activity by July 2016. However, based on requests from individual survey respondents who had expressed an intent to complete the survey, but who needed additional time to do so, the deadline for accepting completed survey responses was extended to December 2016.

2.C. Analysis of submitted survey responses

From January 2017 to March 2017, the core task group analysed submitted survey responses to summarise results and to develop key insights. The first step was to develop a preliminary summary of survey responses to: 1) determine whether and to what extent a planned expanded task group workshop described in the next section would be needed; and 2) identify and prioritise the key technical challenges or modelling issues that would be addressed in the planned expanded task group workshop, if the group determined a workshop was needed or would add significant value.

Once it was determined whether an expanded task group workshop would be held, the core task group performed more detailed analysis of submitted survey responses to summarise results and to develop key insights derived from items in each part of the survey.

2.D. Planned expanded task group workshop

As described in the CAPS provided in Appendix A, the original plan was to use the survey to generate results and insights for use in a subsequent expanded task group workshop that would provide a forum for more detailed discussions and information exchange on key Level 3 PSA topics. However, it was later determined that the time and resources needed to conduct an expanded task group workshop would not be justified for two main reasons: 1) the collective information provided in the submitted survey responses was generally judged to be of sufficient scope and level of detail to satisfy the objectives of this activity; and 2) where additional information was needed, this information could be collected by direct e-mail or telephone correspondence with individual survey respondents. The task group thus agreed that the additional benefit that could be obtained from an expanded task group workshop did not warrant the use of additional time and resources.

2.E. Final task report

Based on a review and analysis of the submitted survey responses, the core task group developed this final task report to summarise the results. This final task report represents the final deliverable for this WGRISK activity.

3. SURVEY RESULTS AND INSIGHTS

This chapter summarises results and insights obtained from analysis of submitted responses to the survey that was administered by e-mail to elicit and characterise international practices with respect to selected modelling issues and other technical challenges. Results and insights are grouped by survey part and – for the Level 3 PSA modelling issues and technical challenges addressed in Part 3 – by the associated Level 3 PSA technical element.

3.A. Part 1: Respondent Information

This section summarises results and insights from Part 1 of the survey, which was designed to obtain respondent information that would allow the task group to determine whether responses differ across different countries or types of organisations.

3.A.1. Countries represented

Representatives from ten WGRISK member countries submitted a total of ten responses to the survey: one survey response per country. Table 1 summarises the countries represented, along with the organisation types that were represented by survey respondents.

All ten survey responses that were submitted in support of this activity, including the example response developed by the USNRC, are provided in Appendix C to this report.

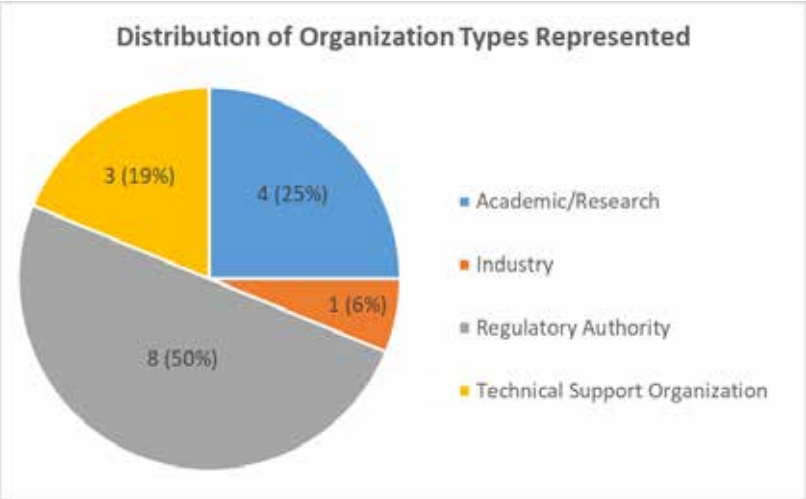
Table 1. Countries and Organisation Types Represented by Survey Respondents

Country	Organisation Types Represented by Survey Respondents
Belgium	Technical Support Organisation (Bel V)
	Industry (Tractebel)
Canada	Regulatory Authority (CNSC)
Finland	Academic/Research (VTT)
	Regulatory Authority (STUK)
Hungary	Academic/Research (MTA EK)
	Technical Support Organisation (NUBIKI)
	Regulatory Authority (HAEA)
Japan	Academic/Research (JAEA)
Korea	Academic/Research (KAERI)
	Regulatory Authority (KINS)
Netherlands	Technical Support Organisation (NRG)
	Regulatory Authority (ANVS)
Sweden	Regulatory Authority (SSM)
Switzerland	Regulatory Authority (ENSI)
United States	Regulatory Authority (USNRC)

3.A.2. Organisation types represented

Table I shows that representatives from multiple organisation types often collaborated to submit a joint response to the survey, thereby resulting in multiple organisation types being represented in a single survey response. Figure 2 further illustrates that survey respondents represented a diverse set of organisation types, with 50% representing regulatory authorities and the other 50% representing a mix of academic/research, industry, and technical support organisations (TSOs).

Figure 2. Distribution of Organisation Types Represented



Note: Survey respondents represented a diverse set of organisation types, with 50% representing regulatory authorities and the other 50% representing a mix of academic/research, industry, and TSOs.

3.B. Part 2: Application of Level 3 PSA

This section summarises results and insights from Part 2 of the survey, which was designed to obtain information about whether and how respondents or organisations use or intend to use Level 3 PSA. Since the purpose or intended use of any PSA can influence multiple aspects of a PSA model (e.g. scope, level of detail, assumptions, or other modelling choices), it was judged that responses to Part 2 would provide important contextual information for understanding and evaluating responses to Part 3 of the survey (Level 3 PSA Modelling Issues and Technical Challenges).

3.B.1. Level 3 PSA Requirements

20% (2 out of 10) of countries surveyed indicated that there is a requirement to perform Level 3 PSAs for nuclear installations in their country. The two countries that require the performance of Level 3 PSAs for nuclear installations are: 1) Korea and (2) Netherlands.

Korea requires applicants for an operating licence for new nuclear power plants to perform a Level 3 PSA that demonstrates two safety goal quantitative health objectives (QHOs) are satisfied:

1. Early fatality risk from nuclear accidents shall not exceed 0.1% of the total early fatality risk arising from other accidents (4E-07 per year).
2. Cancer fatality risk from nuclear accidents shall not exceed 0.1% of the total cancer fatality risk (1E-06 per year).

Emergency response protective actions (e.g. evacuation, sheltering, potassium iodide) are not credited in determining whether these risk acceptance criteria are satisfied. However, the regulatory authority, Nuclear Safety and Security Commission (NSSC), recently required confirmation of effectiveness of emergency response protective actions in the licensing process for Shin-Hanul Units 1 and 2 (Advanced Power Reactor (APR)-1400 series). The licensee is thus assessing the effectiveness of relevant protective actions in sensitivity analyses.

Netherlands requires a Level 3 PSA to show that a nuclear installation containing fissile materials and/or ores complies with certain risk acceptance criteria when applying for a licence (or a change of the license) to establish, construct, commission, operate, or decommission the installation. These risk acceptance criteria include:

1. The probability that an individual who resides permanently and unprotected outside the facility grounds dies from a beyond design-basis accident (BDBA) is less than 1E-06 per year. To calculate this individual risk, the characteristics of this individual (e.g. age, location, habits) are chosen to result in the worst-case scenario that would still be reasonable, but does not have to be a real person. In practice, this is a 1-year-old infant residing next to the edge of the nuclear installation. In determining the total dose resulting from a BDBA, long-term (stochastic) effects for at least 50 years are considered, including ongoing exposures over that time-period due to contamination of the environment.
2. The probability that a BDBA leads to at least 10 direct fatalities within a few weeks, due to deterministic effects, is less than 1E-05 per year, and the probability of n times more fatalities is n^2 times smaller. For example, the probability of at least 100 fatalities is less than 1E-07 per year and the probability of at least 1 000 fatalities is less than 1E-09 per year. To calculate this group (societal) risk, the actual population distribution around the nuclear installation is to be considered.

Like Korea, emergency response protective actions are also not credited in determining whether these risk acceptance criteria are satisfied in the Netherlands.

3.B.2. Use of Level 3 PSA

70% (7 out of 10) countries surveyed indicated they are currently performing, planning to perform, or considering performing Level 3 PSAs for nuclear installations. These seven countries include: 1) Finland (performing); 2) Hungary (planning to perform); 3) Japan (performing); 4) Korea (required and performing); 5) Netherlands (required and performing); 6) Sweden (considering performing); and 7) United States (performing).

Table 2 summarises uses of Level 3 PSA across all countries surveyed, including any specific applications that were identified by survey respondents. Specific risk metrics for each type of application and any corresponding criteria used to evaluate results are identified in the individual responses provided in Appendix C. In general, risk metrics used are driven by the objectives of the Level 3 PSA. Commonly used risk metrics include: 1) population dose risk; 2) individual risk of radiological health effects (e.g. individual early fatality risk, individual latent cancer fatality risk); and 3) group (societal) risk of radiological health effects (e.g. risk of exceeding a specified number of fatalities).

Table 2. Uses of Level 3 PSA across all countries surveyed

Country	Use of Level 3 PSA	Applications
Belgium	No	Not applicable.
Canada	No	Not applicable.
Finland	Yes	Applied research.
Hungary	Planning	Applied research.
Japan	Yes	Comparison with risk acceptance criteria, safety goals, or other quantitative objectives. Evaluation of protective action effectiveness to inform emergency preparedness and response guidelines.
Korea	Yes Required	Comparison with risk acceptance criteria, safety goals, or other quantitative objectives.
		Inform development of severe accident management plans.
Netherlands	Yes Required	Comparison with risk acceptance criteria, safety goals, or other quantitative objectives.
Sweden	Considering	Applied research.
Switzerland	No	Not applicable.
United States	Yes	Comparison with risk acceptance criteria, safety goals, or other quantitative objectives as part of regulatory or backfit analyses.
		Environmental assessments.
		Applied research.

Among countries that are not currently performing, planning to perform, or considering performing Level 3 PSAs for nuclear installations (Belgium, Canada, Switzerland), four potential barriers to performing Level 3 PSA were identified: 1) absence of a technical or legal framework to perform Level 3 PSA; 2) large uncertainties in Level 3 PSA results, especially when combined with uncertainties propagated from Level 1 and Level 2 analyses; 3) limited expected benefit in terms of potential back fits or safety improvements; and 4) additional resources required to perform detailed off-site radiological consequence analyses. These countries further identified a range of alternative

methods to estimating off-site public risks attributable to accidental releases of radiological materials from nuclear installations. These alternative methods include use of: 1) surrogate metrics based on results from Level 1 PSAs (e.g. core damage frequency (CDF)) and Level 2 PSAs (e.g. large early release frequency (LERF), large release frequency (LRF)); 2) deterministic design-basis accident (DBA) dose consequence assessments with conservative assumptions to demonstrate compliance with prescribed dose limits; and 3) dose consequence assessments to demonstrate compliance with a range of specified dose limits that vary with the expected frequency of events.

3.B.3. Siting of nuclear installations

Although survey respondents generally acknowledged the potential benefit of using results from Level 3 PSAs to inform decisions with respect to siting of nuclear installations, most indicated that Level 3 PSAs are not being used to support such applications. Instead, respondents stated that results from deterministic DBA dose consequence assessments are used to demonstrate that doses to individuals located at any point along the boundaries of defined areas around a nuclear installation for specified periods of time following postulated fission product releases are within prescribed limits.

3.B.4. Establishment of emergency planning or protective action zones

Practices appear to vary with respect to consideration of Level 3 PSA insights in establishing the sizes and boundaries of emergency planning or protective action zones, with some countries actively re-evaluating the technical basis for their existing zones. Some countries relied solely on deterministic DBA analyses and various dose-distance criteria in establishing existing zones. Others supplemented such analyses with risk insights from previous Level 3 PSAs (e.g. WASH-1400) or considered the effect of weather variability and assessed the probability of exceeding specified dose thresholds for implementing various protective actions.

A typical practice with respect to establishing emergency planning or protective action zones is that results from deterministic analyses provide the primary analytical basis for established zones. If Level 3 PSA insights are used in developing requirements for the sizes and boundaries of emergency planning or protective action zones, such risk insights are typically considered as a complement to results from deterministic analyses as part of a risk-informed approach.

3.B.5. Development of safety goals or risk acceptance criteria

A previous WGRISK project addressed the development and application of probabilistic risk criteria and safety goals in many countries [3]. Among countries with defined risk acceptance criteria, safety goals, or other quantitative objectives that participated in the current project (Japan, Korea, Netherlands, United States), survey respondents generally indicated that results from Level 3 PSAs are used to evaluate compliance with or attainment of established criteria related to radiological health effects, rather than to inform development of the established criteria. In particular, for those countries that identified criteria related to individual and/or group (societal) risk of radiological health effects, two respondents (Korea and United States) indicated that these criteria reflect a value judgement about the level of incremental (additional) risk to the public from nuclear installation operations relative to background risk that was judged to be acceptable; in both countries, a level of 0.1% of the background risk was selected to establish QHOs for individual early fatality risk and individual latent cancer fatality risk.

Although results from Level 3 PSAs were not explicitly used to develop these QHOs, respondents from three out of these four countries (Japan, Korea, United States) indicated that Level 3 PSA results are used in defining and establishing criteria with respect to subsidiary or surrogate risk metrics that are calculated as part of Level 1 PSAs (e.g. CDF) or Level 2 PSAs (e.g. LERF, LRF, containment failure frequency (CFF)) to support risk-informed decision making.

3.B.6. Other applications of off-site radiological consequence analyses

Survey respondents identified a wide range of other uses of off-site radiological consequence analyses that are performed independently of Level 3 PSA. Notable examples of these other uses include: 1) deterministic DBA dose consequence assessments used to demonstrate that doses to individuals located at any point along the boundaries of defined areas around a nuclear installation for specified periods of time following postulated fission product releases are within prescribed limits as part of siting or emergency planning applications; (2) decision support for real-time response to nuclear incidents or accidents, including dose projections to inform or evaluate protective action recommendations; and (3) environmental assessments in support of licensing applications.

3.C. Part 3: Level 3 PSA modelling issues and technical challenges

3.C.1. General

3.C.1.1. Computer codes for off-site radiological consequence analysis

Several computer codes for performing off-site radiological consequence analyses were identified in survey responses. Some of these identified codes are used exclusively for applications not related to Level 3 PSA that were described in Section 3.B.6. Table 3 lists the probabilistic consequence analysis codes that respondents from each country identified for use as part of Level 3 PSA. A key finding from the analysis of survey responses is that many typical practices identified as part this WGRISK activity are driven by the limitations or capabilities of these probabilistic consequence analysis codes. This finding is described in more detail in Chapter 4.

Table 3. Probabilistic consequence analysis computer codes used for Level 3 PSA applications

Country	Level 3 PSA Probabilistic Consequence Analysis Computer Code(s)
Belgium	Not applicable.
Canada	Not applicable.
Finland	ARANO MELCOR Accident Consequence Code System (MACCS)
Hungary	COSYMA
Japan	Off-Site Consequence Analysis code for Atmospheric Release in reactor accident (OSCAAR)
Korea	MELCOR Accident Consequence Code System (MACCS)
Netherlands	COSYMA NUDOS2 MELCOR Accident Consequence Code System (MACCS)
Sweden	LENA_P
Switzerland	Not applicable.
United States	MELCOR Accident Consequence Code System (MACCS)

3.C.1.2. Computer codes for risk characterisation

A notable finding from analysis of the survey responses is that none of these probabilistic consequence analysis codes can directly perform risk characterisation. Risk characterisation is the

process of developing and evaluating risk triplets comprised of accident scenarios, frequencies, and conditional consequences to produce: 1) qualitative insights about accident scenarios and significant risk contributors; and 2) quantitative estimates for risk metrics of interest. These quantitative estimates are typically calculated by combining estimates of radiological release category frequencies from Level 2 PSA analyses with conditional consequence metric estimates from Level 3 PSA analyses for each radiological release category.

Among countries that perform Level 3 PSA, three countries (Korea, Netherlands, and United States) indicated that risk characterisation is performed by post processing of results using an application external to the probabilistic consequence analysis code (e.g. spread sheet software or graphical user interface software). For example, Korea (KAERI) utilises KOSCA-OUTPUT (Korea off-site consequence analysis software package – Output post-processing module) for risk characterisation. KOSCA-OUTPUT is a graphical user interface module that is used to post-process the consequence results from MACCS and release frequency data from the Level 2 PSA to calculate selected risk metrics.

3.C.1.3. Scope of Level 3 PSAs

Among countries that perform Level 3 PSA, survey respondents identified a range of potential accident sequence initiating event hazards that would be considered as part of the Level 1 analysis in establishing the scope of a Level 3 PSA. Some individual responses in Appendix C identify specific initiating event hazards or hazard groups that are typically considered. Although some countries do not have experience with considering external events in the context of Level 3 PSA, most countries indicated that both internal and external initiating event hazards should be considered.

Practices vary with respect to the treatment of potential correlation between the causes of accident sequence initiating events and off-site phenomenological and consequence modelling. Finland indicated it accounts for this potential correlation and cited an example of the correlation between the time of year in which an accident sequence is initiated and use of appropriate meteorological data for that time of year. Netherlands indicated it only accounts for this potential correlation if failure to account for the correlation would result in a non-conservative estimate of off-site consequences. Korea and United States described a partial treatment of this correlation by developing unique consequence models for seismic events that impact protective action parameters (e.g. delay time to evacuation, evacuation speed, shielding parameters). Other countries stated that this potential correlation is not treated.

3.C.1.4. Ongoing research and development

Survey respondents identified a range of ongoing research and development activities pertaining to Level 3 PSA and off-site radiological consequence analysis. Specific activities are identified in the individual responses provided in Appendix C. Notable examples of research and development areas that are being addressed through ongoing efforts include: 1) optimisation of emergency response and post-accident recovery decisions; 2) development of Level 3 PSA guidance documents; and 3) enhancing the modelling capabilities of probabilistic consequence analysis codes (e.g. incorporating additional atmospheric transport and dispersion (ATD) and economic models, incorporating models to account for potential concurrent releases from multiple co-located radiological sources with unique radionuclide inventories and accident progression timelines).

3.C.2. Radionuclide release characterisation

3.C.2.1. Interface between Level 2 analysis and Level 3 analysis

Specific practices vary with respect to radionuclide release characterisation and treatment of the interface between the Level 2 analysis and the Level 3 analysis. Detailed descriptions of processes and criteria are provided in the individual survey responses in Appendix C.

Level 2 PSA analysts typically work closely with Level 3 PSA analysts in radionuclide release characterisation to: 1) define radiological release categories or source term groups; and 2) select a representative accident sequence for each radiological release category or source term group to estimate the source term characteristics that will serve as an input to the off-site radiological consequence analysis. This practice increases the efficiency of selecting representative accident sequences and reduces the computational time needed to generate time-dependent radiological release data using relevant severe accident analysis computer codes. For most countries that perform Level 3 PSA, on-site severe accident mitigation actions are considered in the Level 2 analysis performed to estimate characteristics of radiological releases (e.g. release fraction timing, release truncation time). In addition, most countries that perform Level 3 PSA described methods for performing sensitivity or uncertainty analyses on release categories or release fractions in their individual responses in Appendix C.

3.C.2.2. Set of radionuclides used to characterise off-site radiological consequences

The set of radionuclides used to characterise the off-site radiological consequences attributed to accidental releases of radiological materials varies across countries with the number ranging from 60 to 800. Based on analysis of survey responses, it appears that the computer codes used to perform off-site radiological consequence analysis is an important driver of the analyst's decision with respect to which radionuclides will be considered.

3.C.2.3. Treatment of concurrent releases from multiple co-located radiological sources

Practices also vary with respect to consideration of concurrent releases from multiple units or radiological sources co-located at the same site. Survey respondents from a few countries indicated they have performed research analyses of the 2011 accident at the Fukushima Daiichi Nuclear Power Stations that considered multi-unit or multi-source releases. Among countries that have considered multi-unit or multi-source releases beyond benchmarking the Fukushima Daiichi accident, modelling practices vary. For example, Korea has developed an approach in which the release fraction from a single unit is multiplied by the number of units on a site to develop the multi-unit source term input for the MACCS computer code. By contrast, Netherlands uses post processing (typically with spread sheet software) to sum the individual risk contributions attributable to accidental releases from multiple units. As part of an ongoing Level 3 PSA research project, the United States is exploring the use of a new multi-source modelling capability in MACCS to model releases from multiple co-located radiological sources that can have unique radionuclide inventories and potentially different accident progression timelines; the objective of this effort is to assess the contribution to total site risk from accidental releases involving different combinations of two or more major radiological sources at the modelled site.

3.C.3. Meteorological data

Among countries that perform Level 3 PSA, a typical practice with respect to the use of meteorological data is the use of site-specific data that includes hourly observations for an entire year

to perform off-site radiological consequence analyses. Using the probabilistic consequence analysis codes identified in Section 3.C.1., most countries implement code-specific weather binning and sampling strategies to account for the impact of temporal variability in weather conditions on off-site radiological consequences. A notable exception to this is the ARANO code used in Finland, which does not rely on sampling to perform probabilistic analyses. In ARANO, annual weather data from one meteorological mast is converted into a joint frequency matrix of annual weather statistics. Doses are calculated for all specified weather conditions and results are then weighted by the probability of each weather condition.

More specific details about the treatment of temporal and spatial variability in weather conditions, including methods for performing sensitivity or uncertainty analyses, are provided in the individual responses in Appendix C.

3.C.4. Environmental transport and dispersion

3.C.4.1. ATD Modelling

Most probabilistic consequence analysis codes identified in Section 3.C.1 are limited to Gaussian straight-line trajectory plume segment or puff ATD models. This is a significant driver typical practices with respect to ATD modelling, including the use of one-hour time steps for updating calculations to be consistent with available site-specific meteorological data. However, some codes used to perform Level 3 PSA are being enhanced to include a Lagrangian particle tracking ATD model, which – along with the Eulerian ATD model – has already been used within primarily deterministic consequence analysis codes that can account for changes in off-site weather conditions (e.g. the Finnish VALMA and SILAM codes). For example, a new particle tracking ATD model based on the HYbrid Single Particle Lagrangian Integrated Trajectory (HYSPLIT) model is under development for addition to MACCS and will be released in a future version. This particle tracking ATD model is being developed to provide an alternative ATD model that addresses known limitations of the Gaussian straight-line trajectory plume segment model. As more ATD model options become available, a process and criteria will be needed to select which model(s) will be used for a particular study. Some factors to consider in making the selection may include: 1) study objectives; 2) computational efficiency; 3) site characteristics; and 4) data availability.

In addition, although it is possible (at least in principle) to account for spatial variability in the ATD response due to variation in the topography or terrain around the modelled site, this is beyond the current state of practice for Level 3 PSA. In particular, available probabilistic consequence analysis codes either do not support terrain modelling or rely on the use of simple terrain models.

3.C.4.2. Spatial modelling

A typical practice among countries that perform Level 3 PSA is to use site-specific information sources (sometimes supplemented with generic information sources) to model various site characteristics, including: 1) population characteristics; 2) agricultural characteristics; and 3) economic characteristics.

Another finding based on analysis of survey responses is that the size of the modelled region and spatial intervals around a site are typically application- and site-specific. In general, a larger number of spatial intervals with finer resolution grid elements are used for the region close to the site, with the number of spatial intervals and grid resolution decreasing as the distance from the site increases.

3.C.4.3. Radiological releases to aqueous pathways

Respondents generally agreed that radiological releases to aqueous pathways should be considered if they could result in significant doses to off-site populations. However, radiological releases to aqueous pathways are typically screened out and excluded from Level 3 PSAs for nuclear installations that have the potential for airborne radiological releases, since airborne releases are expected to dominate radiological health risks. In particular, the United States provided two reasons for this: 1) movement of radionuclides to the accessible environment through aquatic pathways is expected to be slow relative to atmospheric transport; and 2) releases to groundwater or surface water are considered easier to interdict [4]. In addition, the United States response in Appendix C includes information about a research project the USNRC initiated after the Fukushima Daiichi accident to assess potential off-site consequences of losing control of highly contaminated water in a severe accident in which the contaminated water flows to a body of water. [5]

3.C.5. Protective action (countermeasure) modelling

As previously described in Section 3.B.1, the two countries that require performance of Level 3 PSAs for comparison with risk acceptance criteria (Korea and Netherlands) conservatively do not model or credit emergency response protective actions in determining whether prescribed risk acceptance criteria are satisfied. However, the Korean regulatory authority NSSC recently required confirmation of the effectiveness of protective actions in the licensing process for Shin-Hanul Units 1 and 2 (APR1400 series).

Among the remaining countries that perform Level 3 PSA, the following protective actions are typically modelled for the early (emergency) phase of accident response: 1) evacuation; 2) sheltering; 3) dose-dependent relocation; and 4) use of potassium iodide (KI) as a supplementary protective action to reduce the radiological dose to the thyroid gland by blocking the uptake of radioiodine. Protective actions that are typically modelled for the intermediate phase include: 1) dose-dependent relocation; and 2) dose-dependent interdiction or bans on agricultural products (e.g. dairy products, meats) and drinking water. Finally, protective actions that are typically modelled for the late (recovery) phase include: 1) dose-dependent relocation; 2) decontamination of contaminated land areas; 3) temporary interdiction or permanent interdiction (condemnation) of contaminated land areas; and 4) dose-dependent interdiction or bans on agricultural products and drinking water.

For those countries that model protective actions, most typically do not model population groups with different protective action behaviours. A notable exception to this is the United States, which uses site-specific information to select and define multiple evacuation cohorts based on several factors, including: 1) location or spatial interval around the nuclear installation; 2) delay times for implementing protective actions; 3) evacuation speeds; and 4) exposure factors and shielding parameters. Moreover, although Finland indicated it had used a simple probabilistic model to account for the probabilities of success or failure for protective actions in a research project, all countries that model protective actions indicated this is not typical practice when performing Level 3 PSAs for nuclear installations.

3.C.6. Radiological exposure and dose assessment

Among countries that perform Level 3 PSA, practices vary with respect to defining potentially exposed populations or cohorts, including consideration of potentially sensitive, vulnerable, or critical groups (e.g. infants, most exposed individual with otherwise normal characteristics). Some countries consider potentially sensitive, vulnerable, or critical groups, while others do not differentiate such groups from average members of the general public and use generic average values for exposure and dose conversion factors.

Since the objective of Level 3 PSA is to assess the risks to the off-site public that are attributable to a spectrum of possible accident scenarios involving a nuclear installation, countries that perform Level 3 PSA typically exclude the on-site worker population from the scope of the linked radiological consequence analysis.

In assessing radiological exposure and dose to the off-site public, countries that perform Level 3 PSA generally model all relevant exposure pathways that can result in a significant contribution to radiological dose. These exposure pathways typically include: 1) direct exposure to external radiation from the plume of released radiological materials (cloud shine); 2) inhalation of radioactivity in the plume; 3) contamination of skin and clothing; 4) direct exposure to external radiation from ground contamination (ground shine); 5) inhalation of resuspended radioactivity; and 6) ingestion of contaminated food and water.

3.C.7. Radiological health effects

3.C.7.1. Types of radiological health effects considered

All countries that perform Level 3 PSA model and estimate radiological health effects in linked off-site radiological consequence analyses. Individual responses provided in Appendix C include more specific information about specific health effects considered, dose-response models, and target organs used in these dose-response models. In general, a range of early radiological health effects are estimated using deterministic models, including: 1) early fatalities; 2) acute radiation syndrome (radiation sickness); and 3) other early injuries arising from acute doses to the red bone marrow, lungs, gastrointestinal tract, and skin. Fatal and non-fatal cancers involving multiple organs generally represent the latent radiological health effects that are estimated using stochastic models.

3.C.7.2. Dose-Response models used to estimate excess cancer risk

Survey respondents from all countries that perform Level 3 PSA indicated they use a dose-response model based on the linear no-threshold (LNT) hypothesis to estimate the excess cancer risk attributable to radiological doses caused by accidental releases from nuclear installations. Although prevailing knowledge of biological mechanisms and available epidemiologic evidence continue to support the LNT hypothesis, emerging evidence from in vitro radiation biology studies has spurred heated debate within the scientific community about the validity of LNT-based dose-response models. However, only the United States indicated that it is evaluating the impact of plausible alternative dose-response models using sensitivity analyses to account for uncertainty about the true dose-response relationship for exposures to low levels of ionizing radiation. In recent U.S. studies [6], the default LNT-based model was used in the base case analysis, and two models involving dose truncation levels were used as sensitivity cases. In the first model, the dose truncation was set at 6.2 mSv/year - a level that corresponds to the background radiation exposure an average individual in the US receives per year, including exposures attributed to medical and other man-made radiological sources. In the second model, the dose truncation was set at 50 mSv/year with a 100 mSv lifetime limit – a level below which the Health Physics Society recommends against quantitative estimation of health risks [7] (though the USNRC has not endorsed this position).

3.C.8. Economic consequences

Among countries that perform Level 3 PSA, respondents from two countries (Korea and Netherlands) indicated that economic consequences are not considered in linked off-site radiological consequence analyses. For the remaining countries, economic models are generally used to estimate the costs attributed to implementation of modelled protective actions to reduce radiological dose to the off-site public from the accidental release of radiological materials. Examples of costs considered include:

1) daily costs of compensation for populations subject to evacuation or short-term relocation arising from food, housing, transportation, lost income, or replacement of lost personal property; 2) costs of long-term relocation of populations and businesses in interdicted land areas; 3) depreciation costs that account for loss of value of interdicted property; 4) decontamination costs; and 5) costs arising from implementation of agricultural countermeasures.

In addition to these economic costs attributed to protective actions that are implemented to reduce radiological dose, Japan indicated that it also translates early and latent radiological health effects into economic consequences using input parameters that estimate the associated lost income. Although not performed as part of a Level 3 PSA, the United States may monetise the benefits of averted radiological health effects in cost-benefit analyses using a conversion factor applied to the averted population dose within a prescribed distance (typically 50 miles) from the nuclear installation. The USNRC is currently revising the value of this dollar per person-rem conversion factor and is proposing a process for routine updates to ensure its value is maintained current. Finally, Korea (KAERI) is currently performing research to evaluate the effects of various factors considered in the economic consequence model.

3.C.9. Consequence quantification and reporting

For the two countries that require performance of Level 3 PSAs for comparison with risk acceptance criteria (Korea and Netherlands), the consequence metrics selected for quantification and reporting are driven by the legal requirements associated with these risk acceptance criteria related to radiological health effects.

For the remaining countries, respondents generally indicated there is no explicit criteria for selecting consequence metrics for quantification and reporting, except that the objectives of the analysis should drive the selection of relevant metrics. In general, this means the consequence metrics will be selected on a case-by-case basis. The United States identified several factors that are typically considered in selecting consequence metrics and the spatial intervals over which they will be quantified. These factors include: 1) analysis objectives – which are typically linked to application-specific requirements or questions to be addressed; 2) stakeholder interests; 3) standards requirements or state-of-practice; 4) potential future uses of results and insights; 5) schedule and resource constraints; and 6) capabilities and limitations of models and analytical tools.

3.C.10. Risk characterisation

3.C.10.1. Selection of risk metrics

In response to a question about the process or criteria used to select risk metrics for effectively communicating Level 3 PSA results, survey respondents generally used the same response provided for the question about selection of consequence metrics described in Section 3.C.9.

Only the United States addressed the question of metrics designed to measure the effects of concurrent radiological releases involving multiple units or radiological sources co-located at the same site. The USNRC indicated that, in principle, risk metrics used to estimate the frequencies of off-site public health, economic, and environmental consequences for accidental releases from single radiological sources can be adapted by adjusting the frequency basis to estimate the same quantities for accidental releases from multiple radiological sources. For example, instead of quantifying risk metrics on a per-reactor-year basis, they can be quantified on a per-site-year or per-calendar-year basis. The USNRC is exploring options in this area as part of an ongoing Level 3 PSA research project.

3.C.10.2. Presentation and communication of risk results

Although practices vary among countries that perform PSA with respect to the presentation and communication of risk results, some typical practices were identified. In general, risk results that characterise variability or aleatory uncertainty arising from inherent randomness or stochastic processes are presented using a variety of formats and graphical displays. Commonly used examples include: 1) point estimates – especially expected (mean) risk of selected consequence metrics over all-weather trials; 2) FN curves that illustrate the frequency of N or more consequences (F) as a function of total number of consequences (N) to express societal risk; and 3) complementary cumulative distribution function (CCDF) curves (also termed exceedance frequency curves or risk curves) – which represent the frequencies of exceeding different consequence levels, and are typically used in a Level 3 PSA to illustrate the effect of variability in off-site weather conditions on off-site radiological consequences.

Commonly used examples of graphical displays used to illustrate uncertainties in risk results include : 1) empirical probability distributions for selected risk metrics – typically in the form of probability density function (PDF) curves and/or cumulative distribution function (CDF) curves; 2) box plots that illustrate the locations of key summary statistics (e.g. mean value, 50^{th} percentile (median value), 95^{th} percentile, and 5^{th} percentile) for selected risk metrics; and 3) sets of CCDF curves – with the different curves representing different probabilities of frequencies of exceeding different consequence levels.

Respondents did not differentiate between decision makers and the general public with respect to methods for presenting and communicating risk results and the uncertainty in risk results. In addition, most respondents indicated they do not have a communication plan or guidelines for risk communication. One exception was the United States; the USNRC has developed guidance documents that aim to enhance risk communication with both internal stakeholders (including decision makers) [8] and external stakeholders (including the general public) [9].

More specific details about the presentation and communication of risk results – including some example displays and citations for representative reports – are provided in the individual responses in Appendix C.

3.C.10.3. Treatment of uncertainties

Among countries that perform Level 3 PSA, practices vary with respect to the use of sensitivity or uncertainty analyses to characterise the uncertainty in risk results. More specific details about the treatment of uncertainties are provided in the individual responses in Appendix C. The range of practices identified in survey responses included use of: 1) information in previous studies to conclude that risk results are accurate to within one order of magnitude; 2) limited scope sensitivity analyses focused on evaluating the impact on risk results of using alternative parameter values or input data sets for a selected set of parameters or input data; and 3) Monte Carlo based sampling methods to propagate parameter uncertainties.

Most respondents who indicated they perform sensitivity analyses by varying parameter values or uncertainty analyses by specifying uncertainty distributions for propagating parameter uncertainty stated that they rely on expert judgement to determine which parameters will be varied or will have uncertainty distributions specified for propagating parameter uncertainty. In addition, respondents indicated that, while state of knowledge correlation between parameters can be treated in principle: 1) this is a difficult task due to insufficient information; and 2) the extent to which it is treated will be driven by the objectives of the analysis.

3.C.11. Additional topics

As shown in the final survey instrument provided in Appendix B, survey respondents were asked to identify: 1) any other questions they believed should have been asked in the survey; and 2) any other information, technical challenges, or notable practices they would like to share with the international community. Although most respondents did not identify any other questions or information to share, the following additional topics were identified by some respondents:

1. Consideration of environmental and non-radiological (e.g. psychosocial) impacts.
2. Development of novel risk metrics and methods for risk communication.
3. Accuracy of Level 3 PSA results.

If provided, more detailed information about each of these additional topics can be found in the individual survey responses in Appendix C.

4. CONCLUSIONS AND RECOMMENDATIONS

This chapter identifies and summarises key conclusions from this activity, including typical practices and common technical challenges or limitations that were identified through analysis of submitted survey responses. The chapter also provides recommendations for future activities that follow from these conclusions.

4.A. Typical practices

Analysis of submitted survey responses resulted in the identification of typical practices across several aspects of Level 3 PSA. These typical practices are summarised in this section and organised by Level 3 PSA technical element or aspect.

4.A.1. Application of Level 3 PSA

Typical practices around Level 3 PSA applications were identified with respect to three aspects: 1) uses of Level 3 PSA; 2) siting of nuclear installations; and 3) establishment of emergency planning or protective action zones. These typical practices are summarised below.

4.A.1.1. Uses of Level 3 PSA

Among countries that are currently performing, planning to perform, or considering performing Level 3 PSAs for nuclear installations, five broad types of Level 3 PSA applications were identified: 1) applied research; 2) comparison with risk acceptance criteria, safety goals, or other quantitative objectives; 3) evaluation of protective action effectiveness to inform emergency preparedness and response guidelines; 4) inform development of severe accident management plans; and 5) environmental assessments. In general, risk metrics used for each type of application and any corresponding criteria used to evaluate results are driven by the objectives of the Level 3 PSA. Commonly used risk metrics include: 1) population dose risk; 2) individual risk of radiological health effects; and 3) group (societal) risk of radiological health effects.

4.A.1.2. Siting of nuclear installations

Although survey respondents generally acknowledged the potential benefit of using results from Level 3 PSAs to inform nuclear installation siting decisions, most indicated that Level 3 PSAs are not being used to support such applications. Results from deterministic DBA dose consequence assessments are typically used to demonstrate that doses to individuals located at any point along the boundaries of defined areas around a nuclear installation for specified periods of time following postulated fission product releases are within prescribed limits.

4.A.1.3. Establishment of emergency planning or protective action zones

Results from deterministic analyses provide the primary analytical basis for establishing emergency planning or protective action zones. If Level 3 PSA insights are used in developing requirements for the sizes and boundaries of emergency planning or protective action zones, such risk insights are

typically considered as a complement to results from deterministic analyses as part of a risk-informed approach.

4.A.2. Level 3 PSA modelling and risk characterisation

This section summarises typical practices that were identified with respect to: 1) modelling choices an analyst faces in addressing certain Level 3 PSA technical elements; and (2) presentation of Level 3 PSA results as part of risk characterisation. Typical practices related to modelling choices are organised by Level 3 PSA technical element.

4.A.2.1. Radionuclide release characterisation

Among countries that perform Level 3 PSA, Level 2 PSA analysts typically work closely with Level 3 PSA analysts in radionuclide release characterisation to: 1) define radiological release categories or source term groups; and 2) select a representative accident sequence for each radiological release category or source term group to estimate the source term characteristics that will serve as an input to the off-site radiological consequence analysis. This practice increases the efficiency of selecting representative accident sequences and reduces the computational time needed to generate time-dependent radiological release data using relevant severe accident analysis computer codes. Most of these countries indicated that they perform sensitivity or uncertainty analyses on release categories or release fractions to evaluate the impact on risk results. Another typical practice in this area is that most countries that perform Level 3 PSA consider on-site severe accident mitigation actions in the Level 2 analysis performed to estimate source term characteristics.

4.A.2.2. Meteorological data and other information sources

Most countries that perform Level 3 PSA: 1) use site-specific data that includes hourly observations for an entire year to perform off-site radiological consequence analyses; and 2) implement code-specific weather binning and sampling strategies to account for the impact of temporal variability in weather conditions on off-site radiological consequences. In addition, most countries use site-specific information sources (sometimes supplemented with generic information sources) to model various other site characteristics, including: 1) population characteristics; 2) agricultural characteristics; and 3) economic characteristics.

4.A.2.3. Spatial modelling

Survey respondents indicated that the size of the modelled region and spatial intervals around a site are typically application - and site-specific. In general, a larger number of spatial intervals with finer resolution grid elements are used for the region close to the site, with the number of spatial intervals and grid resolution decreasing as the distance from the site increases.

4.A.2.4. Radiological releases to aqueous pathways

Respondents generally agreed that radiological releases to aqueous pathways should be considered if they could result in significant doses to off-site populations. However, radiological releases to aqueous pathways are typically screened out and excluded from Level 3 PSAs for nuclear installations that have the potential for airborne radiological releases, since airborne releases are expected to dominate radiological health risks. Two justifications for this practice were identified: 1) movement of radionuclides to the accessible environment through aquatic pathways is expected to be slow relative to atmospheric transport; and 2) releases to groundwater or surface water are considered easier to forbid [4].

4.A.2.5. Protective action (countermeasure) modelling

Among most countries that perform Level 3 PSA, a range of protective actions are typically modelled for the early (emergency), intermediate, and late (recovery) phases of accident response. Protective actions that are typically modelled in each phase include:

- **Early (Emergency) Phase:** 1) evacuation; 2) sheltering; 3) dose-dependent relocation; and 4) use of KI as a supplementary protective action to reduce the radiological dose to the thyroid gland by blocking the uptake of radioiodine.
- **Intermediate Phase:** 1) dose-dependent relocation; and 2) dose-dependent interdiction or bans on agricultural products and drinking water.
- **Late (Recovery) Phase:** 1) dose-dependent relocation; 2) decontamination of contaminated land areas; 3) temporary interdiction or permanent interdiction (condemnation) of contaminated land areas; and 4) dose-dependent interdiction or bans on agricultural products and drinking water.

In addition, most countries that model protective actions typically do not: 1) model population groups with different protective action behaviours; or 2) use probabilistic models to account for the probabilities of success or failure for protective actions.

4.A.2.6. Radiological exposure and dose assessment

Since the objective of Level 3 PSA is to assess the risks to the off-site public that are attributable to a spectrum of possible accident scenarios involving a nuclear installation, countries that perform Level 3 PSA typically exclude the on-site worker population from the scope of the linked radiological consequence analysis.

4.A.2.7. Radiological health effects

All countries that perform Level 3 PSA model and estimate radiological health effects in linked off-site radiological consequence analyses. In general, a range of early radiological health effects are estimated using deterministic models, including: 1) early fatalities; 2) acute radiation syndrome (radiation sickness); and 3) other early injuries arising from acute doses to various tissues or organs. Fatal and non-fatal cancers involving multiple organs generally represent the latent radiological health effects that are estimated using stochastic models.

Respondents from all countries that perform Level 3 PSA indicated they use a default dose-response model based on the LNT hypothesis to estimate the excess cancer risk attributable to radiological doses caused by accidental releases from nuclear installations. However, most indicated they do not perform sensitivity or uncertainty analyses to evaluate the impact of dose-response model uncertainty or use of plausible alternative dose-response models on estimates of excess cancer risk.

4.A.2.8. Economic consequences

Although there are exceptions, most countries that perform Level 3 PSA consider economic consequences in linked off-site radiological consequence analyses. These countries use economic models to estimate the costs attributed to implementation of modelled protective actions to reduce radiological dose to the off-site public from the accidental release of radiological materials. Examples of costs considered include: 1) daily costs of compensation for populations subject to evacuation or short-term relocation arising from food, housing, transportation, lost income, or replacement of lost personal property; 2) costs of long-term relocation of populations and businesses in interdicted land

areas; 3) depreciation costs that account for loss of value of interdicted property; 4) decontamination costs; and (5) costs arising from implementation of agricultural countermeasures.

4.A.2.9. Risk characterisation

In general, risk results that characterise variability or aleatory uncertainty arising from inherent randomness or stochastic processes are presented using a variety of formats and graphical displays. Commonly used examples include: 1) point estimates – especially expected (mean) risk of selected consequence metrics over all-weather trials; 2) FN curves that illustrate the frequency of N or more consequences (F) as a function of total number of consequences (N) to express societal risk; and 3) CCDF curves. Commonly used examples of graphical displays used to illustrate uncertainties in risk results include: 1) empirical probability distributions for selected risk metrics – typically in the form of PDF curves and/or CDF curves; 2) box plots that illustrate the locations of key summary statistics (e.g. mean value, 50^{th} percentile (median value), 95^{th} percentile, and 5^{th} percentile) for selected risk metrics; and 3) sets of CCDF curves.

4.B. Common technical challenges or limitations

In addition to the typical practices summarised in the previous section, common technical challenges or limitations across several aspects of Level 3 PSA were also identified through analysis of submitted survey responses. These common technical challenges or limitations are summarised in this section and organised by Level 3 PSA technical element or aspect.

4.B.1. Application of Level 3 PSA

Among countries that are not currently performing Level 3 PSAs for nuclear installations, four potential barriers to performing Level 3 PSA were identified: 1) absence of a technical or legal framework to perform Level 3 PSA; 2) large uncertainties in Level 3 PSA results, especially when combined with uncertainties propagated from Level 1 and Level 2 analyses; 3) limited expected benefit in terms of potential back fits or safety improvements; and 4) additional resources required to perform detailed off-site radiological consequence analyses.

4.B.2. Level 3 PSA Technical Elements

This section summarises common technical challenges and limitations that were identified with respect to various Level 3 PSA technical elements.

4.B.2.1. ATD Modelling

Most computer codes for performing probabilistic analyses of off-site radiological consequences are limited to straight-line Gaussian plume segment or puff trajectory ATD models. However, some codes are being enhanced for Level 3 PSA applications to include a Lagrangian particle tracking ATD model (e.g. VALMA and MACCS). Use of weather data and sampling strategies that account for weather variability over defined time intervals is also driven primarily by code limitations and capabilities. As previously stated in Section 4.A.2.2., most respondents indicated that they use code-specific weather binning and sampling strategies to account for the impact of temporal variability in weather conditions on off-site radiological consequences.

In addition, although it is possible (at least in principle) to account for spatial variability in the ATD response due to variation in the topography or terrain around the modelled site, countries that perform Level 3 PSA do not account for this effect. This is partly because available probabilistic consequence analysis codes rely on the use of simple terrain models.

4.B.2.2. Risk characterisation

None of the available probabilistic consequence analysis computer codes can directly combine conditional consequence results with frequency results from the Level 2 analysis to estimate and characterise off-site public risks. A common solution to this limitation is to post-process the results by mathematically combining the radiological release category frequencies with the corresponding conditional consequences using an application external to the probabilistic consequence analysis code.

In addition, while some commonly used examples of risk metrics were identified in Section 4.A.1.1., most respondents indicated they do not have a communication plan or guidelines for risk communication. A notable exception is the USNRC that has developed guidance documents aiming to enhance risk communication with both internal stakeholders (including decision makers) [8] and external stakeholders (including the general public) [9].

4.C. Recommendation for future activities

A key finding from analysis of submitted survey responses is that many typical practices or common technical challenges identified as part this WGRISK activity are driven by the limitations or capabilities of available probabilistic consequence analysis codes. Aspects of Level 3 PSA that appear to be impacted by limitations or capabilities of the probabilistic consequence analysis code used for a particular analysis include:

- The set of radionuclides used to characterise the off-site radiological consequences attributed to accidental releases of radiological materials.
- Treatment of concurrent radiological releases from multiple radiological sources co-located at the same site.
- Selection of the ATD model and options for accounting for the impact of temporal and spatial variability in ATD response.

This finding that the choice of probabilistic consequence analysis code can influence typical practices and technical challenges or limitations underscores the potential need for future research and leads to the following recommendation stemming from this WGRISK activity.

Recommendation: Consider performing a study to benchmark available probabilistic consequence analysis computer codes used for Level 3 PSA applications.

The NEA previously performed similar studies to benchmark probabilistic consequence analysis codes [10,11]. Yet the most recent of these studies was completed more than 20 years ago. Since probabilistic consequence analysis codes have evolved considerably during the past two decades, it seems reasonable to perform a follow-on benchmarking study to compare more recent versions of the computer codes. However, many of these probabilistic consequence analysis codes are currently being enhanced to address various modelling issues and technical challenges, some of which were brought into focus by the 2011 accident at the Fukushima Daiichi nuclear power plant in Japan. Examples of these code enhancements include:

- Addition of the capability to model concurrent releases from multiple co-located radiological sources that can have unique radionuclide inventories and potentially different accident progression timelines.
- Addition of Lagrangian particle tracking ATD models.
- Addition of improved economic models.

Based on these ongoing code enhancements, a logical path forward may to be to delay any future benchmarking study (if one is to be undertaken) until after these enhancements have been implemented, with verification and validation testing completed, as appropriate.

5. REFERENCES

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APPENDIX A: CSNI ACTIVITY PROPOSAL SHEET (CAPS) WGRISK(2015)3[1]

[1] The CAPS provided in Appendix A represents an exact duplicate of the document that CSNI approved in 2015. When the CAPS was developed and approved, IAEA was performing a project to develop guidelines for performing Level 3 PSA, which would be documented in an IAEA TECDOC report. This project has since been cancelled and will no longer result in publication of a TECDOC.

CSNI Activity Proposal Sheet (CAPS) for a Proposed New Activity
WGRISK(2015)3

Project/Activity Title	Status of Practice for Level 3 Probabilistic Safety Assessment
Objective	.The mission of the Working Group on Risk Assessment (WGRISK) is to advance the understanding and utilization of Probabilistic Safety Assessment (PSA) in ensuring continued safety of nuclear installations in Member countries. In pursuing this goal, the WGRISK routinely shares information regarding PSA methodologies used to identify nuclear power plant risk contributors and assess their importance. Based on feedback from recent WGRISK meetings and information in report NEA/CSNI/R(2012)11, "Use and Development of Probabilistic Safety Assessment An Overview of the situation at the end of 2010," published in December 2012, an increasing number of countries are pursuing development of PSAs that explicitly calculate public health, safety, and economic impacts (Level 3 PSA). The objective of this task is to: (1) survey member and observer countries to determine current methodological practices in Level 3 PSA, (2) identify common challenges and notable practices, and (3) summarize the results of this activity in a final task report.
Scope	The purpose of this task is to identify current practices for the conduct of Level 3 PSA (consequence analysis)[1] for nuclear power plants. This activity will be accomplished through the use of a focused survey questionnaire to gather information on the current state of practice in member and observer countries and to solicit views on several challenging technical areas. The questionnaire will serve as an important input to an expanded task group[2] meeting/workshop that will provide a forum for more detailed discussions and information exchange on key topics for Level 3 PSA. The results of the survey and meeting/workshop will be summarized in a final task report. This activity will complement ongoing work being done by IAEA on Level 3 PSA. The current IAEA work is focused on developing a TECDOC which will outline of state-of-the- art guidance on Level-3 PSA for NPPs[3].

[1] In this CAPS, "Level 3 PSA" refers to only the portion of a PSA from the release categories and source terms to public health and economic consequences (i.e. the scope does not include initiating events to plant damage states (Level 1 PSA) or plant damage states to release categories/source terms (Level 2 PSA).

[2] The "expanded task group" includes task core group members and other WGRISK participants familiar with both the topic and the development of the questionnaire. The expanded task group workshop will include an expanded agenda that will include sufficient time for participants to conduct a more detailed analysis and discussion of the insights obtained from the survey questionnaire.

[3] The IAEA TECDOC on Level 3 PSA is expected to be published in 2016. Following issuance of the TECDOC, IAEA anticipates the publication a safety guide to provide state-of-the-art guidance for Level 3 PSA to replace the outdated guidance contained in IAEA Safety Series 50-P-12 (1996), "Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 3) Off-Site Consequences and Estimation of Risks to the Public".

Scope (continued)	This WGRISK activity will complement the ongoing IAEA work by providing more detailed current practices information for specific technical areas (e.g., those noted in the "Justification" section). Although this task is expected to identify areas requiring future research and development, the resolution of these issues is beyond the scope of this task (though these issues will be identified and discussed in the task report). Approaches and methods that are considered notable practices by the task group will be highlighted.
Justification	The use of Level 3 PSA (consequence analysis) appears to be increasing in WGRISK member and observer countries. A Level 3 PSA is used to calculate public health and safety impacts (e.g., early fatality and latent cancer risk) as well as economic impacts. This information supports a more complete characterization of nuclear power plant risk and provides an important input into cost-benefit analysis of proposed regulatory changes. Based on the experience from several WGRISK members, there are a numbering of challenging issues that would benefit from increased information exchange and sharing. This information would help harmonize approaches currently being used for Level 3 PSA and assist countries in developing new Level 3 analyses. If approved, this task will examine a number of challenging issues associated with Level PSA consequence analysis. Examples of issues that may be examined include the following: · Risk metrics for effectively communicating Level 3 PSA results, including metrics appropriate for capturing multiunit issues. · Modelling practices, including distance truncation, dose response (linear no-threshold, low dose truncation, etc.), exposure pathways (aquatic, airborne), scenario binning (identifying appropriate plant damage states and release categories), and radiological transport. · Release truncation time, including consideration of severe accident mitigation and offsite response capability. · Determination of economic impacts due to property and health impacts, including valuation of life and property impacts, costs associated with emergency preparedness and radiological countermeasures (with emphasis on economic impacts that may influence public safety). · Consideration of site radiological sources including multiple units, spent fuel, and other sources. · Communication practices for the public and decision makers, including use of expected values; frequency/consequence curves; uncertainty quantification; and placing low likelihood events into an understandable context.

Users	Expected users include industry, regulators, and technical support organizations interested in current approaches for PSA methods, tools, and applications. This task would also be of interest to other NEA working groups and organizations interested determination of the health, safety, and economic impacts from nuclear power plant severe accidents.
Relation to other projects	It is expected that other CSNI and CNRA working groups with an interest in severe accident modelling (e.g., WGAMA) and PSA applications related to public health will have an interest in activity. Additionally, it is expected that this task will serve to better inform ongoing activities to improve the estimation of severe accident health impacts and costs (e.g., severe accident cost study currently being conducted by the NEA Nuclear Development Division). This project will coordinated with the CNRA Working Group on Public Communication (WGPC), the Committee on Radiation Protection and Public Health (CRPPH), and the NEA Nuclear Development Division Expert Group on Costs of Nuclear Accidents, Liability Issues and their Impact on Electricity Costs (EG-COSTNA). This task will also be coordinated with the IAEA representative to WGRISK, to ensure proper coordination with the ongoing IAEA efforts in the Level 3 PSA area.
Safety significance/ priority (see priority criteria in Section IV)	Regarding the priority criteria set in Section IV of the CSNI Operating Plan: · **Criterion 1: Relevance to CSNI challenges and technical goals** – As noted in the following section on "Technical Goal(s) Covered," this task relates to several important CSNI challenges and technical goals. · **Criterion 2: Better accomplished by international group** – The purpose of this task is to collect and summarize current developments in Level 3 (consequence analysis) PSA methodologies and applications from member and observer countries. This task is best accomplished by an international group such as WGRISK. · **Criterion 3: Likely to bring conclusive results in reasonable time frame** – The objective of this task is to identify the current state of practice for Level 3 PSA in member and observer countries and identify, where possible, notable practices. Although it is expected that the activity will identify a number of difficult challenges for consequence analysis, it is not within the scope of this effort to resolve these complex issues. Therefore, it is likely that the objectives of this task can be accomplished in a reasonable time frame.

Safety significance/ priority (see priority criteria in Section IV) (continued)	· **Criterion 4: Maintain and preserve strategic safety competence** – A key objective of this task is to share the state of current practice for Level 3 PSA methods and applications among member and observer countries. This information sharing will facilitate knowledge management and promulgation of current practices throughout the CSNI membership.
Technical Goal(s) covered	The following CSNI Challenges and Technical Goals are addressed by this task: · **Effectiveness and efficiency of activities related to safety**: (2c) To build and harmonize international approaches to safety issues through the use of multi-national collaborative efforts CSNI, and cost and resource sharing · **Safe operation of current nuclear installations**: (3g) To further review and assess the development of PSA methods; to promote further PSA applications in the operation of nuclear installations and review risk-informed approaches. · **Safety in New Nuclear Installations**: (4c)To review current analytical tools as well as risk assessment approaches regarding their applicability to safety assessments of new designs and further develop and validate them where needed. · **Safety in Advanced Nuclear Designs**: (5c) To review the adequacy of analytical tools and risk assessment approaches for application to advanced designs and evaluate and validate new analytical approaches when called for by specific features of new designs.
Knowledge Management and Transfer covered	A main objective of this task is to document current methods and applications of Level 3 PSA (consequently analysis) among member and observer countries. As such, the final task report serves as a valuable information sharing and knowledge management PSA document.
Milestones (deliverables vs. time)	If approved, the following milestone schedule is proposed: · September 2015: Development of draft survey instrument to gather information regarding member and observer country experience with Level 3 PSA and views on a range of technical topics.

Milestones (deliverables vs. time) (continued)	· December 2015: Finalization of survey questionnaire and review by WGRISK membership and representatives from WGAMA, the Working Group on Public Communication (CNRA/WGPC), IAEA, CRPPH and EG-COSTNA. Distribution of survey to WGRISK members and observers. See note below. · April 2016: Survey responses requested from member and observer countries. · October 2016: Develop summary of survey responses and hold expanded task group meeting/workshop to discuss preliminary results. · February 2017: Development of draft task report. · June 2017: CSNI approval of task report. Note: CSNI has requested a briefing from WGRISK to describe the coordination and interaction of this activity with other CSNI groups, the CNRA WGPC, IAEA, and CRPPH.
Lead organization(s) and coordination	The lead organization for this task is U.S. NRC. A core task group has been identified to coordinate this activity and develop the final task report. The core task group includes representatives from the NRC (US), NRG (Netherlands), CNSC (Canada), KAERI (Korea), JAEA (Japan), and NUBIKI (Hungary).
Participants (individuals and organizations)	All WGRISK members and observers will be asked to participate.
Resources	Leader (US NRC): 2-3 months Each member of the core task group would require 1 man-month in order to participate in task group meetings and summarize member and observer inputs and develop the final task report. Each participating organization would require approximately 0.5 man-months to respond to the survey questionnaire.
CSNI Decision	Endorsed subject to the following being recognised
CSNI Observation	It was considered to be a complex CAPs which was overambitious in its outcomes. The topic crossed a number of working groups as well as all three main safety standing committees CNRA, CSNI and CRPPH. It was considered important when developing the detailed plans for delivering this CAPS there was clarity regarding interaction with CSNI groups, the WG on Public Communications, IAEA and CRPPH. In addition clarity regarding the differences in the work of the IAEA and the NEA. The title should be modified to "Status of Practice for Level 3 PSA" as the methodologies were not seen to be mature enough to define best practice. When available the clarity requested above should be presented to CSNI by WGRISK.

APPENDIX B:

FINAL SURVEY INSTRUMENT

PURPOSE

The purpose of this WGRISK survey is to elicit and characterize the international state-of-practice with respect to modelling issues and other technical challenges that practitioners encounter in performing offsite radiological consequence analyses in support of Level 3 probabilistic safety assessment (PSA)[1] applications for nuclear installations. Your response to this survey will contribute to an enhanced understanding of how practitioners across the international PSA community are addressing specific modelling issues and other technical challenges in performing Level 3 PSA offsite radiological consequence analyses in support of nuclear installation applications.

Core task group members[2] for this activity will collect and analyse survey responses to generate results and insights for use in a subsequent expanded task group[3] workshop that will provide a forum for more detailed discussions and information exchange on key Level 3 PSA topics.

BACKGROUND AND TASK OBJECTIVES

The mission of WGRISK is to advance the understanding and utilization of PSA in ensuring the continued safety of nuclear installations in member countries. In support of this mission, WGRISK routinely collects and shares information regarding PSA methods and practices used to estimate nuclear accident risks and to identify significant risk contributors in support of nuclear installation applications.

An increasing number of countries are pursuing the development and application of Level 3 PSAs that directly estimate the frequencies of offsite public health, environmental, and economic consequences attributable to accidental releases of radiological materials from nuclear installations.[4,5] The experience of several WGRISK members indicates there are a number of challenging issues that would benefit from increased information exchange and sharing of methods and practices in this area.

CSNI therefore approved a new WGRISK Level 3 PSA activity in June 2015. The objectives of this activity are to: (1) survey member and observer countries to determine current methodological

[1] A Level 3 PSA is an assessment of the offsite public risks attributable to a spectrum of possible accident scenarios involving a nuclear installation. In the traditional PSA framework for commercial nuclear power plants, a Level 3 PSA includes three progressive levels of analysis: (1) core damage accident or plant damage state frequency analysis; (2) accident progression, containment performance, and radiological release analysis; and (3) offsite radiological consequence analysis. Although a Level 3 PSA includes all three levels of analysis, this WGRISK activity focuses only on the offsite radiological consequence analysis and its interface with other levels of analysis.

[2] The core task group includes representatives from the following WGRISK member countries: (1) Canada, (2) Hungary, (3) Japan, (4) Korea, (5) Netherlands, and (6) United States (lead organization).

[3] The expanded task group includes core task group members and other interested NEA participants. In addition, in order to ensure effective co-ordination of this activity, other NEA groups will be kept informed of the progress of this task, including the (1) Working Group on Analysis and Management of Accidents (WGAMA), (2) Committee on Nuclear Regulatory Activities (CNRA) - Working Group on Public Communication (WGPC), (3) Committee on Radiation Protection and Public Health (CRPPH), (4) NEA Nuclear Development Division - Expert Group on Costs of Nuclear Accidents, Liability Issues and their Impact on Electricity Costs (EG-COSTNA), and (5) International Atomic Energy Agency (IAEA).

[4] Working Group on Risk Assessment, Committee on the Safety of Nuclear Installations. *Use and Development of Probabilistic Safety Assessment: An Overview of the Situation at the end of 2010. NEA/CSNI/R(2012)11*. Paris, France: Nuclear Energy Agency; 2012. Available at: https://www.oecd-nea.org/nsd/docs/2012/csni-r2012-11.pdf.

[5] International Atomic Energy Agency. *Output of the IAEA Technical Meeting on Level 3 Probabilistic Safety Assessment July 2-6 2012*. Vienna, Austria: International Atomic Energy Agency; 2012.

practices in Level 3 PSA; (2) identify common challenges and notable practices; and (3) summarize the results of this activity in a final task report.[1]

SURVEY STRUCTURE

This survey is comprised of three parts. Part 1 is designed to obtain respondent information that will allow the group to determine whether responses differ across different countries or types of organizations.

Part 2 is designed to obtain information about whether and how respondents or organizations use or intend to use Level 3 PSA. Since the purpose or intended use of any PSA can influence multiple aspects of a PSA model (e.g., scope, level of detail, assumptions, or other modelling choices), responses to Part 2 will provide important contextual information for understanding and evaluating responses to Part 3.

Part 3 is designed to elicit information about whether and how respondents or organizations are addressing specific modelling issues and other technical challenges that practitioners encounter in performing offsite radiological consequence analyses in support of Level 3 PSA applications for nuclear installations. To facilitate identification and organization of this information, survey items in Part 3 are grouped by the associated major technical elements or tasks that typically comprise such offsite radiological consequence analyses.

PART 1: RESPONDENT INFORMATION[2]

Name:	
Country:	
Organization:	
Type of Organization:	☐ **Regulatory Authority** ☐ **Utility** ☐ **Vendor** ☐ **Academic/Research** ☐ **Other (please specify):** _____
Mailing Address:	
E-mail Address:	
Telephone Number:	

[1] Working Group on Risk Assessment. *CSNI Activity Proposal Sheet (CAPS) for a Proposed New Activity: Status of Practice for Level 3 Probabilistic Safety Assessment. WGRISK (2015) 3*. Paris, France: Committee on the Safety of Nuclear Installations; 2015.

[2] Many survey items use the term "you" in asking questions about if or how particular issues or challenges are being addressed. If you are responding to this survey on behalf of your organization, it is assumed that your response will be representative of practice within your organization. If your practice as an individual differs from that of your organization - and if you are providing information about your personal practice - please ensure this is clearly documented in your response.

PART 2: APPLICATION OF LEVEL 3 PSA

2.1 Does your country require the performance of Level 3 PSAs for nuclear installations? If yes, for what types of applications[1] are Level 3 PSAs required?

2.2 With regard to the use of Level 3 PSA:

2.2.1. Do you perform, or are you considering performing Level 3 PSAs for nuclear installations? If you do, for what types of applications do you perform them?

2.2.2. What calculated metrics or results from Level 3 PSAs are used in these applications and what (if any) requirements, goals, or criteria are used to evaluate these results? What is the basis for these requirements, goals, or decision criteria?

2.2.3. If you are not currently performing or considering performing Level 3 PSAs, have you previously considered performing them? If you have, what was the basis for your decision to not perform Level 3 PSA? Please list any barriers you perceive to performing and applying Level 3 PSA in your country.[2]

2.2.4. What (if any) alternative methods do you use to estimate offsite public risks attributable to accidental releases of radiological materials from nuclear installations, and for what types of applications are they used?

2.3 One application that could potentially benefit from Level 3 PSAs is the siting of nuclear installations, including establishing the size and boundary (shape) for each emergency planning or protective action zone. Are Level 3 PSAs used to support the siting of nuclear installations in your country? If they are, how are they used?

2.4 With regard to nuclear power plant emergency planning zones:

2.4.1. What process is used to establish sizes and boundaries for emergency planning zones around a nuclear installation in which arrangements shall be made at the preparedness stage for effectively taking protective and other response actions?

2.4.2. What stakeholder groups are involved in the process and what are their respective responsibilities?

2.4.3. Under what conditions (if any) can emergency planning zones be reduced in size? Can a Level 3 PSA be used to establish a probabilistic cut off for events that need to be considered for emergency planning purposes?

2.5 Another application that could potentially benefit from Level 3 PSAs is the development of safety goals or risk acceptance criteria.[3] Are Level 3 PSAs used to support this type of application in your country? If they are, how are they used?

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[1] In this survey, the term "application" is used to broadly represent the purpose or intended use of a Level 3 PSA. Example Level 3 PSA applications for nuclear installations can include, but are not limited to: (1) siting of nuclear installations; (2) establishing emergency planning or preparedness requirements; (3) developing or applying safety goals or risk acceptance criteria; (4) comparison with public risks attributed to alternative energy technologies; or (5) developing nuclear liability insurance requirements and compensation schemes for potential third-party damages.

[2] Examples of potential barriers include, but are not limited to: (1) cost or limited resource availability; (2) limited availability of expertise, information, or technology; or (3) legal requirements.

[3] Working Group on Risk Assessment, Committee on the Safety of Nuclear Installations. *Probabilistic Risk Criteria and Safety Goals. NEA/CSNI/R(2009)16.* Paris, France: Nuclear Energy Agency; 2009. Available at: https://www.oecd-nea.org/nsd/docs/2009/csni-r2009-16.pdf.

2.6 Offsite radiological consequence analyses may be performed to support applications that are not related to Level 3 PSA.[1] Please list any additional applications supported by offsite radiological consequence analyses that you support or perform.

[1] Examples of such applications include, but are not limited to: (1) analysis of design-basis accident (DBA) scenarios to support development of safety analysis reports; (2) analysis of a limited set of beyond-design-basis accident (BDBA) scenarios to support development of severe accident management strategies; or (3) determination of compensation awarded for third-party damages.

PART 3: LEVEL 3 PSA MODELING ISSUES AND TECHNICAL CHALLENGES[1]

General

3.1 What computer code(s) do you use, or are you developing, to perform the tasks listed below?

3.1.1. Offsite radiological consequence analyses:
3.1.2. Risk characterization[2]

3.2 With regard to the scope of Level 3 PSAs:

3.2.1. What internal and external accident sequence initiating events and hazards are considered in developing a Level 3 PSA?
3.2.2. Do you account for correlation between causes of accident sequence initiating events and offsite phenomenological and consequence modelling?[3] If you do, how is this correlation treated?

3.3 Are you participating in any ongoing research and development activities related to offsite radiological consequence analyses? If you are, please provide a brief description of these activities.

Radionuclide Release Characterization

3.4 With regard to the interface between Level 2 and Level 3 PSA:

3.4.1. How do you handle the transition from Level 2 PSA analyses to Level 3 PSA analyses? What (if any) difficulties have you encountered in this area?
3.4.2. If you use representative release categories or source term groups, what criteria do you use to assign radiological release sequences in the Level 2 PSA model to release categories or source term groups? How do you address potential inter-sequence variability within each representative release category or source term group?
3.4.3. What process do you use to define release fraction timing (time and duration of release) and release truncation time? Do you consider onsite severe accident mitigation actions or offsite emergency response actions in this process?[4] If you do, how are they treated?

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[1] Many survey items in Part 3 aim to elicit whether and how you address specific modelling issues that practitioners have found to be particularly difficult or challenging. In providing your response to these questions, please feel free to identify and describe any related issues that you have found to be challenging in developing and applying offsite radiological consequence analyses in support of Level 3 PSA applications for nuclear installations.

[2] Risk characterization is the process of developing and evaluating risk triplets comprised of accident scenarios, frequencies, and conditional consequences to produce: (1) qualitative insights about accident scenarios and significant risk contributors; and (2) quantitative estimates for risk metrics of interest. These quantitative estimates are typically calculated by combining estimates of radiological release category frequencies from Level 2 PSA analyses with conditional consequence metric estimates from Level 3 PSA analyses for each radiological release category.

[3] For example, if high winds cause an initiating event that is modelled in the Level 1 PSA, are high winds modelled in the offsite radiological consequence analysis for the radiological release associated with the accident scenario? Another example could be the correlations between time of year or season (e.g., summer vs. winter) and both initiating event frequencies and offsite population density.

[4] For example, in specifying the release duration, do you consider the time to implement mitigation measures to stabilize the reactor core and terminate releases?

3.5 What radionuclides do you use to characterize the offsite radiological consequences attributed to accidental releases of radiological materials? What is your basis for selecting these radionuclides?

3.6 Do you consider releases from multiple units or radiological sources co-located at the same site? If you do, how are these treated?

3.7 Do you perform sensitivity or uncertainty analyses on release categories or release fractions? If you do, what methods are used and what is the basis for using them?[1]

Meteorological Data

3.8 How do your account for temporal and spatial variability in meteorological conditions? If you use a subset (sampling) of available meteorological data to serve as a representative data set, what criteria do you use for selecting these representative data?

3.9 Do you perform sensitivity or uncertainty analyses on meteorological conditions? If you do, what methods are used and what is the basis for using them?

Environmental Transport and Dispersion

3.10 What atmospheric transport and dispersion (ATD) model(s) do you use? What process and criteria do you use to select: (1) an ATD model; and (2) a time scale for updating calculations to account for time dependence?

3.11 What process and criteria do you use to define the boundaries and intervals of the spatial grid or domain used for performing offsite consequence calculations?

3.12 With regard to the spatial modelling around a nuclear power plant for Level 3 PSA:

3.12.1. What information sources do you use to develop geographical or topographical parameters? Do you use generic or site-specific data?
3.12.2. Does the ATD response vary spatially with respect to varying topographic parameters? What is the spatial resolution of the ATD and do topographic parameters of the ATD vary within this resolution?

3.13 Do you consider radiological releases to water sources and aqueous transport and dispersion phenomena? If you do, how are these treated? If you have previously considered modelling radiological releases to water sources and aqueous transport and dispersion phenomena, but decided not to do so, what was the basis for your decision?

Protective Action (Countermeasure) Modelling

3.14 How do you define, and what protective actions (countermeasures) do you model for each of the nuclear or radiological incident or accident response phases listed below?

3.14.1. Early/Emergency Phase:
3.14.2. Intermediate Phase:
3.14.3. Late/Recovery Phase:

[1] For this and other survey items pertaining to sensitivity or uncertainty analyses, it is recognized that performing a complete uncertainty analysis can require extensive resources. One purpose for including these survey items is to identify and understand how practitioners are addressing uncertainty in light of practical resource limitations.

3.15 Do you model population groups with different protective action (countermeasure) behaviours? If you do, how do you define them? To what extent do you account for population density or age and gender distribution effects?[1]

3.16 Do you use probabilistic models of protective action (countermeasure) behaviours that model the probabilities of success or failure for protective actions (countermeasures)? If you do, please describe the models and their bases.

3.17 What information sources[2] do you use as a technical basis for protective action (countermeasure) modelling? Do you use generic or site-specific data?

Radiological Exposure and Dose Assessment

3.18 How do you define potentially exposed populations or cohorts? Do you consider onsite (e.g., worker) populations or potentially sensitive, vulnerable, or critical groups? If you do, what is your basis for including these groups, and how are they modelled?

3.19 What exposure pathways do you model?[3] What is your basis for selecting these pathways? What exposure duration is assumed in your models?

3.20 What information sources do you use to develop exposure and dose conversion factors? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

Radiological Health Effects

3.21 Do you model and estimate radiological health effects in your offsite radiological consequence analyses? If you do, please respond to questions 3.22 through 3.24. If you do not, please describe any other methods you use to address the offsite radiological health effects attributable to possible accident scenarios involving nuclear installations and proceed to question 3.25.

3.22 What types of early and latent health effects (including fatal and non-fatal effects) do you model? What target organs do you use for the associated health effects models?

3.23 What dose-response model(s) do you use to estimate the numbers of latent health effects attributable to radiological doses caused by accidental releases from nuclear installations, and what is the basis for their use? What (if any) methods do you use to account for uncertainty about the true dose-response relationship for exposures to low levels of ionizing radiation?

3.24 What information sources do you use to develop input parameters for the health effect models? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

[1] Population variability or heterogeneity with respect to radiological exposure and dose assessment is addressed in item 3.18.

[2] Examples of information sources for this and other questions include, but are not limited to: (1) national or international standards; (2) national or international guidelines; or (3) generic or site-specific studies.

[3] Examples of exposure pathways include, but are not limited to: (1) direct exposure to external radiation from the plume of released radiological materials (cloud shine); (2) inhalation of radioactivity in the plume; (3) contamination of skin and clothing; (4) direct exposure to external radiation from ground contamination (ground shine); (5) inhalation of resuspended radioactivity; and (6) ingestion of contaminated food and water.

Economic Consequences

3.25 Do you model and estimate economic consequences in your offsite radiological consequence analyses? If you do, please respond to questions 3.26 and 3.27. If you do not, please describe any other methods you use to address the offsite economic consequences attributable to possible accident scenarios involving nuclear installations and proceed to question 3.28.

3.26 With regard to potential economic consequences considered within a Level 3 PSA:

3.26.1. What types of economic consequences do you model? What process do you use to select cost categories and models for estimating economic consequences?
3.26.2. If you translate radiological health effects into economic consequences, please describe your process for doing so.
3.26.3. If you model land contamination, please describe your process for estimating the health, environmental, or economic consequences attributed to land contamination.

3.27 What information sources do you use to develop cost parameters for economic consequence models? Do you use generic, region-specific, or site-specific data?

Consequence Quantification and Reporting

3.28 What process and criteria[1] do you use for selecting consequence metrics for quantification and reporting, including spatial intervals or distances from release points?

Risk Characterization

3.29 What process and criteria[1] do you use for selecting risk metrics for effectively communicating Level 3 PSA results, including metrics designed to measure the effects of radiological releases involving multiple units or radiological sources co-located at the same site, if applicable?

3.30 For each of the groups listed below, what methods do you use for presenting and communicating risk results and the uncertainty in risk results?[2] How do you present the results from low-probability/high-consequence events in an understandable context for each group? Who is responsible for communicating the results to each group?

(1) Decision makers:
(2) General Public:

3.31 With regard to the treatment of uncertainties:

3.31.1. Do you perform sensitivity or uncertainty analyses as part of risk estimation? If you do, what process and methods do you use?

[1] Processes and criteria for selecting consequence or risk metrics can come from a variety of sources including, but not limited to: (1) legal or regulatory requirements; (2) standards or guidelines; or (3) application- or decision-specific characteristics (e.g., ability to compare and choose among alternatives).

[2] Example methods include, but are not limited to: (1) expected values (means) or other point estimates; (2) frequency-consequence curves; (3) complementary cumulative distribution function (CCDF) curves; or (4) a combination of these or other methods.

3.31.2. How do you determine which parameters will be varied or will have uncertainty distributions specified for propagating uncertainty? What parameters do you evaluate?

3.31.3. To what extent do you consider the effect of correlation on parameters?

Other

3.32 Are there any other questions that you believe should have been asked in this survey? Is there any other information or are there any other technical challenges or notable practices you would like to share with the international community?

APPENDIX C: SUBMITTED SURVEY RESPONSES BY COUNTRY

Belgium: Bel V and Tractebel

PART 1: RESPONDENT INFORMATION

Name:	**D. Gryffroy***
Country:	**Belgium**
Organization:	**Bel V**
Type of Organization:	**□ Regulatory Authority** **□ Utility** **□ Vendor** **□ Academic/Research** **x Other (please specify): Technical Support Organisation (TSO) of the Regulatory Authority**
Mailing Address:	**Bel V** **Rue Walcourt 148** **B-1070 Brussels** **(Belgium)**
E-mail Address:	**dries.gryffroy@belv.be**
Telephone Number:	

*** The following persons reviewed and supplemented this survey response:**

- **members of the Bel V technical staff:**
  Pieter De Gelder
  Didier Degueldre

- **for Tractebel Engineering S.A.:**
  Stanislas Mitaillé

PART 2: APPLICATION OF LEVEL 3 PSA

2.1 Does your country require the performance of Level 3 PSAs for nuclear installations? If yes, for what types of applications are Level 3 PSAs required?

Such requirement does currently not exist for nuclear installations in Belgium (see Royal Decree of 2011-11-30[1], which covers the transposition of WENRA Safety Reference Levels (2008)[2] for existing NPPs into Belgian regulations and also covers non-NPPs).

Level 3 PSA is not required either in the updated WENRA Reference Levels (September 2014)[3]. These updated WENRA Reference Levels will also be transposed into Belgian regulations (by means of a new Royal Decree).

2.2 With regard to the use of Level 3 PSA:

2.2.1. Do you perform, or are you considering performing Level 3 PSAs for nuclear installations? If you do, for what types of applications do you perform them?

The performance of Level 3 PSA for existing reactors is currently not considered in Belgium. The focus is currently on the extension of PSA Level 1 and Level 2 for internal and external hazards, which is requiring important resources.

2.2.2. What calculated metrics or results from Level 3 PSAs are used in these applications and what (if any) requirements, goals, or criteria are used to evaluate these results? What is the basis for these requirements, goals, or decision criteria?

Not applicable (see 2.1 and 2.2.1).

2.2.3. If you are not currently performing or considering performing Level 3 PSAs, have you previously considered performing them? If you have, what was the basis for your decision to not perform Level 3 PSA? Please list any barriers you perceive to performing and applying Level 3 PSA in your country.

The performance of Level 3 PSA has not been considered previously.

Currently, a major barrier to performing and applying Level 3 PSA is that all resources are currently used for PSA Level 1 and Level 2 developments with higher priority (see 2.2.1), linked to applicable regulatory requirements (Royal Decree of 2011-11-301**).**

Potential barriers to performing and applying Level 3 PSA could also be the high uncertainties of the results (considering the propagation of the Level 1 and Level 2 uncertainties as well) and the limited expected benefit in terms of potential backfits or safety improvements.

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1 Royal Decree of November 30, 2011, *Safety requirements for nuclear installations.* Available at: http://www.fanc.fgov.be/GED/00000000/3400/3493.pdf

2 Western European Nuclear Regulators Association (WENRA). *WENRA Reactor Safety Reference Levels* (January 2008). Available at: http://www.wenra.org/media/filer_public/2012/11/05/list_of_reference_levels_january_2008.pdf

3 Western European Nuclear Regulators Association (WENRA). WENRA Reactor Safety Reference Levels (September 2014). Available at: http://www.wenra.org/media/filer_public/2016/07/19/wenra_safety_reference_level_for_existing_reactors_september_2014.pdf

2.2.4. What (if any) alternative methods do you use to estimate offsite public risks attributable to accidental releases of radiological materials from nuclear installations, and for what types of applications are they used?

Not applicable (see 2.1 and 2.2.1).

2.3 One application that could potentially benefit from Level 3 PSAs is the siting of nuclear installations, including establishing the size and boundary (shape) for each emergency planning or protective action zone. Are Level 3 PSAs used to support the siting of nuclear installations in your country? If they are, how are they used?

The latest siting of NPPs in Belgium dates from the early eighties and at that time no requirement existed for performing and applying Level 3 PSA for siting. Since then, no requirements have been published for Level 3 PSA to support siting of nuclear power plants or other nuclear installations in Belgium. In this context, it should also be noted that the nuclear phase-out law of 2003-01-31 excludes building new NPPs in Belgium.

However, for new "class I" nuclear installations (including both NPPs and non-NPPs and defined in the Royal Decree of 2001-07-201[1]), a Guidance on Safety Demonstration[2] has been developed by the Belgian regulatory authority (FANC), requiring dose assessments for different Design Basis Categories (DBC) to demonstrate that doses to individuals located at any point along the boundaries of specified areas around a nuclear installation for specified periods of time following postulated fission product releases are within prescribed "quantitative radiological safety objectives".

Because of the nuclear phase-out law of 2003-01-31, this Guidance is only applied to new nuclear installations which are not NPPs.

2.4 With regard to nuclear power plant emergency planning zones:

2.4.1. What process is used to establish sizes and boundaries for emergency planning zones around a nuclear installation in which arrangements shall be made at the preparedness stage for effectively taking protective and other response actions?

There is no formalized process implemented to define the size of the EPZ used around the nuclear installations (NPP and other facilities) in the context of the emergency preparedness & response arrangements. These EPZs are designed as circles of appropriate radius around nuclear installations and their dimension depends, among others, on the potential source term specific for each facility/installation and the generic intervention criteria adopted for a specific protective action (evacuation, sheltering, iodine thyroid blocking, …).

The present EPZs (established in 2003) are currently reassessed in the framework of the revision of the Belgian national EP&R. The results and principles defined in the "HERCA-WENRA approach for a better cross-border coordination of protective actions during the early phase of a nuclear accident" (document issued in Oct-2014) are of course used in that context.

[1] Royal Decree of July 20, 2001, *General Regulation on the Protection of the Population, the Workers and the Environment Against the Danger of Ionizing Radiation*. Available at: http://www.fanc.fgov.be/GED/00000000/3400/3492.pdf

[2] FANC - Class I Guidances - *Safety demonstration of new class I nuclear installations: approach to Defence-in-Depth, radiological safety objectives and the application of a graded approach to external hazards* (February 2015). Available at: http://www.fanc.fgov.be/GED/00000000/3800/3883.pdf.

The feedback from nuclear emergency exercises clearly demonstrates that the transposition of the actual emergency intervention zone/sector into an operational response area is not straightforward and leads to lengthy discussions between the experts in charge of the assessment of the emergency situation and the decision makers and between the decision makers and the local authorities. In order to circumvent this difficulty and avoid unnecessary delays, Belgium recently developed a practical concept consisting in dividing the EPZ into blocks of reasonable dimensions, not too large but not too small either, in order to allow enough flexibility when determining the area(s) where protective actions are needed. These arrangements taken at the preparedness stage are expected to improve the efficiency of the response actions.

2.4.2. What stakeholder groups are involved in the process and what are their respective responsibilities?

The federal and local authorities are involved together with the nuclear safety authority (FANC & Bel V, as supporting TSO) in the definition of the EPZs. In addition, the federal and local police services have been deeply involved (through an iterative process) in the implementation of the concept of dividing the EPZ into operational blocks (see 2.4.1 above).

2.4.3. Under what conditions (if any) can emergency planning zones be reduced in size? Can a Level 3 PSA be used to establish a probabilistic cut off for events that need to be considered for emergency planning purposes?

Not applicable.

2.5 Another application that could potentially benefit from Level 3 PSAs is the development of safety goals or risk acceptance criteria. Are Level 3 PSAs used to support this type of application in your country? If they are, how are they used?

Not applicable.

2.6 Offsite radiological consequence analyses may be performed to support applications that are not related to Level 3 PSA. Please list any additional applications supported by offsite radiological consequence analyses that you support or perform.

Offsite radiological consequence analyses are also performed to support incident response applications and to support emergency planning exercises. In these applications, dose assessments are performed in real-time to inform or evaluate offsite protective action recommendations.

PART 3: LEVEL 3 PSA MODELING ISSUES AND TECHNICAL CHALLENGES

Since performance of Level 3 PSA for existing reactors is currently not considered in Belgium, questions 3.1 to 3.31 are considered not applicable.

General

3.1 What computer code(s) do you use, or are you developing, to perform the tasks listed below?

3.1.1. Offsite radiological consequence analyses:
3.1.2. Risk characterization:

3.2 With regard to the scope of Level 3 PSAs:

3.2.1. What internal and external accident sequence initiating events and hazards are considered in developing a Level 3 PSA?
3.2.2. Do you account for correlation between causes of accident sequence initiating events and offsite phenomenological and consequence modelling? If you do, how is this correlation treated?

3.3 Are you participating in any ongoing research and development activities related to offsite radiological consequence analyses? If you are, please provide a brief description of these activities.

Radionuclide Release Characterization

3.4 With regard to the interface between Level 2 and Level 3 PSA:

3.4.1. How do you handle the transition from Level 2 PSA analyses to Level 3 PSA analyses? What (if any) difficulties have you encountered in this area?
3.4.2. If you use representative release categories or source term groups, what criteria do you use to assign radiological release sequences in the Level 2 PSA model to release categories or source term groups? How do you address potential inter-sequence variability within each representative release category or source term group?
3.4.3. What process do you use to define release fraction timing (time and duration of release) and release truncation time? Do you consider onsite severe accident mitigation actions or offsite emergency response actions in this process? If you do, how are they treated?

3.5 What radionuclides do you use to characterize the offsite radiological consequences attributed to accidental releases of radiological materials? What is your basis for selecting these radionuclides?

3.6 Do you consider releases from multiple units or radiological sources co-located at the same site? If you do, how are these treated?

3.7 Do you perform sensitivity or uncertainty analyses on release categories or release fractions? If you do, what methods are used and what is the basis for using them?

Meteorological Data

3.8 How do you account for temporal and spatial variability in meteorological conditions? If you use a subset (sampling) of available meteorological data to serve as a representative data set, what criteria do you use for selecting these representative data?

3.9 Do you perform sensitivity or uncertainty analyses on meteorological conditions? If you do, what methods are used and what is the basis for using them?

Environmental Transport and Dispersion

3.10 What atmospheric transport and dispersion (ATD) model(s) do you use? What process and criteria do you use to select: (1) an ATD model; and (2) a time scale for updating calculations to account for time dependence?

3.11 What process and criteria do you use to define the boundaries and intervals of the spatial grid or domain used for performing offsite consequence calculations?

3.12 With regard to the spatial modelling around a nuclear power plant for Level 3 PSA:

3.12.1. What information sources do you use to develop geographical or topographical parameters? Do you use generic or site-specific data?
3.12.2. Does the ATD response vary spatially with respect to varying topographic parameters? What is the spatial resolution of the ATD and do topographic parameters of the ATD vary within this resolution?

3.13 Do you consider radiological releases to water sources and aqueous transport and dispersion phenomena? If you do, how are these treated? If you have previously considered modelling radiological releases to water sources and aqueous transport and dispersion phenomena, but decided not to do so, what was the basis for your decision?

Protective Action (Countermeasure) Modelling

3.14 How do you define, and what protective actions (countermeasures) do you model for each of the nuclear or radiological incident or accident response phases listed below?

3.14.1. Early/Emergency Phase:
3.14.2. Intermediate Phase:
3.14.3. Late/Recovery Phase:

3.15 Do you model population groups with different protective action (countermeasure) behaviours? If you do, how do you define them? To what extent do you account for population density or age and gender distribution effects?

3.16 Do you use probabilistic models of protective action (countermeasure) behaviours that model the probabilities of success or failure for protective actions (countermeasures)? If you do, please describe the models and their bases.

3.17 What information sources do you use as a technical basis for protective action (countermeasure) modelling? Do you use generic or site-specific data?

Radiological Exposure and Dose Assessment

3.18 How do you define potentially exposed populations or cohorts? Do you consider onsite (e.g., worker) populations or potentially sensitive, vulnerable, or critical groups? If you do, what is your basis for including these groups, and how are they modelled?

3.19 What exposure pathways do you model? What is your basis for selecting these pathways? What exposure duration is assumed in your models?

3.20 What information sources do you use to develop exposure and dose conversion factors? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

Radiological Health Effects

3.21 Do you model and estimate radiological health effects in your offsite radiological consequence analyses? If you do, please respond to questions 3.22 through 3.24. If you do not, please describe any other methods you use to address the offsite radiological health effects attributable to possible accident scenarios involving nuclear installations and proceed to question 3.25.

3.22 What types of early and latent health effects (including fatal and non-fatal effects) do you model? What target organs do you use for the associated health effects models?

3.23 What dose-response model(s) do you use to estimate the numbers of latent health effects attributable to radiological doses caused by accidental releases from nuclear installations, and what is the basis for their use? What (if any) methods do you use to account for uncertainty about the true dose-response relationship for exposures to low levels of ionizing radiation?

3.24 What information sources do you use to develop input parameters for the health effect models? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

Economic Consequences

3.25 Do you model and estimate economic consequences in your offsite radiological consequence analyses? If you do, please respond to questions 3.26 and 3.27. If you do not, please describe any other methods you use to address the offsite economic consequences attributable to possible accident scenarios involving nuclear installations and proceed to question 3.28.

3.26 With regard to potential economic consequences considered within a Level 3 PSA:

3.26.1. What types of economic consequences do you model? What process do you use to select cost categories and models for estimating economic consequences?
3.26.2. If you translate radiological health effects into economic consequences, please describe your process for doing so.
3.26.3. If you model land contamination, please describe your process for estimating the health, environmental, or economic consequences attributed to land contamination.

3.27 What information sources do you use to develop cost parameters for economic consequence models? Do you use generic, region-specific, or site-specific data?

Consequence Quantification and Reporting

3.28 What process and criteria do you use for selecting consequence metrics for quantification and reporting, including spatial intervals or distances from release points?

Risk Characterization

3.29 What process and criteria do you use for selecting risk metrics for effectively communicating Level 3 PSA results, including metrics designed to measure the effects of radiological releases involving multiple units or radiological sources co-located at the same site, if applicable?

3.30 For each of the groups listed below, what methods do you use for presenting and communicating risk results and the uncertainty in risk results? How do you present the results from low-probability/high-consequence events in an understandable context for each group? Who is responsible for communicating the results to each group?

(1) Decision makers:
(2) General Public:

3.31 With regard to the treatment of uncertainties:

3.31.1. Do you perform sensitivity or uncertainty analyses as part of risk estimation? If you do, what process and methods do you use?
3.31.2. How do you determine which parameters will be varied or will have uncertainty distributions specified for propagating uncertainty? What parameters do you evaluate?
3.31.3. To what extent do you consider the effect of correlation on parameters?

Other

3.32 Are there any other questions that you believe should have been asked in this survey? Is there any other information or are there any other technical challenges or notable practices you would like to share with the international community?

3.33

Canada: CNSC

PART 1: RESPONDENT INFORMATION

*

Name:	**Chantal Morin/Usha Menon***
Country:	**Canada**
Organization:	**Canadian Nuclear Safety Commission (CNSC)**
Type of Organization:	**X Regulatory Authority** **☐ Utility** **☐ Vendor** **☐ Academic/Research** **☐ Other (please specify): _____**
Mailing Address:	**280 Slater St. Ottawa, Ontario, Canada, K1P 5S9**
E-mail Address:	**chantal.morin@canada.ca and usha.menon@canada.ca**
Telephone Number:	**1-613-947-8222**

The following members of the CNSC technical staff reviewed and supplemented this example survey response:

Y. Akl, Probabilistic Safety Assessment and Reliability Division
Melanie Rickard, Directorate of Environment and Radiation Protection Thuy N'Guyen, Reactor Behaviour Division
Doug Miller, Directorate of Regulatory Improvements and Major Projects

PART 2: APPLICATION OF LEVEL 3 PSA

2.1 Does your country require the performance of Level 3 PSAs for nuclear installations? If yes, for what types of applications are Level 3 PSAs required?

The Canadian Nuclear Safety Commission does not require that a Level 3 PSA be performed. However, dose consequence assessments are performed as part of environmental assessments that are required, for example, for major projects such as new build licensing or major refurbishment of existing reactors. The dose consequence assessments examine bounding accidents and malfunctions to evaluate doses to ensure that they are below the prescribed dose limits in the *Radiation Protection Regulations*. Further, dose consequence assessments are also performed as part of requirements for deterministic safety assessment as described in CNSC REGDOC-2.4.1 (entitled: Safety Analysis: Deterministic Safety Analysis). This document sets out requirements of the Canadian Nuclear Safety Commission for deterministic safety analysis for nuclear power plants and small reactor facilities.

2.2 With regard to the use of Level 3 PSA:

2.2.1. Do you perform, or are you considering performing Level 3 PSAs for nuclear installations? If you do, for what types of applications do you perform them?

CNSC does not currently perform neither requires a Level 3 PSA. Historically the CNSC (at the time Atomic Energy Control Board) licensing requirements led some licensees to do some level 3 PSA. Current requirements only necessitate level 2 PSA.

2.2.2. What calculated metrics or results from Level 3 PSAs are used in these applications and what (if any) requirements, goals, or criteria are used to evaluate these results? What is the basis for these requirements, goals, or decision criteria?

Only dose consequence assessments are performed (see response to 2.1.) No level 3 PSA are currently performed.

2.2.3. If you are not currently performing or considering performing Level 3 PSAs, have you previously considered performing them? If you have, what was the basis for your decision to not perform Level 3 PSA? Please list any barriers you perceive to performing and applying Level 3 PSA in your country.

Yes, the possibility of level 3 PSA was raised by many public interveners following Fukushima. So far the CNSC position is that safety goals based on Level 2 PSA results as well as using a deterministic safety assessments are sufficient to ensure safety. Level 3 PSA results are viewed as containing too much uncertainty

2.2.4. What (if any) alternative methods do you use to estimate offsite public risks attributable to accidental releases of radiological materials from nuclear installations, and for what types of applications are they used? [DERPA to review]

The Canadian Nuclear Safety Commission does not require that a Level 3 PSA be performed. However, dose consequence assessments are performed as part of environmental assessments that are required, for example, for major projects such as new build licensing or major refurbishment of existing reactors. The dose consequence assessments examine bounding accidents and malfunctions to evaluate doses to ensure that they are below the prescribed dose limits in the Radiation Protection Regulations. Further, dose consequence assessments are also performed as part of requirements for

deterministic safety assessment as described in CNSC REGDOC-2.4.1 (entitled: Safety Analysis: Deterministic Safety Analysis). This document sets out requirements of the Canadian Nuclear Safety Commission for deterministic safety analysis for nuclear power plants and small modular reactor facilities.

2.3 One application that could potentially benefit from Level 3 PSAs is the siting of nuclear installations, including establishing the size and boundary (shape) for each emergency planning or protective action zone. Are Level 3 PSAs used to support the siting of nuclear installations in your country? If they are, how are they used?

Currently full Level 3 PSA are not used to support the siting of nuclear installations. See Answer to question 2.1

2.4 With regard to nuclear power plant emergency planning zones:

2.4.1. What process is used to establish sizes and boundaries for emergency planning zones around a nuclear installation in which arrangements shall be made at the preparedness stage for effectively taking protective and other response actions?

2.4.2. What stakeholder groups are involved in the process and what are their respective responsibilities?

Provincial Authorities

2.4.3. Under what conditions (if any) can emergency planning zones be reduced in size? Case-by-case assessment

2.4.4. Can a Level 3 PSA be used to establish a probabilistic cut off for events that need to be considered for emergency planning purposes**?**

At this time, Level 3 PSAs are not used for this application.

2.5 Another application that could potentially benefit from Level 3 PSAs is the development of safety goals or risk acceptance criteria. Are Level 3 PSAs used to support this type of application in your country? If they are, how are they used?

Not at this time

2.6 Offsite radiological consequence analyses may be performed to support applications that are not related to Level 3 PSA. Please list any additional applications supported by offsite radiological consequence analyses that you support or perform.

Dose consequence assessments are performed as part of environmental assessments that are required, for example, for major projects such as new build licensing or major refurbishment of existing reactors. The dose consequence assessments examine bounding accidents and malfunctions to evaluate doses to ensure that they are below the prescribed dose limits in the Radiation Protection Regulations. Further, Dose consequence analysis in the context of accidents/emergencies is performed to confirm deterministic safety analysis acceptance criteria (e.g., Design Basis Accident consequences should be less than 20mSv).

PART 3: LEVEL 3 PSA MODELING ISSUES AND TECHNICAL CHALLENGES

General

3.1 What computer code(s) do you use, or are you developing, to perform the tasks listed below?

3.1.1. Offsite radiological consequence analyses:

Depending on the application (emergency response and related exercises or safety assessment) different codes are used. MAACS 2 is generally used by licensees for safety assessment and RASCAL and ERP (Emergency Response Projection, a code used by the Province of Ontario) are used in emergency response or related exercises. Note these applications are not within the context of PSA III.

3.1.2. Risk characterization:

Level 1 and 2 PSA use CAFTA

3.2 With regard to the scope of Level 3 PSAs: **N/A**

3.2.1. What internal and external accident sequence initiating events and hazards are considered in developing a Level 3 PSA?

N/A

3.2.2. Do you account for correlation between causes of accident sequence initiating events and offsite phenomenological and consequence modelling? If you do, how is this correlation treated?

N/A

3.3 Are you participating in any ongoing research and development activities related to offsite radiological consequence analyses? If you are, please provide a brief description of these activities.

The CNSC is involved in a number of post-accident/recovery phase initiatives, including participation in the IAEA's Modelling and Data for Radiological Impact Assessments Programme. Working groups within this initiative are studying a variety of topics, including model testing and comparison for accidental tritium releases and the use of decision-making tools in the post-accident recovery phase.

Radionuclide Release Characterization

3.4 With regard to the interface between Level 2 and Level 3 PSA:

3.4.1. How do you handle the transition from Level 2 PSA analyses to Level 3 PSA analyses? What (if any) difficulties have you encountered in this area?

3.4.2. If you use representative release categories or source term groups, what criteria do you use to assign radiological release sequences in the Level 2 PSA model to release categories or source term groups? How do you address potential inter-sequence variability within each representative release category or source term group?

Level 3 PSAs are not performed; however, the Level 2 PSA release categories are generally defined as in the table below, by the licensees.

Table 2-5: Proposed Release Categorization Scheme

Release Category #	Description	Definition
PI-RC0[+]	Early[++] very large release	> ~3% core inventory of I-131 occurring mainly within 24 hours
PI-RC1	Late very large release	> ~3% core inventory of I-131 occurring mainly after 24 hours
PI-RC2	Early RD-152 Large Release	Mixture of fission products containing > 10^{14} Bq of Cs-137 but <~3% core inventory of I-131 (PI-RC0) occurring mainly within 24 hours
PI-RC3	Late RD-152 Large Release	Mixture of fission products containing > 10^{14} Bq of Cs-137 but < ~3% core inventory of I-131 (PI-RC1)occurring mainly after 24 hours
PI-RC4	Early RD-152 Small Release	Mixture of fission products containing > 10^{15} Bq of I-131 but < 10^{14} Bq of Cs-137 occurring mainly within 24 hours
PI-RC5	Late RD-152 Small Release	Mixture of fission products containing >10^{15} Bq of I-131 but < 10^{14} Bq of Cs-137 occurring mainly after 24 hours
PI-RC6	Mitigated release	Mixture of fission products containing > 10^{14} Bq of I-131 but < 10^{15} Bq of I-131 occurring mainly after 24 hours
PI-RC7	CET Success Path	Slow release containing < 10^{14} Bq of I-131
PI-RC8[+++]	Basemat Melt-through	Penetration of FMD concrete basemat due to CCI

\+ PI = Plant Identifier.
++ Early < 24 hours, late > 24 hours after reactor trip.
+++ Sequences assigned to this category may also be assigned to other release categories

3.4.3. What process do you use to define release fraction timing (time and duration of release) and release truncation time? Do you consider onsite severe accident mitigation actions or offsite emergency response actions in this process? If you do, how are they treated?

N/A

3.5 What radionuclides do you use to characterize the offsite radiological consequences attributed to accidental releases of radiological materials? What is your basis for selecting these radionuclides?

Expectations for dose assessment are best described in a Canadian Standards Association Standard entitled: *N288.2 Guidelines for Calculating the Radiological Consequences to the Public of a Release of Airborne Radioactive Material for Nuclear Reactor Accidents*. This standard is intended to be used by Nuclear Power Facilities for safety assessment as well as real time emergency response. The requirements include that the radionuclides considered in the assessment shall contribute more than 95% to the total dose. A justification shall be provided for the specific radionuclides selected for inclusion in a given accident

3.6 Do you consider releases from multiple units or radiological sources co-located at the same site? If you do, how are these treated?

Multiple units or radiological sources co-located at the same site are not yet considered in level 2 PSA. However multi-unit impacts, such as some cross link on support systems are modelled in the level 1 PSA.

3.7 Do you perform sensitivity or uncertainty analyses on release categories or release fractions? If you do, what methods are used and what is the basis for using them?

Level 2 PSA results contain sensitivity analysis on release categories *

"The objective of the uncertainty analysis is to determine whether the uncertainty associated with consequence assessment could potentially result in a change in results that could impact the conclusions of the Level 2 PRA. The uncertainty analysis approach provides a reasonable tool to gauge the robustness of the binning of various PDS sequences into the respective release categories. For each representative sequence in RC2, RC4, RC6 and RC7 the following distributions are reported:

- **Cs137 and I131 releases,**
- **Corium relocation time into the FMD, and**
- **The total mass of hydrogen produced.**

The uncertainty analysis study can only provide reasonable confidence to the upper limit of the distributions. A probability density function to correlate specific output trends with input distributions is not possible, as the sample size is too small. However, in some cases general trends can be formulated on the overall uncertainty effects of the input parameters on the output distributions."[1]

Meteorological Data

All of this (3.8 to 3.12) detail is covered in extensive detail in CSA N228.2 for emergency response and accident assessment purposes.

3.8 How do you account for temporal and spatial variability in meteorological conditions? If you use a subset (sampling) of available meteorological data to serve as a representative data set, what criteria do you use for selecting these representative data?

N/A

3.9 Do you perform sensitivity or uncertainty analyses on meteorological conditions? If you do, what methods are used and what is the basis for using them?

N/A

1 From Bruce B level 2 PSA report

Environmental Transport and Dispersion

3.10 What atmospheric transport and dispersion (ATD) model(s) do you use? What process and criteria do you use to select: (1) an ATD model; and (2) a time scale for updating calculations to account for time dependence?

3.11 What process and criteria do you use to define the boundaries and intervals of the spatial grid or domain used for performing offsite consequence calculations?

3.12 With regard to the spatial modelling around a nuclear power plant for Level 3 PSA:

3.12.1. What information sources do you use to develop geographical or topographical parameters? Do you use generic or site-specific data?
3.12.2. Does the ATD response vary spatially with respect to varying topographic parameters? What is the spatial resolution of the ATD and do topographic parameters of the ATD vary within this resolution?

3.13 Do you consider radiological releases to water sources and aqueous transport and dispersion phenomena? If you do, how are these treated? If you have previously considered modelling radiological releases to water sources and aqueous transport and dispersion phenomena, but decided not to do so, what was the basis for your decision?

No, we don't consider it for now.

Protective Action (Countermeasure) Modelling

N/A, since no Level 3 PSAs are performed

3.14 How do you define, and what protective actions (countermeasures) do you model for each of the nuclear or radiological incident or accident response phases listed below?

The answer here is provided with respect to dose consequence assessment that is not related to Level III PSA (as it is not required-discussed above), but generally, with regards to assessments that support other initiatives or processes, such as those practiced in emergency response exercises. Health Canada and the applicable provinces as well as some Nuclear Power Plants, have models to simulate radiation dispersion, transport and dose consequences and typically those assessments focus on the early phase; mainly the first seven days. Broadly speaking, Canada follows guidance in IAEA GSR Part 7 for the emergency phase. The actions to be taken in the transition from the emergency phase to recovery and for recovery are currently being developed in Canada. To the extent possible, guidance from the ICRP, the IAEA and countries with well-developed frameworks on recovery are being relied upon.

3.14.1. Early/Emergency Phase:
3.14.2. Intermediate Phase:
3.14.3. Late/Recovery Phase:

3.15 Do you model population groups with different protective action (countermeasure) behaviours? If you do, how do you define them? To what extent do you account for population density or age and gender distribution effects?

Some models do account for different age classes, typically those defined by the ICRP. The most conservative results (resulting from assessment of multiple age groups) are typically applied to the entire population for urgent and early protective actions. While not yet formalized, the expectations for the recovery phase would follow those of the ICRP on such dose assessment. As

such, the dose assessment for those individuals or groups examined would depend on different habits and behaviours and thus heterogeneity would be assumed. That is, to the extent practical, different ages, sex and lifestyle habits would be considered.

3.16 Do you use probabilistic models of protective action (countermeasure) behaviours that model the probabilities of success or failure for protective actions (countermeasures)? If you do, please describe the models and their bases.

These are not considered at this time. Since this is an area of current development, it is unknown if this approach would be used or advisable. At this time, dose consequence assessment for urgent/early countermeasures is not based on a probabilistic modelling of protective actions and associated predictions of success or failure. Notwithstanding, there is an expectation that decision makers consider risk benefit in taking countermeasures.

3.17 What information sources do you use as a technical basis for protective action (countermeasure) modelling? Do you use generic or site-specific data?

The modelling used will depend on the off-site authority involved as each province as well as the federal authority (Health Canada) have different modelling capabilities. In terms of the protective measures taken and their basis, this will also vary somewhat depending on the responsible authority. However, in all cases, the countermeasures and their basis are consistent with IAEA GSR Part 7. Each province (with a Nuclear Power Plant) as well as Health Canada has recommendations and plans with regards to off-site countermeasures.

Radiological Exposure and Dose Assessment

3.18 How do you define potentially exposed populations or cohorts? Do you consider onsite (e.g., worker) populations or potentially sensitive, vulnerable, or critical groups? If you do, what is your basis for including these groups, and how are they modelled?

The concept of receptors and associated requirements are defined in CSA 288.2. The receptor should be selected based on the safety objective of the calculation. For safety assessments, a representative person shall be defined and shall include at a minimum, a hypothetical adult located at the site boundary in the downwind direction at the time of the event. For safety assessments involving collective doses and for probabilistic safety assessments, the population distribution around the reactor shall be used. In terms of protective measures being considered at the preparedness and response stages, these should be evaluated on the basis of doses calculated for the vulnerable populations. Vulnerable populations can be defined as members of the population who have higher radiation sensitivity or additional needs before, during or after a release. In practice, vulnerable populations could be viewed as expecting, pregnant or nursing mothers or vulnerable based on age. For example, models run doses based on different age classes and the protective measure would be based on the highest dose from one age group.

3.19 What exposure pathways do you model? What is your basis for selecting these pathways? What exposure duration is assumed in your models?

CSA N288.2 requires that cloud shine, ground shine and inhalation pathways be modelled, including re-suspension, where applicable. The basis for these pathways is, in part, in the assumption that the release is airborne and is less than 30 days in duration. There is also as assumption that effective food controls are in place and that dose from ingestion of contaminated food and water are minimized. The pathways considered represent standard practice.

3.20 What information sources do you use to develop exposure and dose conversion factors? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

For those most part, DCFs are based on those available from the ICRP and the U.S EPA. Some codes use DCFs that are not based on the most recent DCFs available but are usually considered acceptable nonetheless, based on the intended use and the small amount of "error" introduced in using these DCFs. Where applicable, age specific DCFs are available within some codes (e.g., ERP).

Radiological Health Effects

3.21 Do you model and estimate radiological health effects in your offsite radiological consequence analyses? If you do, please respond to questions 3.22 through 3.24. If you do not, please describe any other methods you use to address the offsite radiological health effects attributable to possible accident scenarios involving nuclear installations and proceed to question 3.25.

Standard N288.2 recommends the use of BIER VII risk coefficients that are based on the risk of cancer mortality. Further, CSA N228.2 recommends that for severe accident assessments, deterministic effects should be calculated and describes how this is done.

Further, after the accident at Fukushima, the CNSC's Commission requested staff to undertake an assessment of health and environmental consequences of accident scenarios that are more severe than typically considered in environmental assessment screening reports. The resulting study is titled *Study of Consequences of a Hypothetical Severe Nuclear Accident and Effectiveness of Mitigation Measures* (referred to in section 3.22 through 3.24 as "the study".

3.22 What types of early and latent health effects (including fatal and non-fatal effects) do you model? What target organs do you use for the associated health effects models?

In terms of human health, the focus of the study was to examine the possible impact on cancer incidence. Cancer is described as a stochastic or latent effect where the probability of occurrence is proportional to exposure or dose. Deterministic effects, such as acute radiation sickness were not examined as the estimated doses in this study were below the thresholds for these types of effects. The target organs considered in this study were the colon, the bone marrow and the thyroid. The colon was the organ used to evaluate the risk of developing a cancer in the category of "all cancers combined". The colon was chosen to represent the absorbed dose to all organs because it can illustrate changes in dose to deeper tissues that experience shielding from the more superficial tissues of the body, it is relatively centred in the body from a physiological perspective, it is a highly radiosensitive organ, and it is not sex-specific. The category of all cancers combined includes many cancers, some of which are more radiosensitive than others. This category provides a general indication of whether there is an increased risk overall. Bone marrow was used to evaluate the risk of developing leukaemia. Leukaemia is a very radiosensitive type of cancer and radiation-induced cases of leukaemia could appear as soon as two years after exposure providing a very strong indication of the occurrence of an effect actually occurring. The thyroid was used to evaluate the risk of developing thyroid cancer. Thyroid cancer was chosen because it was one of the main health effects that resulted from Chernobyl. Thyroid cancer is also very radiosensitive and radiation-induced cases of thyroid cancer could appear as soon as five years after exposure.

3.23 What dose-response model(s) do you use to estimate the numbers of latent health effects attributable to radiological doses caused by accidental releases from nuclear installations, and

what is the basis for their use? What (if any) methods do you use to account for uncertainty about the true dose-response relationship for exposures to low levels of ionizing radiation?

For this study, risk calculations were performed using the U.S National Cancer Institute's radiation risk assessment tool known as "RadRAT". The risk models used by the risk calculator are broadly based on those developed by the Biological Effects of Ionizing Radiation (BEIR) VII committee for estimating lifetime risk for radiation-related cancer. A linear dose-response model was chosen for all cancers combined and thyroid cancer, and a linear-quadratic risk model was chosen for leukaemia. RadRAT is based on a US population. Canada was deemed to have similar enough life expectancy and cancer rates to the US population. The RadRAT program reports the mean risk and 90 per cent uncertainty interval from the resulting distribution. Furthermore, for the solid cancer risk models, the assumption in RadRAT follows that of BEIR VII by applying an uncertain Dose and Dose Rate Effectiveness Factor (DDREF) for all chronic exposures and for acute exposures below 100 mGy described by a lognormal with a geometric mean (GM) of 1.5 and geometric standard deviation (GSD) of 1.35.

3.24 What information sources do you use to develop input parameters for the health effect models? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

It is important to note that the dose assessment and human health risk assessment were performed by different organizations. With regards to the human health risk assessment the parameters are fixed by the program. RadRAT is publicly available and designed to be very user friendly. The input parameters include demographic information (gender and birth year) and exposure information (number of exposure events, exposure year, organ exposed, exposure rate (acute or chronic), and the distribution type (lognormal, normal, triangular or log triangular, uniform or log uniform, or fixed value).

Economic Consequences

3.25 Do you model and estimate economic consequences in your offsite radiological consequence analyses? If you do, please respond to questions 3.26 and 3.27. If you do not, please describe any other methods you use to address the offsite economic consequences attributable to possible accident scenarios involving nuclear installations and proceed to question 3.28.

There is no Level 3 PSA performed currently and no modelling of economic consequences. However, cost-benefit analyses were performed on Level 2 PSA results by estimating a cost for a Large Release event and comparing the delta LRF with cost of modifications.

3.26 With regard to potential economic consequences considered within a Level 3 PSA:

N/A since no level 3 PSA performed

3.26.1. What types of economic consequences do you model? What process do you use to select cost categories and models for estimating economic consequences?
3.26.2. If you translate radiological health effects into economic consequences, please describe your process for doing so.
3.26.3. If you model land contamination, please describe your process for estimating the health, environmental, or economic consequences attributed to land contamination.

3.27 What information sources do you use to develop cost parameters for economic consequence models? Do you use generic, region-specific, or site-specific data?

N/A

Consequence Quantification and Reporting

3.28 What process and criteria do you use for selecting consequence metrics for quantification and reporting, including spatial intervals or distances from release points?

Risk Characterization

3.29 What process and criteria do you use for selecting risk metrics for effectively communicating Level 3 PSA results, including metrics designed to measure the effects of radiological releases involving multiple units or radiological sources co-located at the same site, if applicable?

N/A since no level 3 PSA are performed

3.30 For each of the groups listed below, what methods do you use for presenting and communicating risk results and the uncertainty in risk results? How do you present the results from low-probability/high-consequence events in an understandable context for each group? Who is responsible for communicating the results to each group?

N/A

(1) Decision makers:
(2) General Public:

3.31 With regard to the treatment of uncertainties:

See response to question 3.7

3.31.1. Do you perform sensitivity or uncertainty analyses as part of risk estimation? If you do, what process and methods do you use?
3.31.2. How do you determine which parameters will be varied or will have uncertainty distributions specified for propagating uncertainty? What parameters do you evaluate?
3.31.3. To what extent do you consider the effect of correlation on parameters?

Other

3.32 Are there any other questions that you believe should have been asked in this survey? Is there any other information or are there any other technical challenges or notable practices you would like to share with the international community?

Not at this time

Finland: VTT and STUK

PART 1: RESPONDENT INFORMATION

Name:	**Ilkka Karanta (VTT)***
Country:	**Finland**
Organization:	**VTT Technical Research Center of Finland Ltd (in cooperation with STUK, the Radiation and Nuclear Safety Authority in Finland)**
Type of Organization:	☐ Regulatory Authority ☐ Utility ☐ Vendor **x** Academic/Research ☐ Other (please specify): _____
Mailing Address:	**VTT, Box 1000, FI-02044 VTT, Finland**
E-mail Address:	Ilkka.karanta@vtt.fi
Telephone Number:	**+358 20 722 4509**

*** The following experts at VTT and the Finnish Radiation and Nuclear Safety Authority STUK reviewed and supplemented this survey response:**

Mikko Ilvonen (VTT)
Marko Marjamäki (STUK)
Jukka Rossi (VTT)
Jorma Sandberg (STUK)

PART 2: APPLICATION OF LEVEL 3 PSA

2.1 Does your country require the performance of Level 3 PSAs for nuclear installations? If yes, for what types of applications are Level 3 PSAs required?

Finland does not require the performance of Level 3 PSAs. Only level 1 and 2 PSAs are required.

2.2 With regard to the use of Level 3 PSA:

2.2.1. Do you perform, or are you considering performing Level 3 PSAs for nuclear installations? If you do, for what types of applications do you perform them?

VTT does not currently perform full-scope level 3 PSAs. VTT has performed offsite radiological consequence analyses, methodologically substantially similar to those performed in level 3 PSA. As a research institute, many of those analyses have been connected with applied research. As an example, the applicability of the integrated deterministic and probabilistic safety assessment (IDPSA) methodology to level 3 studies has been investigated.

2.2.2. What calculated metrics or results from Level 3 PSAs are used in these applications and what (if any) requirements, goals, or criteria are used to evaluate these results? What is the basis for these requirements, goals, or decision criteria?

Metrics used relate mainly to health effects from ionizing radiation, such as total population dose, dose to the most exposed individual, number of cancer deaths.

2.2.3. If you are not currently performing or considering performing Level 3 PSAs, have you previously considered performing them? If you have, what was the basis for your decision to not perform Level 3 PSA? Please list any barriers you perceive to performing and applying Level 3 PSA in your country.

As a research institute, VTT performs analyses primarily when commissioned to do so. Thus, the reason for not performing level 3 PSA analyses is one to which VTT's customers are in a better position to reply.

A reason for the fact that Level 3 PSA is not performed and applied more in Finland is that it is not required by law or the regulator. Nevertheless, both the regulator Radiation and Nuclear Safety Authority (STUK) and nuclear power companies (Fortum, Teollisuuden Voima) have performed level 3 PSA analyses.

A major barrier to performing level 3 PSA analyses is the computational burden of extensive analyses. Also, the tediousness of detailed analyses is a barrier.

2.2.4. What (if any) alternative methods do you use to estimate offsite public risks attributable to accidental releases of radiological materials from nuclear installations, and for what types of applications are they used?

Typically, it is demonstrated that even with a large release, some threshold dose is not exceeded in almost any weather condition.

2.3 One application that could potentially benefit from Level 3 PSAs is the siting of nuclear installations, including establishing the size and boundary (shape) for each emergency

planning or protective action zone. Are Level 3 PSAs used to support the siting of nuclear installations in your country? If they are, how are they used?

There is one older siting study conducted in the 1970s. In this study site alternatives for district-heating power plant were compared.

2.4 With regard to nuclear power plant emergency planning zones:

2.4.1. What process is used to establish sizes and boundaries for emergency planning zones around a nuclear installation in which arrangements shall be made at the preparedness stage for effectively taking protective and other response actions?

The zones around the Finnish nuclear facilities are defined in STUK's Regulation and Regulatory Guide YVL A.2s.

- **the *site area* comprises the area surrounding the plant, with only activities related to the plant, it extends to about 0.5 – 1 km from the plant**
- ***precautionary action zone* surrounds the site area and extends to about 5 km from the plant, land use restrictions are in force in this area (e.g. no dense settlement, no hospitals or such, no significant production that could be impacted by a severe nuclear accident or endanger the plant)**
- ***emergency planning zone* extending to about 20 kilometres from the plant; the zone is covered by a detailed external rescue plan for the protection of the public drawn up by authorities.**

The zones are established according to the above deterministic principles.

Decisions on the exact boundaries of the zones are made in connection with the licensing process and the general land use planning process. In addition, the environmental impact assessment supports the licensing process.

2.4.2. What stakeholder groups are involved in the process and what are their respective responsibilities?

The licensing process is coordinated by the Ministry of Employment and the Economy. The first licensing step is the government decision in principle for which a supporting statement from the local municipal council is an absolute precondition. The decision in principle on a nuclear project requires ratification by the Parliament. The construction and operating licenses are granted by the government. STUK's positive statement is a precondition at all licensing stages. ***All*** *the processes include public hearings where statements and opinions are asked from ministries, the Radiation and Nuclear Safety Authority (STUK), municipalities, local authorities, rescue services, non-governmental organizations and from the general public. More information on the Finnish licensing process can be found at the www address* ***http://tem.fi/en/environmental-impact-assessments*****.**

2.4.3. Under what conditions (if any) can emergency planning zones be reduced in size? Can a Level 3 PSA be used to establish a probabilistic cut off for events that need to be considered for emergency planning purposes?

Conditions for reducing the size of the zones have not been set in the regulations. The dimensions described above are approximate, but in practice, the reduction of the sizes has not been under discussion. If the boundary of the precautionary action zone goes

through a relatively densely populated village, in recent cases the zone has rather been expanded to include the whole village.

2.5 Another application that could potentially benefit from Level 3 PSAs is the development of safety goals or risk acceptance criteria. Are Level 3 PSAs used to support this type of application in your country? If they are, how are they used?

We are not aware the level 3 PSAs would have been used for such a purpose in Finland.

Probabilistic offsite dose assessments have been carried out with the source term of 100 TBq of Cs-137. Long term weather data statistics from the power plant sites were used. Results indicated that acute health effects are not expected and long-term restrictions on land use in large areas are avoided, which was the goal of the 100 TBq of Cs-137 limit set in 1991 for severe accidents.

2.6 Offsite radiological consequence analyses may be performed to support applications that are not related to Level 3 PSA. Please list any additional applications supported by offsite radiological consequence analyses that you support or perform.

Offsite radiological consequence analyses are used to support incident response planning and readiness at the regulatory agency STUK. Doses caused by the radioactive emissions of normal operation are calculated. Emergency preparedness cases are based usually on quite short time period and then weather is better known, easier to estimate and less probabilistic. Offsite radiological consequence analyses have also been used for research purposes. Also, the topics listed in footnote 11 are relevant to VTT; in such cases, a consequence analysis is called for.

PART 3: LEVEL 3 PSA MODELING ISSUES AND TECHNICAL CHALLENGES

General

3.1 What computer code(s) do you use, or are you developing, to perform the tasks listed below?

3.1.1. Offsite radiological consequence analyses:

VTT uses the following codes for offsite radiological consequence analyses. In-house codes: ARANO (straight-line atmospheric dispersion, dose assessment, health consequences, economic consequences), VALMA (atmospheric dispersion in dynamic weather conditions, dose assessment), DETRA (biospheric transportation). ARANO is the main tool for Level 3 PSA. Available also MACCS, RADTRAD and RASCAL

3.1.2. Risk characterization:

At VTT, tools for this are not available in the in-house codes.

3.2 With regard to the scope of Level 3 PSAs:

3.2.1. What internal and external accident sequence initiating events and hazards are considered in developing a Level 3 PSA?

Any that may result in large release with significant probability. Initiating events and hazards are specified in PSA level 1.

3.2.2. Do you account for correlation between causes of accident sequence initiating events and offsite phenomenological and consequence modelling? If you do, how is this correlation treated?

Such correlations are usually a part of the model. For example, if an external event is that ice has blocked a water intake, then only weather data for winter will be used in that scenario.

3.3 Are you participating in any ongoing research and development activities related to offsite radiological consequence analyses? If you are, please provide a brief description of these activities.

VTT is participating in a Nordic research project on Level 3 PSA called L3PSA. Other partners in the project are Swedish nuclear consultancies. The objectives of the project are to survey the use and utility of level 3 analyses, follow international standardization work in the field, conduct pilot studies to gain experience in level 3 PSA, and develop guidance for performing level 3 analyses. An industry survey has been conducted, two pilots have been constructed (one Swedish, one Finnish), and a guidance document has been written. The project will end this year.

Research on offsite radiological consequences is also carried out in CASA project by VTT.

Radionuclide Release Characterization

3.4 With regard to the interface between Level 2 and Level 3 PSA:

3.4.1. How do you handle the transition from Level 2 PSA analyses to Level 3 PSA analyses? What (if any) difficulties have you encountered in this area?

If the release fraction from Level 2 PSA is available, there is no problem. But MELCOR primarily outputs the released masses and if MELMACCS is not available, we use the ORIGEN code to convert masses to activities.

3.4.2. If you use representative release categories or source term groups, what criteria do you use to assign radiological release sequences in the Level 2 PSA model to release categories or source term groups? How do you address potential inter-sequence variability within each representative release category or source term group?

Usually in Level 2 PSA inventory is specified for nuclide groups (e.g. noble gases), categorisation of nuclides into release categories is based on chemical properties of nuclides.

3.4.3. What process do you use to define release fraction timing (time and duration of release) and release truncation time? Do you consider onsite severe accident mitigation actions or offsite emergency response actions in this process? If you do, how are they treated?

Depends on the code. In ARANO release duration is reduced to few hours, because weather is remaining the same during dispersion. Also, offsite emergency response actions cannot be changed during dispersion.

3.5 What radionuclides do you use to characterize the offsite radiological consequences attributed to accidental releases of radiological materials? What is your basis for selecting these radionuclides?

There are 84 nuclides available in ARANO. In the case of LWR release not all are significant.

Cs-137, I-131, other nuclides depending on the analysis. In VALMA, 496 radionuclides can be taken into account. Selection is based on significance from the dose effects point of view.

3.6 Do you consider releases from multiple units or radiological sources co-located at the same site? If you do, how are these treated? Adjacent releases are co-located at the same site.

VTT has analysed the Fukushima accident, but beyond that, releases from multiple units have not been considered.

3.7 Do you perform sensitivity or uncertainty analyses on release categories or release fractions? If you do, what methods are used and what is the basis for using them?

Yes, VTT has conducted uncertainty analyses on release fractions. The method used was developed by VTT and essentially concerns conducting computational experiments with different fractions, tabulating the results, constructing statistical models based on them, and using these models in simulations which are then analysed. The VALMA code has uncertainty analysis capabilities, but computation burden is high.

Meteorological Data

3.8 How do you account for temporal and spatial variability in meteorological conditions? If you use a subset (sampling) of available meteorological data to serve as a representative data set, what criteria do you use for selecting these representative data?

At VTT, this is handled differently in different in-house codes. VALMA can handle both temporal and spatial variability in meteorological conditions. The weather input is given to VALMA as series of weather observations at different weather masts (i.e. different locations). In ARANO, annual weather data from one meteorological mast is converted into joint frequency matrix of annual weather statistics. Doses are calculated in all specified weather situations and the result is weighted with the probability of the weather situation. Sampling is not needed.

3.9 Do you perform sensitivity or uncertainty analyses on meteorological conditions? If you do, what methods are used and what is the basis for using them?

VTT has performed uncertainty analysis on wind speed, wind direction and precipitation. Statistical models for these have been constructed and used in simulations.

Environmental Transport and Dispersion

3.10 What atmospheric transport and dispersion (ATD) model(s) do you use? What process and criteria do you use to select: (1) an ATD model; and (2) a time scale for updating calculations to account for time dependence?

The basic model ARANO for Level 3 PSA is a Gaussian straight-line dispersion model in which vertical dispersion is replaced with the Kz model. ARANO is fast and versatile providing e.g. countermeasures and calculation of health effects. The Lagrangian trajectory model VALMA was initially prepared for an emergency preparedness tool, but currently it is under development to enable also risk calculations. VALMA can work with many kinds of weather data, starting from single-point measurements at the weather mast of an NPP (or several masts) and ending with Monte Carlo particles (even a limited number) that can be calculated, based on NWP (numerical weather prediction) models.

3.11 What process and criteria do you use to define the boundaries and intervals of the spatial grid or domain used for performing offsite consequence calculations?

In ARANO, environmental data is given in polar coordinates (segmented by radial lines - r,Θ). In this annular grid the angle size is 30 degrees (fixed) which means that data is given in 12 sectors for distance intervals defined by the user. Spatial intervals can be obtained case by case. Typically, spatial grid is dense near the source point, but sparse at longer distances.

3.12 With regard to the spatial modelling around a nuclear power plant for Level 3 PSA:

3.12.1. What information sources do you use to develop geographical or topographical parameters? Do you use generic or site-specific data?

Topographic parameters are not used in the codes ARANO and VALMA used by VTT. The ARANO code uses site-specific meteorological data as well as site-specific population, agricultural, and economic data estimated on a polar grid. VALMA code uses weather data from SILAM, a Finnish code; SILAM takes topography into account, and thus topography is 'in-built' into its weather data.

3.12.2. Does the ATD response vary spatially with respect to varying topographic parameters? What is the spatial resolution of the ATD and do topographic parameters of the ATD vary within this resolution?

Topographic parameters are not used in the codes ARANO and VALMA used by VTT.

3.13 Do you consider radiological releases to water sources and aqueous transport and dispersion phenomena? If you do, how are these treated? If you have previously considered modelling radiological releases to water sources and aqueous transport and dispersion phenomena, but decided not to do so, what was the basis for your decision?

ARANO or VALMA do not have model for water bodies. Releases to water sources and aqueous transport are handled in the DETRA code from the biospheric transport point of view. DETRA uses compartment models.

Protective Action (Countermeasure) Modelling

3.14 How do you define, and what protective actions (countermeasures) do you model for each of the nuclear or radiological incident or accident response phases listed below?

3.14.1. Early/Emergency Phase

In early/emergency phase of an accident short-term countermeasures are: sheltering, evacuation and iodine tablets. In ARANO there are two parameters affecting the evacuation: distance and evacuation time. The distance means that all the inhabitants are evacuated up to that range instantly after the time given for the evacuation. The time is calculated since the plume has reached the point under consideration. If the time value is 0, the population has been evacuated before the plume is spread to the point and dose to the population is 0. The parameter "warning time" affects evacuation. If there is a sure knowledge that the release will start after a certain time period, there is a period of warning time available for the initiation of evacuation before the release to the environment begins. Also, it is possible to define combined effects, which means that population can be divided into groups with different shielding conditions and the combined result is the sum of all elements with their weights. In ARANO there is not a direct input for stable iodine tablets, but it can be taken into account in calculation by reducing internal dose from iodine isotopes.

3.14.2. Intermediate Phase:

It can be assumed that after the early phase of an accident there is sufficiently time to consider and evaluate different countermeasures and their combinations. Relocation in this phase refers to moving people away from the contaminated area for a longer time period (weeks, months, years). Food prohibition is based on ingestion dose levels or ground concentrations.

3.14.3. Late/Recovery Phase:

In late/recovery phase possible term countermeasures include relocation, land decontamination and food ban. In the case that external radiation dose exceeds a limit value a potential protection measure could be decontamination. If the cleaned area after decontamination is still too contaminated, staying there shall be reduced or denied. Criterion for the contaminated area can be based on the external radiation dose from fallout during 30 years.

3.15 Do you model population groups with different protective action (countermeasure) behaviours? If you do, how do you define them? To what extent do you account for population density or age and gender distribution effects?

See response to 3.14.1 Age- and gender-specific variability in protective action behaviours are not considered.

3.16 Do you use probabilistic models of protective action (countermeasure) behaviours that model the probabilities of success or failure for protective actions (countermeasures)? If you do, please describe the models and their bases.

Not included in ARANO. However, the probability of evacuation success has been modelled in a very simple way in a research project. In that model, the evacuation was a success with a given probability that depended on the distance of the municipality from the accident site. If the evacuation was considered a success, all people were considered to have been evacuated, and if a failure, no evacuation was considered to have taken place.

3.17 What information sources do you use as a technical basis for protective action (countermeasure) modelling? Do you use generic or site-specific data?

Site-specific data on population amounts in different municipalities or different locations in the map grid. Specification of the countermeasures are based on the Guides of the Radiation Safety Authority.

Radiological Exposure and Dose Assessment

3.18 How do you define potentially exposed populations or cohorts? Do you consider onsite (e.g., worker) populations or potentially sensitive, vulnerable, or critical groups? If you do, what is your basis for including these groups, and how are they modelled?

Off-site doses do not include on-site workers. Normally individual dose represents the most exposed individual of the population, who has normal living habits. Calculation parameters should be realistic or conservative if not well-known.

3.19 What exposure pathways do you model? What is your basis for selecting these pathways? What exposure duration is assumed in your models?

Cloud shine, ground shine, inhalation and ingestion (cow milk and meat, grain, green and root vegetables) pathways. These are considered to be the most significant pathways. Exposure duration varies by analysis. STUK's guides define one year exposure for the dose limits.

3.20 What information sources do you use to develop exposure and dose conversion factors? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

The normalized external dose conversion factor from the plume is calculated with a separate numerical method due to 3-dimensional dispersion model adapted. Ingestion dose conversion factors are based on the data of NRPB but applied to Nordic cultivation conditions. In some cases, age-dependent DCFs are implemented but normally adult dose factors are used. Otherwise generic dose calculation parameters are used. In the VALMA code, D.C. Kocher's coefficients for external dose are used.

Radiological Health Effects

3.21 Do you model and estimate radiological health effects in your offsite radiological consequence analyses? If you do, please respond to questions 3.22 through 3.24. If you do not, please describe any other methods you use to address the offsite radiological health effects attributable to possible accident scenarios involving nuclear installations and proceed to question 3.25.

Yes, we model and estimate radiological health effects.

3.22 What types of early and latent health effects (including fatal and non-fatal effects) do you model? What target organs do you use for the associated health effects models?

Early health effects: radiation illness and fatal cases, these are modelled by deterministic models. Latent health effects: cancers modelled by statistical model. In early health effects the target organ is bone marrow, in latent health effects effective dose.

3.23 What dose-response model(s) do you use to estimate the numbers of latent health effects attributable to radiological doses caused by accidental releases from nuclear installations, and what is the basis for their use? What (if any) methods do you use to account for uncertainty about the true dose-response relationship for exposures to low levels of ionizing radiation?

Linear no-threshold dose response is used based on generic approach; uncertainty is not considered. Usually the estimated number of cancer deaths is calculated as 0.05 times population dose (manSv).

3.24 What information sources do you use to develop input parameters for the health effect models? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

Generic average data.

Economic Consequences

3.25 Do you model and estimate economic consequences in your offsite radiological consequence analyses? If you do, please respond to questions 3.26 and 3.27. If you do not, please describe any other methods you use to address the offsite economic consequences attributable to possible accident scenarios involving nuclear installations and proceed to question 3.28.

Sometimes this is asked from the commercial utility side.

3.26 With regard to potential economic consequences considered within a Level 3 PSA:

3.26.1. What types of economic consequences do you model? What process do you use to select cost categories and models for estimating economic consequences?

Economic consequences are based on the civil defines measures which are dependent on accident severity. Such measures include e.g. evacuation and land use restrictions (lost estate, lost production). Countermeasures are based on the action levels determined by STUK. Various countermeasures are available in ARANO.

3.26.2. If you translate radiological health effects into economic consequences, please describe your process for doing so.

We have not included health effects into economic consequences.

3.26.3. If you model land contamination, please describe your process for estimating the health, environmental, or economic consequences attributed to land contamination.

Land contamination may restrict the use of the land, prevent ingestion of the foodstuffs cultivated on the contaminated surface.

The need of evacuation is assessed based on the dose during one week. The area where 50 mSv is exceeded should be evacuated and based on population data, the number of the evaluated people can be calculated. Evacuation costs x €/person and this affects costs.

Basically, the same procedure is carried out if land decontamination, relocation or food ban are considered. Only the integration time of the dose is longer e.g. 30 years and costs are based on other monetary losses.

3.27 What information sources do you use to develop cost parameters for economic consequence models? Do you use generic, region-specific, or site-specific data?

Statistics Centre of Finland can provide data, which is region-specific.

Consequence Quantification and Reporting

3.28 What process and criteria do you use for selecting consequence metrics for quantification and reporting, including spatial intervals or distances from release points?

Depends on the needs of the analysis, so it is specified case by case. The Guides of STUK define to some degree what consequences and how should be reported. There are no strict specifications for spatial intervals or distances from the release point. Results should be clearly presented.

Risk Characterization

3.29 What process and criteria do you use for selecting risk metrics for effectively communicating Level 3 PSA results, including metrics designed to measure the effects of radiological releases involving multiple units or radiological sources co-located at the same site, if applicable?

Answer is similar to the previous item. Depends on the analysis specification. Generally, population dose, dose to the most exposed individual, health effects, cancer deaths. Multiple radiological sources at the same site have not been analysed.

3.30 For each of the groups listed below, what methods do you use for presenting and communicating risk results and the uncertainty in risk results? How do you present the results

from low-probability/high-consequence events in an understandable context for each group? Who is responsible for communicating the results to each group?

(1) Decision makers:

The results of VTT's analyses are communicated by experts at the regulatory authority or NPP companies. Ordinary methods of statistics are used to present the results and the uncertainties associated with them.

We present risk metrics such as the expected number of cancer deaths and the probability that cancer deaths will result, and graphics such as Farmer's curve.

Uncertainties are presented as the results of uncertainty analyses. The main vehicle of uncertainty representation is graphics, and more precisely, uncertainty distributions of various parameters such as the probability of cancer deaths. Our analyses are usually rather simple, and we have not represented aleatory and epistemic uncertainties separately.

Examples of our way of representing risk results and uncertainties are contained in the following two reports:

http://www.vtt.fi/inf/julkaisut/muut/2014/VTT-R-05661-14.pdf
http://www.vtt.fi/inf/julkaisut/muut/2015/VTT-R-05819-15.pdf

And the following conference paper:
Tyrväinen, Tero; Karanta, Ilkka; Rossi, Jukka. Finnish experiments on level 3 PRA. 13th International Conference on Probabilistic Safety Assessment and Management, PSAM 13, 2 - 7 October 2016, Seoul, Korea. International Association for Probabilistic Safety Assessment and Management, IAPSAM (2016).

(2) General Public:

VTT is not involved in communicating analysis results to the general public.

3.31 With regard to the treatment of uncertainties:

3.31.1. Do you perform sensitivity or uncertainty analyses as part of risk estimation? If you do, what process and methods do you use?

Yes. Normally Monte Carlo simulation is used.

3.31.2. How do you determine which parameters will be varied or will have uncertainty distributions specified for propagating uncertainty? What parameters do you evaluate?

VTT has conducted studies in which weather parameters, radionuclide contents of the release, and countermeasure success probability parameters have been varied. These have been varied in the VTT code FinPSA where variation of parameters has been conducted in an event tree model applied at level 3. VTT code ARANO does not allow variation of parameters. VALMA code has the capability to do so, but computation is burdensome.

3.31.3. To what extent do you consider the effect of correlation on parameters?

Correlations between parameters have not been handled.

Other

3.32 Are there any other questions that you believe should have been asked in this survey? Is there any other information or are there any other technical challenges or notable practices you would like to share with the international community?

Questions concerning environmental consequences (metrics, uncertainty analysis etc.) would have been relevant.

Hungary: NUBIKI, HAEA, and MTA EK

PART 1: RESPONDENT INFORMATION

Name:	**Tamás Pázmándi***
Country:	**Hungary**
Organization:	**Centre for Energy Research, Hungarian Academy of Sciences**
Type of Organization:	**□ Regulatory Authority** **□ Utility** **□ Vendor** **x Academic/Research** **□ Other (please specify): _____**
Mailing Address:	**1525 Budapest 114., P.O. Box 49., Hungary**
E-mail Address:	**pazmandi.tamas@energia.mta.hu**
Telephone Number:	**+36 1 392 22 22 / 1361**

*** with contribution of**

- **Attila Bareith (NUBIKI, private company)**
- **András Siklósi (Hungarian Atomic Energy Authority)**
- **Péter Szántó (Centre for Energy Research, Hungarian Academy of Sciences, Academic)**

and reviewed by

- **Gábor Lajtha (NUBIKI, private company)**
- **Előd Holló (NUBIKI, private company)**

PART 2: APPLICATION OF LEVEL 3 PSA

2.1 Does your country require the performance of Level 3 PSAs for nuclear installations? If yes, for what types of applications are Level 3 PSAs required?

No, there is not any regulatory requirement regarding Level 3 PSA in Hungary.

2.2 With regard to the use of Level 3 PSA:

2.2.1. Do you perform, or are you considering performing Level 3 PSAs for nuclear installations? If you do, for what types of applications do you perform them?

No, but it is planned as research project in the short/medium term.

2.2.2. What calculated metrics or results from Level 3 PSAs are used in these applications and what (if any) requirements, goals, or criteria are used to evaluate these results? What is the basis for these requirements, goals, or decision criteria?

Regulatory requirements are for public dose and to avoid countermeasures although these requirements do not refer to results from a level 3 PSA.

2.2.3. If you are not currently performing or considering performing Level 3 PSAs, have you previously considered performing them? If you have, what was the basis for your decision to not perform Level 3 PSA? Please list any barriers you perceive to performing and applying Level 3 PSA in your country.

When the level 2 PSA was performed for NPP Paks in the early 2000s, a limited scope analysis was also conducted to yield insights into consequences in terms of public dose and expected exceedance frequencies for fatalities. The results were discussed internally with plant management, but the details of the analysis were not made public in lack of regulatory requirements or any other obligations.

2.2.4. What (if any) alternative methods do you use to estimate offsite public risks attributable to accidental releases of radiological materials from nuclear installations, and for what types of applications are they used?

Deterministic analysis with specified source term(s) and 2-3 meteorological situations are performed for these purposes.

2.3 One application that could potentially benefit from Level 3 PSAs is the siting of nuclear installations, including establishing the size and boundary (shape) for each emergency planning or protective action zone. Are Level 3 PSAs used to support the siting of nuclear installations in your country? If they are, how are they used?

No. However, safety regulations require that radiological consequence hazards attributable to existing nuclear installations have to be explicitly considered in siting nuclear installations. It has recently been realized that level 3 PSA would be much helpful (if not indispensable) to adequately meet that requirement.

2.4 With regard to nuclear power plant emergency planning zones:

2.4.1. What process is used to establish sizes and boundaries for emergency planning zones around a nuclear installation in which arrangements shall be made at the preparedness stage for effectively taking protective and other response actions?

Dose calculations for Design Basis Accidents.

2.4.2. What stakeholder groups are involved in the process and what are their respective responsibilities?

Individual experts were involved and public hearing was organized.

2.4.3. Under what conditions (if any) can emergency planning zones be reduced in size? Can a Level 3 PSA be used to establish a probabilistic cut off for events that need to be considered for emergency planning purposes?

No.

2.5 Another application that could potentially benefit from Level 3 PSAs is the development of safety goals or risk acceptance criteria. Are Level 3 PSAs used to support this type of application in your country? If they are, how are they used?

Not yet, but it is planned for the new NPP at Paks in the near future.

2.6 Offsite radiological consequence analyses may be performed to support applications that are not related to Level 3 PSA. Please list any additional applications supported by offsite radiological consequence analyses that you support or perform.

Deterministic analysis and dose calculations for Design Basis Accidents and Beyond-design-basis accidents.

Offsite radiological consequence analyses are performed for decision support systems for protective action recommendations.

(In the SINAC decision support system developed in MTA EK in our research group and used as an interactive expert system in the Hungarian Atomic Energy Authority Centre for Emergency Response, Training and Analysis)

PART 3: LEVEL 3 PSA MODELING ISSUES AND TECHNICAL CHALLENGES

Since we have only limited experience in Level 3 PSA and there are no ongoing activities in place at present, we could only partially answer the questions related to modelling issues below.

General

3.1 What computer code(s) do you use, or are you developing, to perform the tasks listed below?

3.1.1. Offsite radiological consequence analyses:

PC Cosyma.
Several computer codes developed in our research group or research institute.

3.1.2. Risk characterization:

None

3.2 With regard to the scope of Level 3 PSAs:

3.2.1. What internal and external accident sequence initiating events and hazards are considered in developing a Level 3 PSA?

Not applicable at present, to be specified if research on Level 3 PSA starts.

3.2.2. Do you account for correlation between causes of accident sequence initiating events and offsite phenomenological and consequence modelling? If you do, how is this correlation treated?

Not applicable at present, to be specified if research on Level 3 PSA starts.

3.3 Are you participating in any ongoing research and development activities related to offsite radiological consequence analyses? If you are, please provide a brief description of these activities.

A new guideline for the new nuclear power plant (Paks-2) is being prepared by the Authority with our participation with offsite radiological consequence analyses.

Radionuclide Release Characterization

3.4 With regard to the interface between Level 2 and Level 3 PSA:

3.4.1. How do you handle the transition from Level 2 PSA analyses to Level 3 PSA analyses? What (if any) difficulties have you encountered in this area?

Not applicable at present. Although we do not have detailed experience in this subject, our experience from performing Level 2 PSA shows that the released activity of radionuclides characterizing the dose rate can be determined from the released rate of fission product groups versus time. We calculated the released rate of fission product groups (groups: noble gases, I, Te, Cs, Ba, Mo, Sr, La, Ce, Sb, U) versus time for each representative sequence of the containment states.

Difficulty: how to specify the representative sequences?

3.4.2. If you use representative release categories or source term groups, what criteria do you use to assign radiological release sequences in the Level 2 PSA model to release categories or source term groups? How do you address potential inter-sequence variability within each representative release category or source term group?

We selected 2-6 events of each containment state (we have 13 main containment states) and calculated the release rates for representative sequences of the containment state. The selection of representative sequences was based on probability (contribution to the containment state) and also we tried to select different physical initiating events and/or containment events resulting in the same containment failure mode.

The containment states were classified according to release categories and severity. The release categories were determined according to the released fission product groups, mainly the Cs-group, the Cs-137 released radioactivity.
The released rate was calculated by the MAAP code. We also used the so-called EUR (European Utility Requirements) criteria and calculated the released activity of 9 isotopes versus time and also took into account the height of the release.

3.4.3. What process do you use to define release fraction timing (time and duration of release) and release truncation time? Do you consider onsite severe accident mitigation actions or offsite emergency response actions in this process? If you do, how are they treated?

The MAAP4/VVER severe accident code was used for the calculation of the sequences, for determining release fraction over time. The calculations were performed until the stable state, meaning that no change is expected in the released fraction. Onsite severe accident mitigation actions were taken into account in Level 2 PSA. These events were modelled by the MAAP code.

3.5 What radionuclides do you use to characterize the offsite radiological consequences attributed to accidental releases of radiological materials? What is your basis for selecting these radionuclides?

We used the Level 2 PSA results as a basis for evaluation with considerations to the following radionuclides: Xe-133, Sr-90, Cs-137, I-131, Te-131M, Ru-103, La-140, Ce-141, Ba-140. It was based on the EUR requirements referred to above under question 3.4.2.

3.6 Do you consider releases from multiple units or radiological sources co-located at the same site? If you do, how are these treated?

Not applicable at present. We consider it a very important issue. Multi-source and / or multi-unit accidents should be considered in a Level 3 PSA for multi-unit / multi-source sites to appropriately determine risk. This supposes the availability of site risk model and the associated site-level Level 2 PSA, which we have not developed yet.

3.7 Do you perform sensitivity or uncertainty analyses on release categories or release fractions? If you do, what methods are used and what is the basis for using them?

Detailed uncertainty calculations were not performed, only a limited one to explore the range of release fractions for a containment state. Severe accident calculations were performed for a selected containment state. 40 parameters of the MAAP code were selected and varied using Monte-Carlo methods (Latin Hypercube sampling) and we performed 200 calculations to get the distribution of released rates of the fission product groups.

Meteorological Data

3.8 How do you account for temporal and spatial variability in meteorological conditions? If you use a subset (sampling) of available meteorological data to serve as a representative data set, what criteria do you use for selecting these representative data?

Not applicable. Only 2-3 (the most likely) meteorological situations are used for the calculations (but not for the purpose of a Level 3 PSA).

3.9 Do you perform sensitivity or uncertainty analyses on meteorological conditions? If you do, what methods are used and what is the basis for using them?

No, only the 2-3 most likely meteorological situations are used for the calculations (but not for the purpose of a Level 3 PSA).

Environmental Transport and Dispersion

3.10 What atmospheric transport and dispersion (ATD) model(s) do you use? What process and criteria do you use to select: (1) an ATD model; and (2) a time scale for updating calculations to account for time dependence?

Gaussian plume model without time dependence.

3.11 What process and criteria do you use to define the boundaries and intervals of the spatial grid or domain used for performing offsite consequence calculations?

DBA calculations performed for the reference group.

BDBA and severe accident calculations are performed for 800 m and 3000 m from the release point (public dose for located at any point along the boundaries around the NPP).

3.12 With regard to the spatial modelling around a nuclear power plant for Level 3 PSA:

3.12.1. What information sources do you use to develop geographical or topographical parameters? Do you use generic or site-specific data?

3.12.2. Does the ATD response vary spatially with respect to varying topographic parameters? What is the spatial resolution of the ATD and do topographic parameters of the ATD vary within this resolution?

Not applicable at present.

3.13 Do you consider radiological releases to water sources and aqueous transport and dispersion phenomena? If you do, how are these treated? If you have previously considered modelling radiological releases to water sources and aqueous transport and dispersion phenomena, but decided not to do so, what was the basis for your decision?

No, presently not.

Research is ongoing on this topic; a new model and computer code are expected in a few years.

Protective Action (Countermeasure) Modelling

3.14 How do you define, and what protective actions (countermeasures) do you model for each of the nuclear or radiological incident or accident response phases listed below?

3.14.1. Early/Emergency Phase:
3.14.2. Intermediate Phase:
3.14.3. Late/Recovery Phase:

In the SINAC decision support system (developed in MTA EK in our research group and used as an interactive expert system in the Hungarian Atomic Energy Authority Centre for Emergency Response, Training and Analysis) countermeasures are taken into consideration.

Short-term countermeasures: sheltering, iodine prophylaxis and evacuation.
Long-term countermeasures: relocation, food ban and pasture ban.

3.15 Do you model population groups with different protective action (countermeasure) behaviours? If you do, how do you define them? To what extent do you account for population density or age and gender distribution effects?

No.

3.16 Do you use probabilistic models of protective action (countermeasure) behaviours that model the probabilities of success or failure for protective actions (countermeasures)? If you do, please describe the models and their bases.

No.

3.17 What information sources do you use as a technical basis for protective action (countermeasure) modelling? Do you use generic or site-specific data?

Generic data are used.

Radiological Exposure and Dose Assessment

3.18 How do you define potentially exposed populations or cohorts? Do you consider onsite (e.g., worker) populations or potentially sensitive, vulnerable, or critical groups? If you do, what is your basis for including these groups, and how are they modelled?

Critical groups are used, living closest to the release point.

3.19 What exposure pathways do you model? What is your basis for selecting these pathways? What exposure duration is assumed in your models?

Exposure pathways taken into account:

- **External doses: cloud shine and ground shine doses.**
- **Internal doses: inhalation and ingestion.**

3.20 What information sources do you use to develop exposure and dose conversion factors? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

Generic data are used, without gender-specificity, but specific conversion factors are used for children.

Radiological Health Effects

3.21 Do you model and estimate radiological health effects in your offsite radiological consequence analyses? If you do, please respond to questions 3.22 through 3.24. If you do not, please describe any other methods you use to address the offsite radiological health effects attributable to possible accident scenarios involving nuclear installations and proceed to question 3.25.

No.

(Models for calculating early and late health effects were implemented in the SINAC software many years ago, but it has never been used. These models and parameters are obsolete.)

3.22 What types of early and latent health effects (including fatal and non-fatal effects) do you model? What target organs do you use for the associated health effects models?

Not applicable.

3.23 What dose-response model(s) do you use to estimate the numbers of latent health effects attributable to radiological doses caused by accidental releases from nuclear installations, and what is the basis for their use? What (if any) methods do you use to account for uncertainty about the true dose-response relationship for exposures to low levels of ionizing radiation?

Not applicable.

3.24 What information sources do you use to develop input parameters for the health effect models? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

Not applicable.

Economic Consequences

3.25 Do you model and estimate economic consequences in your offsite radiological consequence analyses? If you do, please respond to questions 3.26 and 3.27. If you do not, please describe any other methods you use to address the offsite economic consequences attributable to possible accident scenarios involving nuclear installations and proceed to question 3.28.

No.

3.26 With regard to potential economic consequences considered within a Level 3 PSA:

3.26.1. What types of economic consequences do you model? What process do you use to select cost categories and models for estimating economic consequences?
3.26.2. If you translate radiological health effects into economic consequences, please describe your process for doing so.
3.26.3. If you model land contamination, please describe your process for estimating the health, environmental, or economic consequences attributed to land contamination.

Not applicable.

3.27 What information sources do you use to develop cost parameters for economic consequence models? Do you use generic, region-specific, or site-specific data?

Not applicable.

Consequence Quantification and Reporting

3.28 What process and criteria do you use for selecting consequence metrics for quantification and reporting, including spatial intervals or distances from release points?

DBA calculations performed for the reference group.

BDBA and severe accident calculations are performed for 800 m and 3000 m from the release point (public dose for located at any point along the boundaries around the NPP).

Risk Characterization

3.29 What process and criteria do you use for selecting risk metrics for effectively communicating Level 3 PSA results, including metrics designed to measure the effects of radiological releases involving multiple units or radiological sources co-located at the same site, if applicable?

Not applicable at present.

3.30 For each of the groups listed below, what methods do you use for presenting and communicating risk results and the uncertainty in risk results? How do you present the results from low-probability/high-consequence events in an understandable context for each group? Who is responsible for communicating the results to each group?

(1) Decision makers:
(2) General Public:

3.31 With regard to the treatment of uncertainties:

3.31.1. Do you perform sensitivity or uncertainty analyses as part of risk estimation? If you do, what process and methods do you use?

No.

3.31.2. How do you determine which parameters will be varied or will have uncertainty distributions specified for propagating uncertainty? What parameters do you evaluate?

Not applicable at present.

3.31.3. To what extent do you consider the effect of correlation on parameters?

Not applicable at present.

Other

3.32 Are there any other questions that you believe should have been asked in this survey? Is there any other information or are there any other technical challenges or notable practices you would like to share with the international community?

No.

Japan: JAEA

PART 1: RESPONDENT INFORMATION

Name:	**Yu Maruyama, Masanori Kimura, Toshimitsu Homma**
Country:	**Japan**
Organization:	**Japan Atomic Energy Agency**
Type of Organization:	**□ Regulatory Authority** **□ Utility** **□ Vendor** **x Academic/Research** **□ Other (please specify): _____**
Mailing Address:	**2-4 Shirakata, Tokai-mura, Naka-gun, Ibaraki-ken 319-1195, Japan**
E-mail Address:	**maruyama.yu@jaea.go.jp, kimura.masanori@jaea.go.jp, homma.toshimitsu@jaea.go.jp**
Telephone Number:	**+81-29-282-5486, +81-29-282-5459, +81-29-282-6862**

PART 2: APPLICATION OF LEVEL 3 PSA

2.1 Does your country require the performance of Level 3 PSAs for nuclear installations? If yes, for what types of applications are Level 3 PSAs required?

Level 3 PSA is not a regulatory requirement for licensing of nuclear power plants (NPPs).

The Nuclear Regulation Authority (NRA) of Japan is initiating discussions to implement risk assessment utilizing Level 3 PSA, as one of methodologies to assess offsite risks by nuclear facilities, in an activity for review of "self-assessment of safety improvement" by licensees.

2.2 With regard to the use of Level 3 PSA:

2.2.1. Do you perform, or are you considering performing Level 3 PSAs for nuclear installations? If you do, for what types of applications do you perform them?

Offsite consequence analysis as an element of Level 3 PSA was performed to technically support activities in the former Nuclear Safety Commission (NSC) of Japan for the discussion on performance goals of NPPs to satisfy quantitative safety goals of nuclear facilities, and activities for the revision or formulation of guidelines associated with emergency preparedness and response (EPR) in the NSC or the NRA subsequently after the accident at the Fukushima Daiichi nuclear power station. Similar analysis is currently underway, aiming mainly at providing local governments with reference technical information for planning of EPR.

2.2.2. What calculated metrics or results from Level 3 PSAs are used in these applications and what (if any) requirements, goals, or criteria are used to evaluate these results? What is the basis for these requirements, goals, or decision criteria?

In the application to the safety and performance goals, health effects of radiation exposure among the public were used as metrics, which were namely the average individual risks of early and cancer fatalities. The effectiveness of emergency protective actions has been examined by comparing calculated dose to the public with the generic criteria shown in the IAEA general safety requirements in the application to EPR.

2.2.3. If you are not currently performing or considering performing Level 3 PSAs, have you previously considered performing them? If you have, what was the basis for your decision to not perform Level 3 PSA? Please list any barriers you perceive to performing and applying Level 3 PSA in your country.

2.2.4. What (if any) alternative methods do you use to estimate offsite public risks attributable to accidental releases of radiological materials from nuclear installations, and for what types of applications are they used?

Not applicable for items 2.2.3 and 2.2.4

2.3 One application that could potentially benefit from Level 3 PSAs is the siting of nuclear installations, including establishing the size and boundary (shape) for each emergency planning or protective action zone. Are Level 3 PSAs used to support the siting of nuclear installations in your country? If they are, how are they used?

In the application of offsite consequence analysis to EPR described in the response to item 2.2.1, one of principal targets is to estimate reference distances from points of radionuclide release for

emergency planning zone (EPZ) or precautionary action zone (PAZ) and urgent protective action planning zone (UPZ).

2.4 With regard to nuclear power plant emergency planning zones:

2.4.1. What process is used to establish sizes and boundaries for emergency planning zones around a nuclear installation in which arrangements shall be made at the preparedness stage for effectively taking protective and other response actions?

In response to the experiences and lessons learned from the accident at the Fukushima Daiichi Nuclear Power Station and on the basis of international standards such as IAEA safety requirements, the NRA issued the Nuclear Emergency Response Guidelines. PAZ and UPZ are introduced and their sizes are typically specified in the guidelines in order to prevent deterministic effects and to minimize stochastic effects resulting from radiation exposure, respectively. Size of PAZ is around 5 km and that of UPZ is around 30 km in radius from NPP.

2.4.2. What stakeholder groups are involved in the process and what are their respective responsibilities?

The government, the NRA, local governments and licensees are involved in the process as stakeholder groups. The government is responsible for formulating thoroughgoing measures for EPR with overall efforts of its organizations and functions. The prime responsibility of the NRA is to issue associated guidelines with EPR (the Nuclear Emergency Response Guidelines). The local governments have responsibility to develop and implement regional EPR plans with the aid of relevant organizations and other local governments. The responsibility of licensees is to establish and maintain arrangements for onsite EPR plans and operations, and so on.

2.4.3. Under what conditions (if any) can emergency planning zones be reduced in size? Can a Level 3 PSA be used to establish a probabilistic cut off for events that need to be considered for emergency planning purposes?

Not applicable

2.5 Another application that could potentially benefit from Level 3 PSAs is the development of safety goals or risk acceptance criteria. Are Level 3 PSAs used to support this type of application in your country? If they are, how are they used?

As mentioned in the response to item 2.2.1, offsite consequence analysis was applied to the discussion on safety and performance goals. A risk metric for performance goals surrogating quantitative safety goals (the average individual risks of early and cancer fatalities), containment failure frequency (CFF), was quantified through offsite consequence analysis. Another risk metric, core damage frequency (CDF), was determined under the consideration of defence in depth.

2.6 Offsite radiological consequence analyses may be performed to support applications that are not related to Level 3 PSA. Please list any additional applications supported by offsite radiological consequence analyses that you support or perform.

Not applicable

PART 3: LEVEL 3 PSA MODELING ISSUES AND TECHNICAL CHALLENGES

General

3.1 What computer code(s) do you use, or are you developing, to perform the tasks listed below?

3.1.1. Offsite radiological consequence analyses:

Offsite radiological consequence is analysed with OSCAAR (Off-Site Consequence Analysis code for Atmospheric Release in reactor accident) code, which is composed of several modules to calculate atmospheric dispersion and deposition (ADD) of radionuclides, early and chronic doses, dose reduction by protective actions, health effects and economic impacts.

3.1.2. Risk characterization:

Not applicable

3.2 With regard to the scope of Level 3 PSAs:

3.2.1. What internal and external accident sequence initiating events and hazards are considered in developing a Level 3 PSA?

There has been no experience of offsite consequence analysis for accident scenarios initiated with external events and hazards. It is practically possible, however, to consider both internal and external events and hazards in offsite consequence analysis with OSCAAR code in a parametric way. Rational modelling could be one of important challenges for, as an example, the effectiveness of protective actions under conditions resulting from external events and hazards.

3.2.2. Do you account for correlation between causes of accident sequence initiating events and offsite phenomenological and consequence modelling? If you do, how is this correlation treated?

No, it is not accounted for.

3.3 Are you participating in any ongoing research and development activities related to offsite radiological consequence analyses? If you are, please provide a brief description of these activities.

The improvement of OSCAAR code has been made and is in progress, including modelling of protective actions such as sheltering, evacuation and iodine thyroid blocking, and ADD modelling, in which topographic effects are explicitly taken into account.

Radionuclide Release Characterization

3.4 With regard to the interface between Level 2 and Level 3 PSA:

3.4.1. How do you handle the transition from Level 2 PSA analyses to Level 3 PSA analyses? What (if any) difficulties have you encountered in this area?

See the response to item 3.4.2.

3.4.2. If you use representative release categories or source term groups, what criteria do you use to assign radiological release sequences in the Level 2 PSA model to release categories or source term groups? How do you address potential inter-sequence variability within each representative release category or source term group?

For source term analysis as an element of Level 2 PSA, THALES2 (Thermal Hydraulics and radionuclide behavior Analysis of Light water reactor to Estimate Source terms) code is being developed. The release characteristics of source term groups specified in THALES2 code are applied to input data of OSCAAR code. Close communication between analysts for source term and offsite consequence analyses is made so as to consistently deliver source term information with requirements of OSCAAR code.

3.4.3. What process do you use to define release fraction timing (time and duration of release) and release truncation time? Do you consider onsite severe accident mitigation actions or offsite emergency response actions in this process? If you do, how are they treated?

No clear definition is set for fraction timing and release truncation time. In general, although those can be evaluated through source term analysis, how they apply to offsite consequence analysis is supposed to be dependent on its targets.

In source term and offsite consequence analyses with THALES2 and OSCAAR codes, severe accident mitigation actions and offsite protective actions are taken into account as much as possible, depending on the status of modelling in the codes and objectives of analysis.

3.5 What radionuclides do you use to characterize the offsite radiological consequences attributed to accidental releases of radiological materials? What is your basis for selecting these radionuclides?

It is possible to cover more than sixty radionuclides in OSCAAR code. Important radionuclide groups include noble gases, iodine, caesium, tellurium and ruthenium, which are based on initial core inventory, half lives, volatility from fuels, mobility in NPPs, physical and chemical forms, impacts on public health and the environment, and so on.

3.6 Do you consider releases from multiple units or radiological sources co-located at the same site? If you do, how are these treated?

Site Level 1 and 2 PSAs are principally required in order to consider the releases from multi units and/or multi sources in Level 3 PSA. No activities are planned at present for site level PSAs. However, in the framework of the OECD/NEA BSAF (Benchmark Study of the Accident at Fukushima Daiichi Nuclear Power Station) project phase 2, an analysis with OSCAAR code is ongoing for the estimation of ADD of radionuclides and consequent ambient dose rate coupled with the outputs from source term analysis with THALES2 code.

3.7 Do you perform sensitivity or uncertainty analyses on release categories or release fractions? If you do, what methods are used and what is the basis for using them?

A methodology has been developed for uncertainty and sensitivity analyses of source term. The methodology is mainly divided into four steps, consisting of screening of uncertain parameters based on elementary effect model, sampling by LHS (Latin Hypercube Sampling) with the consideration of dependence among uncertain parameters, code runs for uncertainty analysis, and global sensitivity analysis using stochastic surrogate model constructed from the results of uncertainty analysis applying Bayesian nonparametric approach.

Meteorological Data

3.8 How do you account for temporal and spatial variability in meteorological conditions? If you use a subset (sampling) of available meteorological data to serve as a representative data set, what criteria do you use for selecting these representative data?

OSCAAR has a multi-puff trajectory model that can take account of changes in wind direction and variable long-duration releases. Trajectory and dispersion of each re-leased puff are calculated using two kinds of grid inputs of meteorological data. The hourly surface wind and atmospheric stability fields on the meso-scale system are constructed by using Grid Point Value (GPV) provided by the Japan Meteorological Agency. The meteorological data on the synoptic-scale system is also available from GPV data every three hours at three standard pressure levels of 925, 850 and 700 hPa. The wind data at the release point are also used at the first step of the trajectory calculations. OSCAAR can handle the spatial and temporal distribution of rainfall to predict wet deposition. OSCAAR calculations can be implemented with a variety kind of meteorological sampling schemes such as random, stratified and bin sampling schemes. We have a specific bin sampling scheme which is appropriate for the trajectory dispersion model in OSCAAR.

3.9 Do you perform sensitivity or uncertainty analyses on meteorological conditions? If you do, what methods are used and what is the basis for using them?

We have done the uncertainty and sensitivity of radiological consequences on meteorological sampling schemes for use in accident consequence assessments. A full set of weather sequences sampled from one year of hourly meteorological data was used to investigate the sensitivity of early health effects on certain meteorological conditions. A stratified sampling scheme was designed for the trajectory model in terms of initial wind direction, rainfall amount, stability category and travel time to a certain distance. The performance of the scheme developed was compared with those of random and cyclic sampling schemes. The statistical variability of the probability distribution of the consequences was also examined. (Homma, T., et al., Proc. 5th International Conference on Probabilistic Safety Assessment and Management, 2753-2758, Osaka, Nov 27-Dec 1, 2000.)

Environmental Transport and Dispersion

3.10 What atmospheric transport and dispersion (ATD) model(s) do you use? What process and criteria do you use to select: (1) an ATD model; and (2) a time scale for updating calculations to account for time dependence?

The Gaussian multi-puff trajectory model is used in OSCAAR code. A typical time step for updating calculations is one hour so as to be consistent with the time span for meso-scale meteorological data. It is possible to decrease time step down to 15 minutes typically, depending mainly on the time scale of the release of radionuclides.

3.11 What process and criteria do you use to define the boundaries and intervals of the spatial grid or domain used for performing offsite consequence calculations?

Basically, the meteorological data on the synoptic-scale system is used to cover whole area of Japan. The criteria to define the boundaries depend on the application of the accident consequence assessment. OSCAAR code uses calculation grids based on a set of annuli centred on the site and segmented by radial lines. The spatial resolution is less detailed at large distances from the site, compared with the level of detail at closer area to the site. No clear definitions are available for the boundaries and intervals of the spatial grid. Those are determined by expert judgement based on experiences to date. Finer spatial discretization is generally applied at an area closer to NPPs.

3.12 With regard to the spatial modelling around a nuclear power plant for Level 3 PSA:

3.12.1. What information sources do you use to develop geographical or topographical parameters? Do you use generic or site-specific data?

Site-specific data such as population, agricultural and livestock products and economic data are taken from domestic statistics in the Ministry of Internal Affairs and Communications, the Ministry of Agriculture, Forestry and Fisheries and local governments, and so on.

3.12.2. Does the ATD response vary spatially with respect to varying topographic parameters? What is the spatial resolution of the ATD and do topographic parameters of the ATD vary within this resolution?

Topographic effects are not considered in OSCAAR code at present. As described in the response to item 3.3, the improvement of ADD model is currently underway to take them into account.

3.13 Do you consider radiological releases to water sources and aqueous transport and dispersion phenomena? If you do, how are these treated? If you have previously considered modelling radiological releases to water sources and aqueous transport and dispersion phenomena, but decided not to do so, what was the basis for your decision?

No, they are not considered.

Protective Action (Countermeasure) Modelling

3.14 How do you define, and what protective actions (countermeasures) do you model for each of the nuclear or radiological incident or accident response phases listed below?

3.14.1. Early/Emergency Phase:

Sheltering, evacuation and iodine thyroid blocking

3.14.2. Intermediate Phase:

Relocation and restriction of ingestion of food, milk and drinking water

3.14.3. Late/Recovery Phase:

Relocation and restriction of ingestion of food, milk and drinking water

3.15 Do you model population groups with different protective action (countermeasure) behaviours? If you do, how do you define them? To what extent do you account for population density or age and gender distribution effects?

No, they are not modelled.

3.16 Do you use probabilistic models of protective action (countermeasure) behaviours that model the probabilities of success or failure for protective actions (countermeasures)? If you do, please describe the models and their bases.

No, they are not used.

3.17 What information sources do you use as a technical basis for protective action (countermeasure) modelling? Do you use generic or site-specific data?

Generic data are typically used for protective action modelling and parameters, which are based on various references including, for example, Regulatory Guide on Emergency Preparedness for Nuclear Facilities by the NSC, IAEA-TECDOC-225, NUREG/CR-4551 and OECD/NEA-EC joint report (EUR 15109) for radiation exposure reduction coefficients, and an academic paper by Jonson, et al. (Journal of Radioanalytical Chemistry. 65, 223-238, 1981) and ICRP documents for metabolism modelling for iodine intake.

Radiological Exposure and Dose Assessment

3.18 How do you define potentially exposed populations or cohorts? Do you consider onsite (e.g., worker) populations or potentially sensitive, vulnerable, or critical groups? If you do, what is your basis for including these groups, and how are they modelled?

Only members of the public are considered as potentially exposed populations.

3.19 What exposure pathways do you model? What is your basis for selecting these pathways? What exposure duration is assumed in your models?

Exposure pathway modelled are (a) external exposure from radioactive plume, (b) internal exposure due to inhalation of radionuclides in radioactive plume, (c) external exposure from radionuclide deposited on the ground, (d) internal exposure due to inhalation of radionuclides resuspended from the ground, and (e) internal exposure due to ingestion of contaminated food, milk and drinking water. The exposure duration assumed are one day, seven, 14, 21, 30, 200 and 365 days for (a) through (c), and 10, 20, 30, 40, 50 and 70 years for (c) through (e).

3.20 What information sources do you use to develop exposure and dose conversion factors? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

Generic but age- and gender-specific data listed in documents by ICRP and USEPA, and so on, are used for conversion factors.

Radiological Health Effects

3.21 Do you model and estimate radiological health effects in your offsite radiological consequence analyses? If you do, please respond to questions 3.22 through 3.24. If you do not, please describe any other methods you use to address the offsite radiological health effects attributable to possible accident scenarios involving nuclear installations and proceed to question 3.25.

Yes, radiological health effects are modelled and estimated.

3.22 What types of early and latent health effects (including fatal and non-fatal effects) do you model? What target organs do you use for the associated health effects models?

OSCAAR code considers both early and latent health effects in the population using method recommended by NUREG/CR-4214. The risk of early health effects is calculated using hazard function approach in which cumulative hazard is given by a two parameter Weibull function. The early fatal effects comprise the potentially hematopoietic, pulmonary, and gastrointestinal syndromes. The risk of latent health effects is given by linear dose response function for each cancer type. For estimating the lifetime risk in the population, the absolute or relative risk projection models are available for each cancer type such as leukemia, bone cancer, breast cancer, lung cancer, G. I. cancer, skin cancer and other cancers.

3.23 What dose-response model(s) do you use to estimate the numbers of latent health effects attributable to radiological doses caused by accidental releases from nuclear installations, and what is the basis for their use? What (if any) methods do you use to account for uncertainty about the true dose-response relationship for exposures to low levels of ionizing radiation?

Linear dose response functions based on NUREG/CR-4214 Rev.2 and EPA 402-R-99-001 are applicable. The uncertainty is not accounted for the true dose-response relationship for exposures to low levels of ionizing radiation.

3.24 What information sources do you use to develop input parameters for the health effect models? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

Generic but age- and gender-specific data listed in documents by USNRC and USEPA, JAERI-Review 2000-029 (2000), and Japanese demographic data are used for input parameters.

Economic Consequences

3.25 Do you model and estimate economic consequences in your offsite radiological consequence analyses? If you do, please respond to questions 3.26 and 3.27. If you do not, please describe any other methods you use to address the offsite economic consequences attributable to possible accident scenarios involving nuclear installations and proceed to question 3.28.

Yes, economic consequences are modelled and estimated.

3.26 With regard to potential economic consequences considered within a Level 3 PSA:

3.26.1. What types of economic consequences do you model? What process do you use to select cost categories and models for estimating economic consequences?

Modelling is made for economic consequences associated with protective actions such as evacuation and relocation of members of the public, and restriction of ingestion of food, milk and drinking water, and early deterministic and latent stochastic health effects.

3.26.2. If you translate radiological health effects into economic consequences, please describe your process for doing so.

Giving an example, calculated early and latent fatalities are multiplied by input values for income and lost duration of income to translate health effects to economic consequences.

3.26.3. If you model land contamination, please describe your process for estimating the health, environmental, or economic consequences attributed to land contamination.
The land contamination is taken into account for radiation exposure as described in the response to item 3.19 and protective action modelling such as relocation as described in the response to item 3.14, which becomes a part of source in the evaluation of health effects. The economic consequences due to the land contamination are taken into account as cost of capital services.

3.27 What information sources do you use to develop cost parameters for economic consequence models? Do you use generic, region-specific, or site-specific data?

OSCAAR code basically uses an approach to estimate the contribution to the Gross Domestic Product (GDP) except for cost of radiation-induced health effects as described in the response to item 3.26.3. Applicable data are taken from domestic statistics.

Consequence Quantification and Reporting

3.28 What process and criteria do you use for selecting consequence metrics for quantification and reporting, including spatial intervals or distances from release points?

There is no clear process or criteria for selecting consequence metrics. The analysis with OSCAAR code results in land contamination, health effects due to early and chronic radiation exposure as a function of distances from release points with the consideration of emergency protective actions, and economic impacts. These outputs are valuable and flexible to be translated into various types of metrics for offsite consequences depending on objectives.

Risk Characterization

3.29 What process and criteria do you use for selecting risk metrics for effectively communicating Level 3 PSA results, including metrics designed to measure the effects of radiological releases involving multiple units or radiological sources co-located at the same site, if applicable?

There is no clear process or criteria for selecting risk metrics. In general, appropriate risk metrics should be selected according to objectives.

3.30 For each of the groups listed below, what methods do you use for presenting and communicating risk results and the uncertainty in risk results? How do you present the results from low-probability/high-consequence events in an understandable context for each group? Who is responsible for communicating the results to each group?

(1) Decision makers:
(2) General Public:

Risk results for representative accident scenarios are typically presented by using mean or median value and 5 and 95 percentile values, considering aleatory and/or epistemic uncertainties as necessary (Homma T, et al., Nuclear Engineering and Technology, 37(3) 245 (2005), Kimura M, et al., Journal of Nuclear Science and Technology, 50(3) 296 (2013)). No specific guidelines are currently available for risk communication.

3.31 With regard to the treatment of uncertainties:

3.31.1. Do you perform sensitivity or uncertainty analyses as part of risk estimation? If you do, what process and methods do you use?

The uncertainty and sensitivity methodologies have been implemented for the OSCAAR code. The parameter uncertainty propagation analyses performed with OSCAAR code to provide quantitative information on the uncertainties of individual fatality risks using the distributions on the input parameter values obtained from a joint EC/USNRC project (Goossens, L.H.J. et al., Radiat. Prot. Dosim., 90 (2000)). The study provided the range of uncertainty for the expected values of individual risks of early and latent cancer fatalities at area close to the site. In the sensitivity analyses, the correlation/regression measures were useful for identifying those input parameters whose uncertainty makes an important contribution to the overall uncertainty for the consequence. (Homma, T, et al., Nuclear Engineering and Technology, 37(3) 245 (2005))

3.31.2. How do you determine which parameters will be varied or will have uncertainty distributions specified for propagating uncertainty? What parameters do you evaluate?

It is considered that selecting and specifying uncertainty distributions of parameters play important roles in uncertainty analysis. A large part of those have to practically depend on expert judgement based on state-of-the-art knowledge. Screening of parameters could be possible in a systematic way with elementary effect model such as Morris method (Technometrics, Vol. 33, No. 2, pp. 161-174, 1991).

3.31.3. To what extent do you consider the effect of correlation on parameters?

In the framework of uncertainty and sensitivity analyses described in the response to item 3.31.1, sampling of parameters are based on LHS with consideration of correlation between parameters. However, the quantification of correlation on parameters is a difficult task because of insufficient information and knowledge.

Other

3.32 Are there any other questions that you believe should have been asked in this survey? Is there any other information or are there any other technical challenges or notable practices you would like to share with the international community?

Korea: KAERI and KINS

PART 1: RESPONDENT INFORMATION

Name:	**Kwang-Il Ahn (KAERI*) / Taesuk Hwang (KINS)**
Country:	**Rep. of Korea**
Organization:	**KAERI / KINS**
Type of Organization:	**■ Regulatory Authority (KINS)** **□ Utility** **□ Vendor** **■ Academic/Research (KAERI)** **□ Other (please specify): _____**
Mailing Address:	**Korea Atomic Energy Research Institute (KAERI), Daedeok-Daero 989-111, Yuseong-gu, Daejeon, KOREA, ZIP: 34057**
E-mail Address:	**kiahn@kaeri.re.kr (KAERI), tshwang@kins.re.kr (KINS)**
Telephone Number:	**+82-42-868-2657 (KAERI), +82-42-868-0653 (KINS)**

(*) The corresponding organization

(**) The following members of the KAERI and KINS technical staff reviewed and supplemented this survey response:

- **Sung-yeop Kim, KAERI**
- **Dohyoung KIM, KINS**
- **Yongjin LEE, KINS**

PART 2: APPLICATION OF LEVEL 3 PSA

2.1 Does your country require the performance of Level 3 PSAs for nuclear installations? If yes, for what types of applications are Level 3 PSAs required?

In 2001, Korean Government announced the Severe Accident Policy to impose conducting PSA on all domestic nuclear power plants to identify the vulnerability against severe accident and to assess public risk.

The implementation plan for the policy, inter alia, addressed clearly that newly built NPP with APR1400 system, should conduct Level 3 PSA and demonstrate to meet safety goal. The policy defined "Level 3 PSA" as the assessment of off-site radiological consequences to evaluate public risk due to severe accident.

In 2015, the Nuclear Safety Acts were amended to accommodate the severe accident management. The Acts clearly require conducting the Level 3 PSA tied with "Accident Management Plan" to be submitted. According to the aforementioned AMP, PSA results shall satisfy the risk target values (early fatality risk (or equivalent performance goal), cancer fatality risk (or equivalent performance goal), frequency of Cs-137 release of more than 100 TBq *as below* 1.0×10^{-6}*/yr.*

2.2 With regard to the use of Level 3 PSA:

2.2.1. Do you perform, or are you considering performing Level 3 PSAs for nuclear installations? If you do, for what types of applications do you perform them?

The Korean utility, KHNP (Korea Hydro and Nuclear Power Co.), carries out the Level 3 PSA on newly built NPPs for operating license. A submitted Level 3 PSA results are reviewed by KINS (Korea Institute of Nuclear Safety, Regulatory Support Organization).

2.2.2. What calculated metrics or results from Level 3 PSAs are used in these applications and what (if any) requirements, goals, or criteria are used to evaluate these results? What is the basis for these requirements, goals, or decision criteria?

In Korea, a QHO (Quantitative Health Object) for determining the Level 3 PSA to meet the goal is as follows:

Metrics	Requirements	Goal	Basis
Early Fatality Risk	Early fatality risk from nuclear accidents shall not exceed 0.1% of the total early fatality risk arising from other accidents	5.0E-7/yr	QHO 0.1% rule
Cancer Fatality Risk	Cancer fatality risk from nuclear accidents shall not exceed 0.1% of the total cancer fatality risk	1.0E-6/yr	QHO 0.1% rule

2.2.3. If you are not currently performing or considering performing Level 3 PSAs, have you previously considered performing them? If you have, what was the basis for your decision to not perform Level 3 PSA? Please list any barriers you perceive to performing and applying Level 3 PSA in your country.

N/A

2.2.4. What (if any) alternative methods do you use to estimate offsite public risks attributable to accidental releases of radiological materials from nuclear installations, and for what types of applications are they used?

N/A

2.3 One application that could potentially benefit from Level 3 PSAs is the siting of nuclear installations, including establishing the size and boundary (shape) for each emergency planning or protective action zone. Are Level 3 PSAs used to support the siting of nuclear installations in your country? If they are, how are they used?

According to the recommendation of EPZ (emergency planning zone) established by IAEA, we have recently re-established the EPZ in terms of precautionary action zone (PAZ) as within 3 to 5 km and urgent protective action planning zone (UPZ) as within 20 to 30 km. In the process, we have assessed both PAZ and UPZ ranges through a comparison of exposure doses to emergency protection criteria which have been assessed using MACCS (MELCOR Accident Consequence Code System). RASCAL (Radiological Assessment Systems for Consequence Analysis) has been also used to verify the decision basis of PAZ and UPZ range of IAEA. In consideration of the analysis results by code and safety margin, the PAZ and UPZ compliance with the regulatory provision was established.

2.4 With regard to nuclear power plant emergency planning zones:

2.4.1. What process is used to establish sizes and boundaries for emergency planning zones around a nuclear installation in which arrangements shall be made at the preparedness stage for effectively taking protective and other response actions?

The Act on physical protection and emergency planning requires for licensee to establish emergency planning zone (EPZ) where intensive protective and other response action shall be implemented in case of abnormal release of radionuclides. The EPZ is characterized by radius of being effected preparedness of prompt emergency actions to lessen population exposure dose and specified by licensee consulting with local government taking into account the network of road and geopolitical circumstance. After approval by the regulatory authority NSSC (Nuclear Safety and Security Commission), the licensee shall reflect the approved EPZ into their emergency planning.

2.4.2. What stakeholder groups are involved in the process and what are their respective responsibilities?

Stakeholder Group	Respective Responsibilities
Division of disaster management related in Local Governments	- Consultation with licensee - Inter-local government coordination and cooperation on boundaries - Integration of review and assessment by local government and relevant agencies or organization
NSSC	- Providing legal basis for EPZ
KINS	- Providing Technical consultation on EPZ and emergency preparedness
Korea Institute of Radiological & Medical Sciences (KIRAMS)	- Providing Technical consultation on emergency medical response and Potassium Iodide (KI)
KHNP	- Providing information on EPZ

2.4.3. Under what conditions (if any) can emergency planning zones be reduced in size? Can a Level 3 PSA be used to establish a probabilistic cut off for events that need to be considered for emergency planning purposes?

In general, the radius of EPZ is estimated more conservatively than the radius derived from the Level 3 PSA. However, we have currently no experience to curtail the size of EPZ. The Level 3 PSA would be partially used if it is required to reduce the size of EPZ.

2.5 Another application that could potentially benefit from Level 3 PSAs is the development of safety goals or risk acceptance criteria. Are Level 3 PSAs used to support this type of application in your country? If they are, how are they used?

The metric of QHO (safety goal) is driven from applying a 0.1% rule to number of annual fatal casualty by all types of accident based on statistics from the National Statistical Office. The two risk acceptance criteria, CDF (Core Damage Frequency) and LERF (Large Early Release Frequency), are used as surrogate of QHO. In this step, the Level 3 PSA is partially utilized to determine the CDF and LERF. Detailed methodology using the Level 3 PSA was described in KINS/ER-190 (Performance Goals for the Korean NPPs). A summary of the report is as follow,

(1) Perform conservative calculation for early and late fatality risk at specific distance using MACCS.
(2) Evaluate the final individual risk combining recommended LERF value and pre-calculated early and late fatality risk.
(3) Verification the risk acceptance criteria through comparison between final individual risk and safety goal.

2.6 Offsite radiological consequence analyses may be performed to support applications that are not related to Level 3 PSA. Please list any additional applications supported by offsite radiological consequence analyses that you support or perform.

In Korea, offsite radiological consequence analyses have been performed to support the identification of risk impact due to the extension of surveillance test interval for ILRT (Integral Leak Rate Test) of the containment. In addition, Radiological Environment Report (RER) has been provided as a licensing document, which is based on the design basis accidents (DBAs) and normal operation procedures.

PART 3: LEVEL 3 PSA MODELING ISSUES AND TECHNICAL CHALLENGES

General

3.1 What computer code(s) do you use, or are you developing, to perform the tasks listed below?

3.1.1. Offsite radiological consequence analyses:

The MACCS (MELCOR Accident Consequence Code System) is used for the offsite radiological consequence analyses. HotSpot and ADMS are supportively used in some cases.

3.1.2. Risk characterization:

Currently, there are no specific codes for the risk characterization

3.2 With regard to the scope of Level 3 PSAs:

3.2.1. What internal and external accident sequence initiating events and hazards are considered in developing a Level 3 PSA?

The internal and external initiating events to be considered in the Level 3 PSA may be characterized by the initiating events considered to perform the Level 1 PSA. For example, a set of initiating events considered in the Level 1 PSA is shown in the below table. Then the Level 3 PSA takes into account the initiating events listed in the table.

Type	Group	Initiating Event
Internal	LOCA	Large LOCA
		Medium LOCA
		Small LOCA
		Reactor Vessel Rupture
		Steam Generator Tube Rupture
		Interfacing System LOCA
	Transient	Large Secondary Side Break
		Loss of Feed Water
		Loss of Condenser Vacuum
		Loss of a CCW System Train
		Loss of a 4.16KV Bus
		Loss of a 125V DC Bus
		Loss of Off-site Power
		Station Blackout
		General Transients
		Anticipated Transients Without Scram
External	Internal Fire	
	Internal Flood	
	Seismic	

3.2.2. Do you account for correlation between causes of accident sequence initiating events and offsite phenomenological and consequence modelling? If you do, how is this correlation treated?

We do not take into consideration of correlation between causes of accident sequence initiating events and offsite phenomenological and consequence modelling for the Level 3 PSA. Whereas, different models are being developed for seismic emergency situation considering delay time of evacuation, evacuation speed, ratio of the cohort successfully evacuated, and etc.

3.3 Are you participating in any ongoing research and development activities related to offsite radiological consequence analyses? If you are, please provide a brief description of these activities.

The research organization, KAERI, has been performing the Level 3 PSAs as a research project, whose purpose is to apply to the site risk assessment and emergency preparedness & response. In parallel with this, KAERI is currently preparing a framework to develop a domestic Level 3 PSA code which will be launched as a government-sponsored 5-year project in March 2017.

Radionuclide Release Characterization

3.4 With regard to the interface between Level 2 and Level 3 PSA:

3.4.1. How do you handle the transition from Level 2 PSA analyses to Level 3 PSA analyses? What (if any) difficulties have you encountered in this area?

We have performed Level 2 PSA severe accident analyses with MAAP (mainly in the utility side) or MELCOR code (mainly in the research and regulatory sides), and with Level 3 PSA with the MACCS code. One of essential parts in the transition process from Level 2 PSA to Level 3 PSA is to determine accident sequence-specific source terms. We have made the transition through regrouping and classification of radionuclides based on characteristics and frequencies of source term categories resulted from the Level 2 PSA.

When MELCOR is used for the Level 2 PSA, MELMACCS is preferred to be used as interfacing software. When MAAP is used for the Level 2 PSA, we use KOSCA-SOURCE (Korea Off-site Consequence Analysis – Source term module) as interfacing software. KOSCA-SOURCE has been developed by KAERI to convert the results of MAAP into the input for MACCS. Plume segmentation into 4 segments is possible.

However, there is currently an asymmetry between two codes (for Level 2 PSA and for Level 3 PSA) in terms of classification of radionuclides. For example, the MAAP code applies 12 groups of radionuclides (version 4.06) and 18 groups (version 5.0 above) while MACCS considers 9 groups. Following table provides the scheme of reclassification of radionuclides for coping with the asymmetry between MAAP 4.06 and MACCS2 for the Level 3 PSA.

Number	MACCS Source-term Group	MAAP Source-term Group
1	Noble Gas	Noble Gas
2	I	CsI, RbI
3	Cs	CsI, CsOH
4	Te	TeO_2, Te_2, Sb
5	Sr	SrO
6	Ru	RuO_2
7	Ba	BaO
8	La	La_2O_3
9	Ce	CeO

3.4.2. If you use representative release categories or source term groups, what criteria do you use to assign radiological release sequences in the Level 2 PSA model to release categories or source term groups? How do you address potential inter-sequence variability within each representative release category or source term group?

The grouping of source term is closely linked with the Level 2 PSA accident scenario and plant-specific characteristics. We have utilized 9 category binning parameters (listed in following table) to group source terms into each relevant source term release category (STC) whose number ranges from 19 to 21. Then a representative accident scenario for each source term group (derived from either frequency-base or consequence-base) is utilized to assess the characteristics of each source term (using MAAP or MELCOR).

Number	Category binning parameter	Description	Branch
1	CONBYPASS	Containment bypass	NOBYPASS ISL SGTR
2	CONISOL	Containment isolation	ISOLATED NOTISOCS NOTISOLNOCS
3	MELSTOP	In-vessel melt retention	MELSTOP RVRUPTURE CFBRB
4	TIMECF	Time of containment failure	NOCF EARLY LATE BMT
5	MODECF	Mode of containment failure	LEAK RUPTURE
6	CSS	Containment spray system	CS NOCS
7	CAVCOND	Cavity condition	DRY WET
8	SCRUB	FP scrubbing for bypass	NOT-SCRUB SCRUB
9	SG	Secondary heat removal (early)	SG NOSG

3.4.3. What process do you use to define release fraction timing (time and duration of release) and release truncation time? Do you consider onsite severe accident mitigation actions or offsite emergency response actions in this process? If you do, how are they treated?

The off-site release point is identified through the MAAP or MELCOR analysis for each source term group that is derived in the Level 2 PSA.

Then, we just use it as input parameter (PDELAY of MACCS). When performing conservative analysis, 72 hours' total release fraction is assumed to be released in 1 hour duration (1 plume segment). If more realistic analysis is needed, plume segmentation is carried out. For example, 50% of 72 hours' total release fraction is assumed to be released in 1 hour and the rest 50% of it is assumed to be released during next 9 hours

The onsite severe accident mitigation actions including cavity flooding strategy or containment depressurization strategy are taken consideration into the Level 2 CET (Containment Event Tree)/DET (Decomposition Event Tree) models or MAAP model in the process of Level 2 PSA. However, an offsite emergency response action is not considered in the domestic Level 3 PSA for conservative purpose.

3.5 What radionuclides do you use to characterize the offsite radiological consequences attributed to accidental releases of radiological materials? What is your basis for selecting these radionuclides?

60 radionuclides are taken into account to characterize the offsite radiological consequences attributed to accidental release. The maximum inventory of each radionuclide can be obtained by computer code under the conservative assumption (e.g., ORIGEN). The chosen 60 radionuclides for the analyses stems from WASH-1400 and other previous research.

3.6 Do you consider releases from multiple units or radiological sources co-located at the same site? If you do, how are these treated?

Currently, the utility side does not consider releases from multiple units. For reference, when using MACCS, KAERI has recently developed a framework to use a release fraction of single unit multiplied by the number of units.

3.7 Do you perform sensitivity or uncertainty analyses on release categories or release fractions? If you do, what methods are used and what is the basis for using them?

In the utility side, sensitivity and uncertainty analyses are not included in the scope of the analyses due to the consideration of most conservative release category or fraction.

However, diverse sensitivity analyses on plume segmentation, plume height, and heat content of plume have been carried out in the research organizations (mostly being performed in KAERI). Our experience shows that effective plume height and surface roughness are very important factors influencing the consequence.

Meteorological Data

3.8 How do you account for temporal and spatial variability in meteorological conditions? If you use a subset (sampling) of available meteorological data to serve as a representative data set, what criteria do you use for selecting these representative data?

In Korea, plant-specific and hourly-based annual meteorological data are used for the offsite consequence analysis. The meteorological data are acquired from meteorological tower located peripherally around the plant. Whereas, the MACCS code in connection with the Level 3 PSA has no capability to deal with spatial variability due to moving plume.

To select representative meteorological data set, we adopt a uniform-bin sampling method based on allocated probability of data bin in terms of wind speed, meteorological stability and degree of precipitation.

3.9 Do you perform sensitivity or uncertainty analyses on meteorological conditions? If you do, what methods are used and what is the basis for using them?

For the Level 3 PSA, we do the sensitivity study based on plant-specific and hourly-based annual meteorological data which are collected over 5 years or 10 years.

Environmental Transport and Dispersion

3.10 What atmospheric transport and dispersion (ATD) model(s) do you use? What process and criteria do you use to select: (1) an ATD model; and (2) a time scale for updating calculations to account for time dependence?

As the ATD model for Level 3 PSA, we use the Gaussian Plume model embedded in the MACCS code.

In Gaussian Plume model, however, it is noted that a time dependency is hard to consider since the trajectory of released radioactive material plume cannot be changed.

3.11 What process and criteria do you use to define the boundaries and intervals of the spatial grid or domain used for performing offsite consequence calculations?

The boundaries for off-site consequence calculation in terms of Level 3 PSA are comprised of three evaluation distance (or radius), namely EAB (Exclusive Area Boundary), early fatality (1.6km, 1mile) and latent fatality (16km, 10mile). The distance interval is set in consideration of the site area and characteristics.

According to our experience, enough number of spatial grids was found to be very important even in short range for early fatality estimation due to steep gradient of plume concentration. For the purpose of detailed segmentation, detailed data of population distribution should be supported. KOSCA-POP, which has been recently developed in KAERI, is used to develop population distribution data in the research level. Three kinds of Korean population data are installed in KOSCA-POP: area population, centre population, and point population.

3.12 With regard to the spatial modelling around a nuclear power plant for Level 3 PSA:

3.12.1. What information sources do you use to develop geographical or topographical parameters? Do you use generic or site-specific data?

The SITE module of MACCS is configured to input the data that reflects the site characteristics such as population and land fraction in each grid. Those site-specific data corresponding to each grid can be obtained from the National Statistical Office.

For reference, a surface roughness may be one of few topographical characteristics which can be applied in MACCS. As only one surface roughness length could be applied as MACCS input, it makes difficulty to decide only one value which represent complex terrain mixture.

3.12.2. Does the ATD response vary spatially with respect to varying topographic parameters? What is the spatial resolution of the ATD and do topographic parameters of the ATD vary within this resolution?

The topographic parameter (land fraction, region index, watershed index) in the MACCS code is normally assigned as single constant value to each grid. However, MACCS uses a mean value that does not vary with ATD response during the calculation.

3.13 Do you consider radiological releases to water sources and aqueous transport and dispersion phenomena? If you do, how are these treated? If you have previously considered modelling radiological releases to water sources and aqueous transport and dispersion phenomena, but decided not to do so, what was the basis for your decision?

N/A

Protective Action (Countermeasure) Modelling

3.14 How do you define, and what protective actions (countermeasures) do you model for each of the nuclear or radiological incident or accident response phases listed below?

For the Level 3 PSA, protective actions are not considered for a conservative purpose. However, we can generally take into account the followings:

3.14.1. Early/Emergency Phase:

In-house sheltering, evacuation, dose-dependent relocation, distribution of potassium iodide.

3.14.2. Intermediate Phase:

Dose-dependent relocation (based on radiation criteria).

3.14.3. Late/Recovery Phase:

Decontamination, temporal interdiction, condemnation.

3.15 Do you model population groups with different protective action (countermeasure) behaviours? If you do, how do you define them? To what extent do you account for population density or age and gender distribution effects?

In Korea, the Level 3 PSA for domestic NPPs doesn't consider credit to the emergency response (evacuation or sheltering) for the purpose of conservatism. Meanwhile, an offsite consequence analysis for extension of ILRT test interval had taken the all-out evacuation within EPZ with the assumption of single population group.

3.16 Do you use probabilistic models of protective action (countermeasure) behaviours that model the probabilities of success or failure for protective actions (countermeasures)? If you do, please describe the models and their bases.

As mentioned above, we do not take into account emergency response for doing the Level 3 PSA for domestic NPP. However, in case of offsite consequence analysis for ILRT, notwithstanding the recommendation of all-out evacuation, 5 % residual (sheltering) and 95% evacuation are assumed. Expert judgment is used to assign the probabilities of cohorts.

3.17 What information sources do you use as a technical basis for protective action (countermeasure) modelling? Do you use generic or site-specific data?

In terms of emergency response, protective actions are based on the plant-specific radiological emergency plan and its evacuation time estimation which is a legal requirement of nuclear regulation, and international guideline.

Radiological Exposure and Dose Assessment

3.18 How do you define potentially exposed populations or cohorts? Do you consider onsite (e.g., worker) populations or potentially sensitive, vulnerable, or critical groups? If you do, what is your basis for including these groups, and how are they modelled?

In Korea, the exposed populations (or cohorts) can be interpreted as the habitant who lives nearby target nuclear power plant, which is benchmarking point to evaluate ingesting pathway. Meanwhile, on-site personnel are not considered in the case of Level 3 PSA because the target audience for radiological exposure and dose assessment is only off-site resident people.

3.19 What exposure pathways do you model? What is your basis for selecting these pathways? What exposure duration is assumed in your models?

For exposure pathways, we utilize two MACCS2 models in the following table.

Exposure Duration	Exposure pathway
EARLY(~7days)	Cloudshine Groundshine Cloud Inhalation Resuspension Inhalation
LATE(~few years)	Groundshine Resuspension Inhalation Ingestion from Contaminated food & water

3.20 What information sources do you use to develop exposure and dose conversion factors? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

We apply the dose conversion factor for each radionuclides and exposure pathway embedded in the MACCS code (i.e., cloud shine, cloud inhalation, ground shine, resuspension inhalation, skin dose, and ingestion), which are recommended by ICRP without consideration of type of gender and age. For reference, KAERI has recently developed a domestic food chain model for ingestion.

Radiological Health Effects

3.21 Do you model and estimate radiological health effects in your offsite radiological consequence analyses? If you do, please respond to questions 3.22 through 3.24. If you do not, please describe any other methods you use to address the offsite radiological health effects attributable to possible accident scenarios involving nuclear installations and proceed to question 3.25.

The radiological health effects have been evaluated with the built-in model of the MACCS code.

3.22 What types of early and latent health effects (including fatal and non-fatal effects) do you model? What target organs do you use for the associated health effects models?

While the organs closely related with early and late fatality are taken into consideration for the analysis, the organs related to non-fatal effects (injury effect) are out of consideration. Following table provides the relationship between health effect and target organ associated with early and latent fatality.

Category	Health Effect	Target Organ
Early fatality	Hematopoietic Syndrome	Rad Marrow
	Pulmonary Syndrome	Lung
	Gastro-intestinal Syndrome	Lower Large Intestine
Latent fatality	Leukemia	Rad Marrow
	Bone Cancer	Bone Surface
	Breast Cancer	Breast
	Lung Cancer	Lungs
	Thyroid Cancer	Thyroid
	Gastrointestinal Cancer	Lower Large Intestine
	Other Cancers	Other

3.23 What dose-response model(s) do you use to estimate the numbers of latent health effects attributable to radiological doses caused by accidental releases from nuclear installations, and what is the basis for their use? What (if any) methods do you use to account for uncertainty about the true dose-response relationship for exposures to low levels of ionizing radiation?

For the conservatism, we use the LNT (Linear No-Threshold) model as the dose-response model. Most of international Agencies including ICRP support the LNT model for cancer and genetic disorder induced from low-dose exposure. The uncertainty analysis in regard with the application of LNT has not been made.

For reference, dose and dose-rate effectiveness factor (DDREF) employed in MACCS2, which is a multiplicative adjustment that results in lower tangent of risk for low dose, are supporting to account for uncertainty in dose-response in case of low dose level and low dose-rate.

3.24 What information sources do you use to develop input parameters for the health effect models? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

We use a generic data for health effect provided in NUREG/CR-4214, which is assigned as single value regardless of age and gender.

Economic Consequences

3.25 Do you model and estimate economic consequences in your offsite radiological consequence analyses? If you do, please respond to questions 3.26 and 3.27. If you do not, please describe any other methods you use to address the offsite economic consequences attributable to possible accident scenarios involving nuclear installations and proceed to question 3.28.

N/A

3.26 With regard to potential economic consequences considered within a Level 3 PSA:

3.26.1. What types of economic consequences do you model? What process do you use to select cost categories and models for estimating economic consequences?
3.26.2. If you translate radiological health effects into economic consequences, please describe your process for doing so.
3.26.3. If you model land contamination, please describe your process for estimating the health, environmental, or economic consequences attributed to land contamination.

N/A

3.27 What information sources do you use to develop cost parameters for economic consequence models? Do you use generic, region-specific, or site-specific data?

N/A

Consequence Quantification and Reporting

3.28 What process and criteria do you use for selecting consequence metrics for quantification and reporting, including spatial intervals or distances from release points?

The main objective of domestic Level 3 PSA is to evaluate the off-site health effect, early and late fatality risk for individual as consequence metrics and to demonstrate its compliance with the safety goal. Other metrics in MACCS code are used as supporting measures. We had already stated the criteria on spatial intervals or distances from release points at the questionnaire 3.11.

Risk Characterization

3.29 What process and criteria do you use for selecting risk metrics for effectively communicating Level 3 PSA results, including metrics designed to measure the effects of radiological releases involving multiple units or radiological sources co-located at the same site, if applicable?

Two risk metrics, early and latent fatality risk are selected for effectively communication on Level 3 PSA in accordance with the safety goal. No experience of Level 3 PSA for collocated multi-unit at the same site has been made.

3.30 For each of the groups listed below, what methods do you use for presenting and communicating risk results and the uncertainty in risk results? How do you present the results from low-probability/high-consequence events in an understandable context for each group? Who is responsible for communicating the results to each group?

A communication with decision makers and general public has been made by the regulatory authority NSSC. Related to risk results and their uncertainties, the mean (expected) values are used to compare early and late cancer fatality risk with safety goal which is set out in question 2.2.2. In the currently submitted Level 3 PSA report, only the mean values of the risk results are listed and the output of the MACCS code which is submitted as an attachment shows the distribution that occurs due to the weather changes. That is, the distribution of only random uncertainty is expressed in CCDF (Complementary Cumulative Distribution Function) form.

High-Consequence-Low-Probability events are generally screened out during performing the Level 1 PSA, on the basis of the frequency of occurrence. 'Screened out' means whether detailed analysis is performed or not. For example, detailed analysis of internal flooding event is performed where CDF exceeds 1E-09/yr for APR-1400.

(1) A communication with decision makers can be made through the risk metrics of early and late fatality risk which makes possible to explain in terms of safety goal.
(2) A communication with general public in terms of health effect has been made by the sunshine law or public hearing by the regulatory authority NSSC.

Although laws related to information disclosure exist, there are no specific guidelines for risk communication on PSA results.

3.31 With regard to the treatment of uncertainties:

3.31.1. Do you perform sensitivity or uncertainty analyses as part of risk estimation? If you do, what process and methods do you use?

We do sensitivity or uncertainty analyses as a part of risk estimation in terms of annual meteorological data and demography. The sensitivity analyses for the meteorological data are made based on 5-year data retrospectively from the timing of estimation. In case of demographic data, the sensitivity on variation demographic data is analysed with 10-year data for 50 years from the standpoint of the date of starting licensing process administratively. All analyses result should be included into submitted report to make possible tracking changes of input data.

3.31.2. How do you determine which parameters will be varied or will have uncertainty distributions specified for propagating uncertainty? What parameters do you evaluate?

For avoiding uncertainty propagation of parameters adopted for the assessment, we delineate outstanding parameters by expert engineering judgment and do the analyses corresponding to the identified outstanding parameter. In recent analysis, 9 outstanding parameters were identified including dispersion parameter; inverse layer; plume rise effect; surface roughness; deposition model; source term; dose conversion factor; health effect model; emergency response and shielding factor.

3.31.3. To what extent do you consider the effect of correlation on parameters?

N/A

Other

3.32 Are there any other questions that you believe should have been asked in this survey? Is there any other information or are there any other technical challenges or notable practices you would like to share with the international community?

N/A

Netherlands: ANVS & NRG

PART 1: RESPONDENT INFORMATION

Name:	**ANVS: Gert Jan Auwerda, Rob Jansen** **NRG: Jacques Grupa, Hans Brinkman**
Country:	**Netherlands**
Organization:	**ANVS (Regulator), NRG**
Type of Organization:	**X Regulatory Authority** **X Utility (research reactor)** **□ Vendor** **□ Academic/Research** **X Other (please specify): TSO_____**
Mailing Address:	**ANVS: Bezuidenhoutseweg 67, Postbus 16001, 2500 BA, Den Haag** **NRG: Westerduinweg 3, 1755 LE Petten, PO Box 25, 1755 ZG Petten**
E-mail Address:	**gertjan.auwerda@anvs.nl, rob.jansen@anvs.nl, grupa@nrg.eu, brinkman@nrg.eu**
Telephone Number:	**ANVS: ++31 6 11376922** **NRG: ++31 224 56 4957**

PART 2: APPLICATION OF LEVEL 3 PSA

2.1 Does your country require the performance of Level 3 PSAs for nuclear installations? If yes, for what types of applications are Level 3 PSAs required?

Yes. A level 3 PSA is required to show the nuclear installation containing fissile materials and/or ores complies to certain risk acceptance criteria when applying for a license (or a change of the license) for establishing, constructing, commissioning, operating or decommissioning of a nuclear installation.

2.2 With regard to the use of Level 3 PSA:

2.2.1. Do you perform, or are you considering performing Level 3 PSAs for nuclear installations? If you do, for what types of applications do you perform them?

Power reactor, research reactors, hot cell facilities, waste treatment and decommissioning facilities, waste storage facilities, enrichment facility

2.2.2. What calculated metrics or results from Level 3 PSAs are used in these applications and what (if any) requirements, goals, or criteria are used to evaluate these results? What is the basis for these requirements, goals, or decision criteria?

Risk criteria are explicitly included in the Nuclear Installations, Fissionable Materials and Ores Decree (Bkse) as assessment principles for licences to be granted to nuclear reactors. The outcomes of a level-3 PSA must be compared with these risk criteria and objectives. The requirements are formulated as follows:

a) That the probability that an individual, whom resides permanently and unprotected outside the facility grounds, dies as a result of a beyond-design accident is less than 10^{-6} per year. To calculate this 'individual risk', the habits, location, age, etc. of this individual are chosen in such a way as to result in the worst-case scenario that would still be reasonable, but does not have to be a real person. In practice this is a 1-year old infant that would live next to the edge of the installation. In determining the total dose resulting from an accident, one has to take into account long-term (stochastic) effects for at least 50 years, including ongoing exposition over that time-period due to contamination of the environment.
b) That the probability that a beyond-design accident leads to at least 10 direct fatalities (within a few weeks, due to deterministic effects) is below 10^{-5} per year, and the probability on n times more fatalities is n^2 times smaller. (e.g., 100 fatalities, probability < 10^{-7} per year; 1000 fatalities, probability <10^{-9} per year; etc.). For the calculation of this 'group-risk' actual distribution of the population is to be taken into account.

In demonstrating compliance with the risk criteria, it has to be assumed that only the usual forms of mitigative action (i.e. fire brigades, hospitals, etc.) are taken. Evacuation, iodine prophylaxis, sheltering or decontamination of the environment may not, therefore, be included in the calculation.

2.2.3. If you are not currently performing or considering performing Level 3 PSAs, have you previously considered performing them? If you have, what was the basis for your decision to not perform Level 3 PSA? Please list any barriers you perceive to performing and applying Level 3 PSA in your country.

2.2.4. What (if any) alternative methods do you use to estimate offsite public risks attributable to accidental releases of radiological materials from nuclear installations, and for what types of applications are they used?

For "design basis accidents", licensees have to show, using a deterministic approach, that for representative postulated events the dose for an individual, whom resides permanently and unprotected outside the facility grounds stays below certain threshold values, depending on the expected frequency of the event. Frequency – dose limit values are: F≥10-1 per year à dose ≤0.1 mSv; 10-1>F≥10-2 per year à dose ≤1 mSv; 10-2>F≥10-4 per year à dose ≤10 mSv; F<10-4 per year à dose ≤100 mSv. For individuals younger than 16 years, the dose limit is multiplied by 0.4. Additionally, the effective thyroid-dose has to be ≤500 mSv.

The NPP Borssele uses a 'Living PSA', which include a spread sheet tool with strategic level-3 results, to easily recalculate the offsite risks when accident frequencies change.

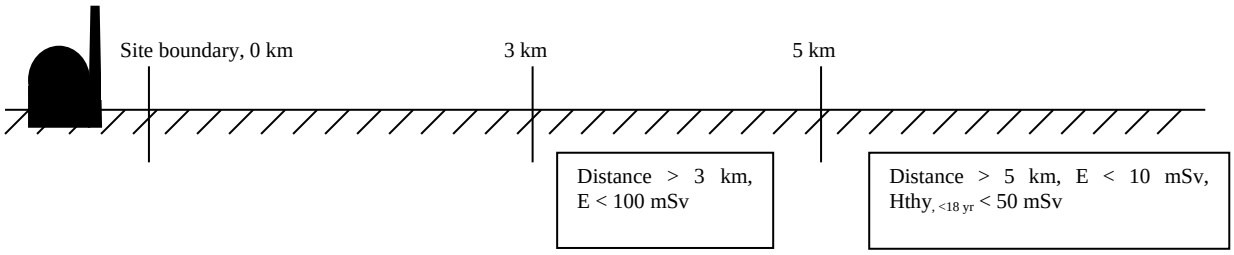
2.3 One application that could potentially benefit from Level 3 PSAs is the siting of nuclear installations, including establishing the size and boundary (shape) for each emergency planning or protective action zone. Are Level 3 PSAs used to support the siting of nuclear installations in your country? If they are, how are they used?

In practice, the 'group risk' mentioned under 2.2.2 takes this role. Additionally, for new build reactors, the newly developed Dutch Safety Requirements (DSR) set some additional goals. The following table contains the design requirements for a core-melt accident that cannot be practically excluded:

Protective action	Evacuation zone (< 3 km)	Sheltering zone (< 5 km)	Beyond sheltering zone
Permanent evacuation	No	No	No
Evacuation	May be needed	No	No
Sheltering	May be needed	May be needed	No
Iodine prophylaxis	May be needed	May be needed	No

The zones serve as design requirements in combination with the Dutch intervention levels. In that context, the following intervention levels apply: for sheltering, the intervention level is an effective dose of E ≥ 10 mSv; for evacuation, the intervention level is an effective dose of E ≥ 100 mSv and for the distribution of iodine prophylaxis, the intervention level for children is a thyroid dose of Hthy, <18 yr ≥ 50 mSv and the intervention level for adults is a thyroid dose of Hthy, ≥18 yr ≥ 100 mSv.

Figure 2: Zones and intervention levels for postulated core-melt accidents.



The zones specified in the requirements are intended for use during the design of a new reactor, not in emergency response. The National Plan for Nuclear and Radiological Emergencies ("Nationaal Crisisplan Stralingsincidenten") and the regional emergency response plans specify the action to be taken in the event of an accident and the associated preparation zones. The requirements apply to the design. Hence, an applicant shall demonstrate that, in the event of any postulated core-melt accident, the requirements will be met. The analyses shall of course take account of the local weather conditions.

2.4 With regard to nuclear power plant emergency planning zones:

2.4.1. What process is used to establish sizes and boundaries for emergency planning zones around a nuclear installation in which arrangements shall be made at the preparedness stage for effectively taking protective and other response actions?

A number of DBA's are selected, and it is calculated to which distance dose thresholds for various measures (shielding, immediate evacuation, evacuation or relocation, food control measures) can be exceeded with a (weather statistics dependent) probability of more than about 5%.

2.4.2. What stakeholder groups are involved in the process and what are their respective responsibilities?

Plant owner, plant operator, local community Mayor, regional emergency organisation (translated from 'veiligheidsregio'), national response organisation.

2.4.3. Under what conditions (if any) can emergency planning zones be reduced in size? Can a Level 3 PSA be used to establish a probabilistic cut off for events that need to be considered for emergency planning purposes?

The emergency response is an independent level of defence, and reducing this motivated by considering other levels of defence (plant safety systems as in a PSA) could compromise the 'defence in depth' approach. The motivation of reducing emergency planning zones should not only rely on a PSA. Ideally, it should be a risk informed approach.

2.5 Another application that could potentially benefit from Level 3 PSAs is the development of safety goals or risk acceptance criteria. Are Level 3 PSAs used to support this type of application in your country? If they are, how are they used?

In NL, the PSA (nuclear) and QRA (for conventional plant: coal / gas fired, chemical, oil & gas industry; QRA is comparable to PSA) is used to determine the size of the unmitigated risks (nuclear and conventional), and this is used to show compliance with the legal limit, mentioned under 2.2.2. These limits are safety goals / risk acceptance criteria.

Additionally, it is required to meet the safety goals/ risk acceptance criteria mentioned under 2.2.4. There is a distinction between these deterministic analyses of design base accidents, where it is shown that each safety system is capable of mitigating the risk of a stylized, but foreseeable accidents (i.e. a safety goal), and PSA, where the remaining (unmitigated) risk is calculated (i.e. risk acceptance). In showing compliance to these requirements, calculation methods very similar to those used for the Level 3 PSA are used.

2.6 Offsite radiological consequence analyses may be performed to support applications that are not related to Level 3 PSA. Please list any additional applications supported by offsite radiological consequence analyses that you support or perform.

As mentioned under 2.2.4 and 2.5, similar calculation methods as for level-3 PSA are used to show compliance to the legal limits (safety goals) for design base accidents.

Additionally, PSA-like methods are used for emergency planning and by the licensees for insurance applications.

PART 3: LEVEL 3 PSA MODELING ISSUES AND TECHNICAL CHALLENGES

General

3.1 What computer code(s) do you use, or are you developing, to perform the tasks listed below?

3.1.1. Offsite radiological consequence analyses:

License holders use COSYMA as well as codes based on the Dutch New National Model (NNM) such as NUDOS2. MACCS2 is rarely used.

3.1.2. Risk characterization:

Spread sheet post processing, GIS-(contour)-mapping

3.2 With regard to the scope of Level 3 PSAs:

3.2.1. What internal and external accident sequence initiating events and hazards are considered in developing a Level 3 PSA?

This is part of the level-1 and -2. Starting point are the international IAEA lists completed with site specific PIE. The Level 3 PSA carried out for NPP is full scope and includes all plant states, both core and spent fuel, internal and external events, internal and external hazards, human errors, common cause errors, etc.

3.2.2. Do you account for correlation between causes of accident sequence initiating events and offsite phenomenological and consequence modelling? If you do, how is this correlation treated?

To take into account these correlations is only required if otherwise results would not be conservative. So a conservative approach is taken to account for such correlations. However, we have no systematic process to evaluate this. For situations where we explicitly know that the risk dominating accident sequence is initiated by an event that correlates to one or more PSA-3 stochastic parameters, ignoring this correlation is - in our experience - in practice conservative. E.g. for the PIE 'impact of a military airplane during an exercise', we know that this only takes place during daytime. That rules out most (stable) mixing layer conditions (occurring at night). Including such correlations in the calculations is perfectly allowable. Similar, high wind speeds can cause damage, but at the same time give a high dilution factor in air.

3.3 Are you participating in any ongoing research and development activities related to offsite radiological consequence analyses? If you are, please provide a brief description of these activities.

We are in the final stages of the development of a new guidance document for performing PSA-3 calculations. Part of the development of this new guide included studies related to the application of the dispersion models in COSYMA versus the dispersion models in the 'NNM' model (Nieuw Nederlands Model = New Dutch Model) and studies related to the applicability of the NNM for emissions related to accidents, as the NNM was originally developed for continuous emissions of hazardous substances (not necessarily related to nuclear activities).

Radionuclide Release Characterization

3.4 With regard to the interface between Level 2 and Level 3 PSA:

3.4.1. How do you handle the transition from Level 2 PSA analyses to Level 3 PSA analyses?

3.4.2. What (if any) difficulties have you encountered in this area?

Level-2 provides masses of species that are released as a function of time, and the masses of the chemical elements in the core. Special attention must be given to chemical elements in a species that is altered during its migration through the plants systems (e.g. iodine conversion in water), chemical elements that occur in more than one species (I2, organic-I, CsI, CsOH, Cs2MoO4, other Mo-species), and radioactive daughter products that grow during the migration through the plant

3.4.3. If you use representative release categories or source term groups, what criteria do you use to assign radiological release sequences in the Level 2 PSA model to release categories or source term groups? How do you address potential inter-sequence variability within each representative release category or source term group?

That is based on an expert judgment by the Level-2 specialists, that pick out a representative accident for each source term group. The practical limitation is in the Level-2: each representative accident can take up to some weeks of computing time in the Level-2 to calculate the release data vs. time. (In level-3 this is a few hours per source term at max)

3.4.4. What process do you use to define release fraction timing (time and duration of release) and release truncation time? Do you consider onsite severe accident mitigation actions or offsite emergency response actions in this process? If you do, how are they treated?

We carefully study the release fraction timing by visual inspection of the graphs. Early fatality risk depends on this timing, but until now we observed that accident releases that cause early fatality risks take only a few hours, so it can be covered by a short sequence of hourly releases. Long lasting releases relate to partial functioning of one of the barriers, e.g. a containment. In those cases the release is smaller, and thresholds for early health effects are not exceeded, in our experience. The stochastic dose-risk is slightly conservatively modelled as linear, no threshold and no DDREF. In that case, it can be mathematically shown that the release can be assumed to occur in one hour without getting wrong results, as long as the stochastic risk is smaller than 1. (The core ingredient of the mathematical proof is that the risk is linear with the dose.)

Onsite severe accident mitigation actions should be considered in the PSA-2 to calculate the source term. For offsite mitigating actions it has to be assumed that only the usual forms of mitigative action (i.e. fire brigades, hospital care, decontamination of persons, etc.) are taken. Evacuation, iodine prophylaxis, sheltering may not be included in the calculation. Indirect mitigating actions such as restrictions on the food-chain should be considered.

3.5 What radionuclides do you use to characterize the offsite radiological consequences attributed to accidental releases of radiological materials? What is your basis for selecting these radionuclides?

The present database contains more than 800 nuclides. To our knowledge this covers all nuclides that may be relevant for radiological consequences. The selection of the 800 nuclides is in the end an expert judgment. In the 1990s, the database was limited to about 50 nuclides, also selected by expert judgment. We have observed that the 750 nuclides that have been added since then do not contribute significantly (except in some very exotic accidents).

3.6 Do you consider releases from multiple units or radiological sources co-located at the same site? If you do, how are these treated?

Sometimes this is required, it depends on the license type. If the initiating event or hazard results in multiple releases, these should be considered together. In that case the risks are summed by post-processing, most often in a spread sheet.

3.7 Do you perform sensitivity or uncertainty analyses on release categories or release fractions? If you do, what methods are used and what is the basis for using them?

No

Meteorological Data

3.8 How do you account for temporal and spatial variability in meteorological conditions? If you use a subset (sampling) of available meteorological data to serve as a representative data set, what criteria do you use for selecting these representative data?

Only temporal variability. The risk metrics only need short distance calculations, and some medium distance averages. These results are not sensitive to the spatial variability in the Netherlands.
Hour by hour meteorological data sequences for the whole year are available.

3.9 Do you perform sensitivity or uncertainty analyses on meteorological conditions? If you do, what methods are used and what is the basis for using them?

A large enough sampling of weather sequences is used. This is code specific. For NUDOS2 the sample size is 8000 to 80000 weather sequences. Straightforward statistics are sufficient.

Environmental Transport and Dispersion

3.10 What atmospheric transport and dispersion (ATD) model(s) do you use? What process and criteria do you use to select: (1) an ATD model; and (2) a time scale for updating calculations to account for time dependence?

This is code specific. NUDOS2 uses one hour averaged Gaussian plume, aerosol size and friction velocity dependent deposition, aerosol size and rain rate dependent wet deposition (standard Dutch air pollution model NNM or SRM3).

3.11 What process and criteria do you use to define the boundaries and intervals of the spatial grid or domain used for performing offsite consequence calculations?

This is code specific. NUDOS2 uses a polar grid of five kilometre for the location dependent result (max risk, max dose), up to 25 km for regional average food contamination estimates. There is no hard limit on the number of result grid point (normally we use about 10000 grid points) in the code. After a calculation, we do a visual inspection of the results to see if the grid was sufficiently dense. Moreover, the code uses an internal Cartesian grid that adapts automatically to the plume shape (explicit method). At the end of each weather sequence, the result on the Cartesian grid is converted to the result output grid (polar).

3.12 With regard to the spatial modelling around a nuclear power plant for Level 3 PSA:

3.12.1. What information sources do you use to develop geographical or topographical parameters? Do you use generic or site-specific data?

Site specific data up to five km, combined with generic data for the Netherlands, which is adequate given the size and flat topography of the Netherlands.

3.12.2. Does the ATD response vary spatially with respect to varying topographic parameters? What is the spatial resolution of the ATD and do topographic parameters of the ATD vary within this resolution?

No

3.13 Do you consider radiological releases to water sources and aqueous transport and dispersion phenomena? If you do, how are these treated? If you have previously considered modelling radiological releases to water sources and aqueous transport and dispersion phenomena, but decided not to do so, what was the basis for your decision?

In principle, yes, should aqueous transport result in significant doses to the public. However, usually aqueous transport and dispersion can be screened out for the required risk metrics if a plant has the potential of causing airborne releases.

Protective Action (Countermeasure) Modelling

3.14 How do you define, and what protective actions (countermeasures) do you model for each of the nuclear or radiological incident or accident response phases listed below?

3.14.1. Early/Emergency Phase:
3.14.2. Intermediate Phase:
3.14.3. Late/Recovery Phase:

Direct offsite protective actions such as evacuation, sheltering and iodine prophylaxis are not considered in the PSA-3. Decontamination of people (after 24 hours) and indirect actions such as food control measures, flushing of contaminated water and protection of drink water areas are allowed to be taken into account. Onsite protective actions should be part of the PSA-2 modelling (SAMGs).

3.15 Do you model population groups with different protective action (countermeasure) behaviours? If you do, how do you define them? To what extent do you account for population density or age and gender distribution effects?

As mentioned above, direct offsite protective actions are not to be included in the risk calculation. The risk should be calculated for the most vulnerable group that is still identifiable as a group. In practice, this is almost always the group of 1-year old infants.

3.16 Do you use probabilistic models of protective action (countermeasure) behaviours that model the probabilities of success or failure for protective actions (countermeasures)? If you do, please describe the models and their bases.

No

3.17 What information sources do you use as a technical basis for protective action (countermeasure) modelling? Do you use generic or site-specific data?

As mentioned above, no credit is taken for direct offsite protective actions. As such, this type of data is not needed for the risk metrics used in the PSA. Decontamination is assumed to take place 24 hours after exposure and food control measures are assumed to be taken should food contamination be above the legal limits.

Radiological Exposure and Dose Assessment

3.18 How do you define potentially exposed populations or cohorts? Do you consider onsite (e.g., worker) populations or potentially sensitive, vulnerable, or critical groups? If you do, what is your basis for including these groups, and how are they modelled?

Onsite exposures are not considered. Only individuals outside the perimeter of the nuclear installation are considered. For the individual risk, critical groups are considered. The legal risk level must be met for each member of a hypothetical group of people. Guidance states that the most critical group that can still be considered a group has to be considered. In practice this is 1-year old infants.

For the group risk (CCDF of the number of early fatalities) real data for the number of inhabitants, workers (in nearby companies) and other volatile population has to be included. this implies for example that at night the population number is different from the numbers during day time.

3.19 What exposure pathways do you model? What is your basis for selecting these pathways? What exposure duration is assumed in your models?

All exposure pathways that are relevant must be included. This includes the 'usual' pathways cloud-dose, ground-dose, inhalation dose, ingestion dose, resuspension dose. For research reactors, we found typically that 'external dose' from the contamination inside the reactor building is dominating the dose rate at the reactor site boundary, since the site of a research reactor is usually relatively small (e.g. a university or a research site with more than one research 'companies'.

The external radiation requires a separate calculation with other tools, such as e.g. micro Shield or MCNP.

The exposure duration is 50 to 70 (for children) years.

3.20 What information sources do you use to develop exposure and dose conversion factors? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

All site independent data are prescribed in the legal documents; the data are age specific. Site dependent data have to be applied. Some site data are difficult to assess and generic values are provided, at the cost of increasing the conservatism. For example, we use a generic aerosol size distribution since aerosol size distributions for the releases are not provided by the level-2 PSA.

Radiological Health Effects

3.21 Do you model and estimate radiological health effects in your offsite radiological consequence analyses? If you do, please respond to questions 3.22 through 3.24. If you do not, please describe any other methods you use to address the offsite radiological health effects attributable to possible accident scenarios involving nuclear installations and proceed to question 3.25.

Yes

3.22 What types of early and latent health effects (including fatal and non-fatal effects) do you model? What target organs do you use for the associated health effects models?

For early health effects, the dose to the red bone marrow, lungs, GI-tract and skin is assessed. For stochastic effects the ICRP effective dose is calculated and an age-dependent (linear) risk factor is used.

3.23 What dose-response model(s) do you use to estimate the numbers of latent health effects attributable to radiological doses caused by accidental releases from nuclear installations, and what is the basis for their use? What (if any) methods do you use to account for uncertainty about the true dose-response relationship for exposures to low levels of ionizing radiation?

$$r = (1 - e^{-\ln(2)\left(D/LD_{50}\right)^{v}}$$

LD50 are taken from ICRP, threshold have been reported by ICRP and UNSCEAR. The shape factor v is calculated using the LD50 and the T, such that r=1% if the dose is equal to the threshold in the above hazard function.

For latent health effects, for the probability of mortality a risk factor of 5% per Sv is assumed for adults and 15% per Sv for children, based on ICRP-103 and ICRP-60 as well as UNSCEAR 2012.

3.24 What information sources do you use to develop input parameters for the health effect models? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

We have a guideline that provides in acceptable parameter values. The values are based on ICRP and UNSCEAR publications. For the early health effects, averaged data are provided; for the stochastic health effects, age-dependent data are provided

Economic Consequences

3.25 Do you model and estimate economic consequences in your offsite radiological consequence analyses? If you do, please respond to questions 3.26 and 3.27. If you do not, please describe any other methods you use to address the offsite economic consequences attributable to possible accident scenarios involving nuclear installations and proceed to question 3.28.

Economic consequences are not assessed by default in the PSA-3. For selected source terms, assessments have been made on request of the plant owner. Furthermore, the 'group risk' (see 2.2.2) can be considered as a measure of the societal impact of a mayor accident.

3.26 With regard to potential economic consequences considered within a Level 3 PSA:

3.26.1. What types of economic consequences do you model? What process do you use to select cost categories and models for estimating economic consequences?

To estimate the costs, areas with high contamination are distinguished, and the costs of temporarily abandoning (e.g. if a nearby plant is abandoned it costs 'n' million euros per day) this area and cleaning it are calculated by very rough estimates.

3.26.2. If you translate radiological health effects into economic consequences, please describe your process for doing so.

We have no experience, for the source terms we considered, the major costs where in the temporarily closing of nearby plants.

3.26.3. If you model land contamination, please describe your process for estimating the health, environmental, or economic consequences attributed to land contamination.

The costs are based on yearly turnover data and other rough estimates.

3.27 What information sources do you use to develop cost parameters for economic consequence models? Do you use generic, region-specific, or site-specific data?

Regional maps, annual reports of companies and generic data from literature.

Consequence Quantification and Reporting

3.28 What process and criteria do you use for selecting consequence metrics for quantification and reporting, including spatial intervals or distances from release points?

There are well defined, legal endpoints. In addition, licensees provide contour plots on geographic maps, and statistics of distances up unto which effects of interest can occur.

Risk Characterization

3.29 What process and criteria do you use for selecting risk metrics for effectively communicating Level 3 PSA results, including metrics designed to measure the effects of radiological releases involving multiple units or radiological sources co-located at the same site, if applicable?

PSA-3 is used to show compliance to legal limits. There have been attempts to develop additional metrics, but it is difficult to find natural reference values, acceptable probability levels, to be able to frame a new metric.

3.30 For each of the groups listed below, what methods do you use for presenting and communicating risk results and the uncertainty in risk results? How do you present the results from low-probability/high-consequence events in an understandable context for each group? Who is responsible for communicating the results to each group?

(1) Decision makers:
(2) General Public:

The individual risk is a single number. However, often a graph is included to show the individual risk as a factor of the distance from the facility.

The group risk is shown as a graph with on the Y-axis the probability and on the X-axis the number of direct casualties (log-log scale). A line representing the requirement (prob $<10^{-5}$ per year for 10 direct casualties, and n2 times lower probability for n times more casualties) is included as a reference.

The licensee/applicant is responsible for communicating the results to the decision makers (the regulatory body). The decision makers (ANVS) is responsible for communicating their decisions to the general public, based on the information of the licensee/applicant. However, communication to the public is also done by the licensee/applicant as part of the environmental impact assessment and the license application, as this information is made available to the public to facilitate public participation.

In The Netherlands, all hazardous (industrial) activities, including non-nuclear activities, have to comply with the legal risk criteria as part of the licensing process. Although for each site the risks are addressed separately by the site owner in a safety report, the authorities have compiled all risk information from the safety reports in the Dutch risk map ("Risicokaart"), which is publicly accessible online. For example, a map of the risk levels from all hazardous activities (nuclear and non-nuclear) within 25 km from the nuclear research site in Petten can be found at: http://www.risicokaart.nl/en/voorvertoning?rd=98753.33-526424.58-118426.73-539499.8&zoom=10

This figure is provided below as an example display.

3.31 With regard to the treatment of uncertainties:

3.31.1. Do you perform sensitivity or uncertainty analyses as part of risk estimation? If you do, what process and methods do you use?

We provide a qualitative estimate of the accuracy of the provided risk numbers. Based on the EC-US NRC uncertainty study [1993 - 2000] we claim that the accuracy is about one order of magnitude.

We would be interested to acquire (qualitative) uncertainty estimates for the full PSA, not the consequence assessment on its own.

3.31.2. How do you determine which parameters will be varied or will have uncertainty distributions specified for propagating uncertainty? What parameters do you evaluate?

Not

3.31.3. To what extent do you consider the effect of correlation on parameters?

Not

Other

3.32 Are there any other questions that you believe should have been asked in this survey? Is there any other information or are there any other technical challenges or notable practices you would like to share with the international community?

We are most interested in:
- **an overview of international attention for level-3**
- **an international cooperation to calibrate our PSA-3 skills**
- **(new) risk metrics and communication**
- **the qualitative overall accuracy of the full PSA results**

Sweden: SSM

PART 1: RESPONDENT INFORMATION

Name:	**Per Hellström**
Country:	**Sweden**
Organization:	**SSM**
Type of Organization:	**x Regulatory Authority** **□ Utility** **□ Vendor** **□ Academic/Research** **□ Other (please specify): _____**
Mailing Address:	**SE-171 16 Stockholm** **Solna strandväg 96**
E-mail Address:	**per.hellstrom@ssm.se**
Telephone Number:	**+46 8 799 4038**

PART 2: APPLICATION OF LEVEL 3 PSA

2.1 Does your country require the performance of Level 3 PSAs for nuclear installations? If yes, for what types of applications are Level 3 PSAs required?

Level 3 PSA is not a regulatory requirement. PSA level 2 is required. The safety analysis reports by the utilities also have sections on radiological consequences in case of accidents (incidents) presenting standard radiological consequence analyses for a set of predefined cases. There are ongoing discussions on potential level 3 requirement to be implemented in new regulations currently being developed. One reason for the new regulations project was to consider potentially new NPPs, where it might have been justified to include level 3 requirements.

2.2 With regard to the use of Level 3 PSA:

2.2.1. Do you perform, or are you considering performing Level 3 PSAs for nuclear installations? If you do, for what types of applications do you perform them?

This discussion is ongoing, see text above. One specific issue is what the objective should be in case of a level 3 PSA requirement, what should be the added value depending on how such a requirement would be formulated. Questions that are being addressed are "When and how can results from the level 3 PSAs be used and provide additional insights compared with PSA level 1 and 2?"

2.2.2. What calculated metrics or results from Level 3 PSAs are used in these applications and what (if any) requirements, goals, or criteria are used to evaluate these results? What is the basis for these requirements, goals, or decision criteria?

See previous response.

2.2.3. If you are not currently performing or considering performing Level 3 PSAs, have you previously considered performing them? If you have, what was the basis for your decision to not perform Level 3 PSA? Please list any barriers you perceive to performing and applying Level 3 PSA in your country.

In Sweden, there was in the beginning of the 70ties a study on urban siting of NPPs and a comparison between different power production options. Methods used were similar to those used in the US at that time. One conclusion was that it is preferred to place NPPs outside highly populated areas.

2.2.4. What (if any) alternative methods do you use to estimate offsite public risks attributable to accidental releases of radiological materials from nuclear installations, and for what types of applications are they used?

As mentioned above, off site radiological consequence analyses are performed for a set of "accidents" and reported in the safety analysis reports. It is necessary to show that the consequences are below certain limits. One such analysis is for a design release from the scrubber (filtered vented containment) in case of a severe accident. Results are expressed in terms of doses.

2.3 One application that could potentially benefit from Level 3 PSAs is the siting of nuclear installations, including establishing the size and boundary (shape) for each emergency

planning or protective action zone. Are Level 3 PSAs used to support the siting of nuclear installations in your country? If they are, how are they used?

See earlier response with some information on the urban siting study from the early 70ties.

2.4 With regard to nuclear power plant emergency planning zones:

2.4.1. What process is used to establish sizes and boundaries for emergency planning zones around a nuclear installation in which arrangements shall be made at the preparedness stage for effectively taking protective and other response actions?

Existing zones are currently being reevaluated. More information can be provided later

2.4.2. What stakeholder groups are involved in the process and what are their respective responsibilities?

A number of authorities, e.g. local, SSM, MSB (Engelska), more information can be provided later.

2.4.3. Under what conditions (if any) can emergency planning zones be reduced in size? Can a Level 3 PSA be used to establish a probabilistic cut off for events that need to be considered for emergency planning purposes?

No answer

2.5 Another application that could potentially benefit from Level 3 PSAs is the development of safety goals or risk acceptance criteria. Are Level 3 PSAs used to support this type of application in your country? If they are, how are they used?

No.

2.6 Offsite radiological consequence analyses may be performed to support applications that are not related to Level 3 PSA. Please list any additional applications supported by offsite radiological consequence analyses that you support or perform.

Offsite radiological consequence analyses are performed to show that events, conditions and scenarios with certain frequencies meet the related dose criteria. These criteria are the "usual" criteria for the design of the NPP SSCs with regard to performance and environmental qualifications.

Certain information on this is collected in the L3 PSA project and in the final report for 2015 text below from this report that can be made available when completed)

Table 1 Reference values in Sweden

Frequency range of accident	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 decision further gives guidance on standardized calculations; it outlines standardized calculation 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). Internal doses from inhalation and effective doses should be shown not to exceed the reference values, without any countermeasures or normal living factors included.

The decision also outlines considerations for even more realistic calculations, so called adjusted calculations. Here the release is not restricted to one hour and countermeasures considerations are included in the assessment.

In the same decision, it is stated that US NRC RG 1.183 [35] should be followed altogether in conservative consequence analysis, except for dose conversion factors and 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.

PART 3: LEVEL 3 PSA MODELING ISSUES AND TECHNICAL CHALLENGES

General

3.1 What computer code(s) do you use, or are you developing, to perform the tasks listed below?

3.1.1. Offsite radiological consequence 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 since it does not require significant input or overhead.

There is also ongoing work with a tool for RApid Source Term Pedicion (RASTEP). This is financed by SSM.

3.1.2. Risk characterization:

The consequences are not combined with frequencies

3.2 With regard to the scope of Level 3 PSAs:

3.2.1. What internal and external accident sequence initiating events and hazards are considered in developing a Level 3 PSA?

Not applicable

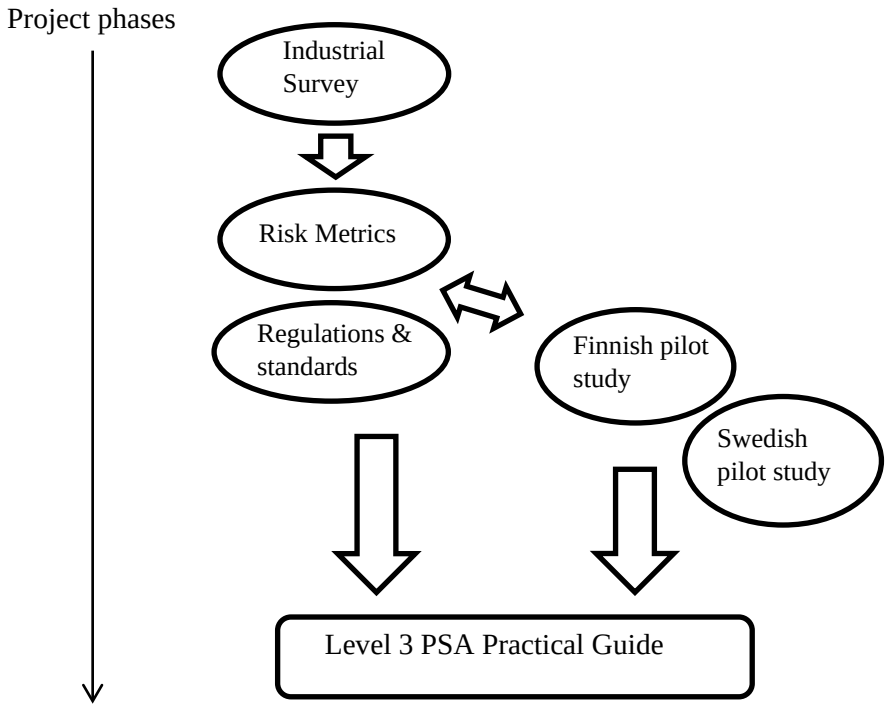
3.2.2. Do you account for correlation between causes of accident sequence initiating events and offsite phenomenological and consequence modelling? If you do, how is this correlation treated?

Not applicable

3.3 Are you participating in any ongoing research and development activities related to offsite radiological consequence analyses? If you are, please provide a brief description of these activities.

Yes, The Nordic PSA group has been financing (together with SSM) a level 3 project. This project also has links to the Finnish SAFIR program. The project has several parts:

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. In the short term this experience will be valuable for adding quality to Level 1 and 2 PSA. In the longer term, the work will set the foundation for performing a state-of-the-art Level 3 PSA. The final report for 2015 is still being edited and will be made available when completed. Part of this work will include ASME/ANS L3 PSA standards development.



Radionuclide Release Characterization

3.4 With regard to the interface between Level 2 and Level 3 PSA:

3.4.1. How do you handle the transition from Level 2 PSA analyses to Level 3 PSA analyses? What (if any) difficulties have you encountered in this area?

Not applicable

3.4.2. If you use representative release categories or source term groups, what criteria do you use to assign radiological release sequences in the Level 2 PSA model to release categories or source term groups? How do you address potential inter-sequence variability within each representative release category or source term group?

Not applicable

3.4.3. What process do you use to define release fraction timing (time and duration of release) and release truncation time? Do you consider onsite severe accident mitigation actions or offsite emergency response actions in this process? If you do, how are they treated?

Not applicable

3.5 What radionuclides do you use to characterize the offsite radiological consequences attributed to accidental releases of radiological materials? What is your basis for selecting these radionuclides?

Not applicable, however focus on Cs for long term (ground contamination) impacts and noble gases and Iodine for short term impacts.

3.6 Do you consider releases from multiple units or radiological sources co-located at the same site? If you do, how are these treated?

The individual plant analyses do not currently consider multiple sources.

3.7 Do you perform sensitivity or uncertainty analyses on release categories or release fractions? If you do, what methods are used and what is the basis for using them?

Not applicable.

Meteorological Data

3.8 How do you account for temporal and spatial variability in meteorological conditions? If you use a subset (sampling) of available meteorological data to serve as a representative data set, what criteria do you use for selecting these representative data?

Not applicable. However, radiological consequence analysis uses conservative assumptions.

3.9 Do you perform sensitivity or uncertainty analyses on meteorological conditions? If you do, what methods are used and what is the basis for using them?

See previous answer.

Environmental Transport and Dispersion

3.10 What atmospheric transport and dispersion (ATD) model(s) do you use? What process and criteria do you use to select: (1) an ATD model; and (2) a time scale for updating calculations to account for time dependence?

Not applicable

3.11 What process and criteria do you use to define the boundaries and intervals of the spatial grid or domain used for performing offsite consequence calculations?

Not applicable.

3.12 With regard to the spatial modelling around a nuclear power plant for Level 3 PSA:

3.12.1. What information sources do you use to develop geographical or topographical parameters? Do you use generic or site-specific data?

Not applicable.

3.12.2. Does the ATD response vary spatially with respect to varying topographic parameters? What is the spatial resolution of the ATD and do topographic parameters of the ATD vary within this resolution?

Not applicable

3.13 Do you consider radiological releases to water sources and aqueous transport and dispersion phenomena? If you do, how are these treated? If you have previously considered modelling radiological releases to water sources and aqueous transport and dispersion phenomena, but decided not to do so, what was the basis for your decision?

Not applicable, however radiological consequence analyses consider gas and water release that together makes up a certain source term released at specific elevations.

Protective Action (Countermeasure) Modelling

3.14 How do you define, and what protective actions (countermeasures) do you model for each of the nuclear or radiological incident or accident response phases listed below?

3.14.1. Early/Emergency Phase:
3.14.2. Intermediate Phase:
3.14.3. Late/Recovery Phase:

Not applicable

3.15 Do you model population groups with different protective action (countermeasure) behaviours? If you do, how do you define them? To what extent do you account for population density or age and gender distribution effects?

Not applicable

3.16 Do you use probabilistic models of protective action (countermeasure) behaviours that model the probabilities of success or failure for protective actions (countermeasures)? If you do, please describe the models and their bases.

Not applicable

3.17 What information sources do you use as a technical basis for protective action (countermeasure) modelling? Do you use generic or site-specific data?

Not applicable

Radiological Exposure and Dose Assessment

3.18 How do you define potentially exposed populations or cohorts? Do you consider onsite (e.g., worker) populations or potentially sensitive, vulnerable, or critical groups? If you do, what is your basis for including these groups, and how are they modelled?

Not applicable, however, the radiological consequence analysis use a set of standard groups.

3.19 What exposure pathways do you model? What is your basis for selecting these pathways? What exposure duration is assumed in your models?

Not applicable, however the radiological consequence analysis takes into account several pathways, e.g. direct radiation, from ground contamination and shining (clouds)

3.20 What information sources do you use to develop exposure and dose conversion factors? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

Radiological consequence analysis use generic factors (NRC?)

Radiological Health Effects

3.21 Do you model and estimate radiological health effects in your offsite radiological consequence analyses? If you do, please respond to questions 3.22 through 3.24. If you do not, please describe any other methods you use to address the offsite radiological health effects attributable to possible accident scenarios involving nuclear installations and proceed to question 3.25.

Not applicable

3.22 What types of early and latent health effects (including fatal and non-fatal effects) do you model? What target organs do you use for the associated health effects models?

3.23 What dose-response model(s) do you use to estimate the numbers of latent health effects attributable to radiological doses caused by accidental releases from nuclear installations, and what is the basis for their use? What (if any) methods do you use to account for uncertainty about the true dose-response relationship for exposures to low levels of ionizing radiation?

3.24 What information sources do you use to develop input parameters for the health effect models? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

Economic Consequences

3.25 Do you model and estimate economic consequences in your offsite radiological consequence analyses? If you do, please respond to questions 3.26 and 3.27. If you do not, please describe any other methods you use to address the offsite economic consequences attributable to possible accident scenarios involving nuclear installations and proceed to question 3.28.

Not applicable, the urban siting study from the 70ties focused on the public and early and late fatalities.

3.26 With regard to potential economic consequences considered within a Level 3 PSA:

3.26.1. What types of economic consequences do you model? What process do you use to select cost categories and models for estimating economic consequences?

3.26.2. If you translate radiological health effects into economic consequences, please describe your process for doing so.

3.26.3. If you model land contamination, please describe your process for estimating the health, environmental, or economic consequences attributed to land contamination.

Not applicable

3.27 What information sources do you use to develop cost parameters for economic consequence models? Do you use generic, region-specific, or site-specific data?

Not applicable

Consequence Quantification and Reporting

3.28 What process and criteria do you use for selecting consequence metrics for quantification and reporting, including spatial intervals or distances from release points?

Not applicable

Risk Characterization

3.29 What process and criteria do you use for selecting risk metrics for effectively communicating Level 3 PSA results, including metrics designed to measure the effects of radiological releases involving multiple units or radiological sources co-located at the same site, if applicable?

Not applicable

3.30 For each of the groups listed below, what methods do you use for presenting and communicating risk results and the uncertainty in risk results? How do you present the results from low-probability/high-consequence events in an understandable context for each group? Who is responsible for communicating the results to each group?

(1) Decision makers:
(2) General Public:

Not applicable

3.31 With regard to the treatment of uncertainties:

3.31.1. Do you perform sensitivity or uncertainty analyses as part of risk estimation? If you do, what process and methods do you use?
3.31.2. How do you determine which parameters will be varied or will have uncertainty distributions specified for propagating uncertainty? What parameters do you evaluate?
3.31.3. To what extent do you consider the effect of correlation on parameters?

Not applicable

Other

3.32 Are there any other questions that you believe should have been asked in this survey? Is there any other information or are there any other technical challenges or notable practices you would like to share with the international community?

Switzerland: ENSI

PART 1: RESPONDENT INFORMATION

Name:	**Rainer Hausherr**
Country:	**Switzerland**
Organization:	**Swiss Federal Nuclear Safety Inspectorate, ENSI**
Type of Organization:	**X Regulatory Authority** **□ Utility** **□ Vendor** **□ Academic/Research** **□ Other (please specify): _____**
Mailing Address:	**Swiss Federal Nuclear Safety Inspectorate, ENSI** **Industriestrasse 19** **CH-5200 Brugg** **Switzerland**
E-mail Address:	**Rainer.Hausherr@ensi.ch**
Telephone Number:	**+41 56 460 86 15**

PART 2: APPLICATION OF LEVEL 3 PSA

2.1 Does your country require the performance of Level 3 PSAs for nuclear installations? If yes, for what types of applications are Level 3 PSAs required?

ENSI: There is no regulatory requirement to perform a Level 3 PSA.

2.2 With regard to the use of Level 3 PSA:

2.2.1. Do you perform, or are you considering performing Level 3 PSAs for nuclear installations? If you do, for what types of applications do you perform them?

ENSI: No

2.2.2. What calculated metrics or results from Level 3 PSAs are used in these applications and what (if any) requirements, goals, or criteria are used to evaluate these results? What is the basis for these requirements, goals, or decision criteria?

ENSI: not applicable

2.2.3. If you are not currently performing or considering performing Level 3 PSAs, have you previously considered performing them? If you have, what was the basis for your decision to not perform Level 3 PSA?

ENSI: The legal requirements do not ask for a Level 3 PSA. The Level 3 PSA is considered less effective to identify specific plant safety improvements than a Level 1 or 2 PSA since the level 3 PSA incorporates phenomena like wind direction which increases the uncertainty of the results and hinder the identification of specific plant potential improvements from the PSA point of view.

Please list any barriers you perceive to performing and applying Level 3 PSA in your country.

ENSI: There are no legal requirements to perform a Level 3 PSA.

2.2.4. What (if any) alternative methods do you use to estimate offsite public risks attributable to accidental releases of radiological materials from nuclear installations, and for what types of applications are they used?

ENSI: The regulatory guideline ENSI-A05 requires the computation of two Level 2 PSA risk metrics:

- **Large early release frequency (LERF): The LERF addresses the risk of a large release (more than $2 \cdot 10^{15}$ Bq I-131) within the first 10 hours after core damage. According to the Swiss emergency organization concept, a major part of the affected population would be protected against immediate effects of the accident within 5 hours. To consider uncertainties a time frame of 10 hours is used for LERF.**
- **Large release frequency (LRF): The LRF is defined as the number of events per year with a release of more than $2 \cdot 10^{14}$ Cs-137. Long-term soil contamination is caused by releases of nuclides with long half-life periods. Cs-137 with a half-life of about 30 years is the most representative of these nuclides.**

2.3 One application that could potentially benefit from Level 3 PSAs is the siting of nuclear installations, including establishing the size and boundary (shape) for each emergency

planning or protective action zone. Are Level 3 PSAs used to support the siting of nuclear installations in your country? If they are, how are they used?

ENSI: Switzerland has decided to phase out nuclear power, so siting of new NPPs is not an issue. After the Fukushima accident, Switzerland revisited the emergency plans and the protective action zones. In a first step, the NPPs had to calculate the frequency of exceeding various source terms. However, in the end it was decided that PSA insights shall not be used for this revision but to base the revision on worst case scenarios.

2.4 With regard to nuclear power plant emergency planning zones:

2.4.1. What process is used to establish sizes and boundaries for emergency planning zones around a nuclear installation in which arrangements shall be made at the preparedness stage for effectively taking protective and other response actions?

ENSI: The bases for the emergency planning zone concept were established in the late 70s and beginning of the 80s of last century. The few PSA-analysis results (WASH-1400) that were available at that time were taken into account. The boundary of the inner zone (zone 1 that is the PAZ) was set on the basis of a radiological criteria, the outer boundary of the adjacent zone 2 (corresponding to the UPZ) was set using engineering judgement. A heightened preparedness compared to the levels of preparedness elsewhere in the country was not deemed necessary beyond the outer boundary of zone 2. The radiuses of the zones were not changed since their establishment.

2.4.2. What stakeholder groups are involved in the process and what are their respective responsibilities?

ENSI: All levels are involved to various degrees, the federal instances, the cantons, the communes and the licensees.

2.4.3. Under what conditions (if any) can emergency planning zones be reduced in size? Can a Level 3 PSA be used to establish a probabilistic cut off for events that need to be considered for emergency planning purposes?

ENSI: As there is no internationally agreed cut off or risk beyond which increased preparedness is not deemed reasonable, a reduction of zone sizes is politically not enforceable.

2.5 Another application that could potentially benefit from Level 3 PSAs is the development of safety goals or risk acceptance criteria. Are Level 3 PSAs used to support this type of application in your country? If they are, how are they used?

ENSI: No

2.6 Offsite radiological consequence analyses may be performed to support applications that are not related to Level 3 PSA. Please list any additional applications supported by offsite radiological consequence analyses that you support or perform.

ENSI: For design basis accidents, the licensees have to assess doses to individuals located at certain distances of the plant for specified periods of time after a postulated fission product release.

PART 3: LEVEL 3 PSA MODELING ISSUES AND TECHNICAL CHALLENGES

General

3.1 What computer code(s) do you use, or are you developing, to perform the tasks listed below?

3.1.1. Offsite radiological consequence analyses:

ENSI: Codes used depend on the purpose they are used for. For design basis accidents and the purpose of verifying that the resulting doses remain below levels set by the radiation protection ordinance, a Gaussian plume model with a constant weather pattern (Pasquill-Gifford classes with fixed wind direction and velocity) and flat terrain is used. The aim is to provide a conservative assessment of the consequences; details can be found in the regulatory guideline ENSI-G14. For emergency response purposes, a complex Lagrangian model called LASAT using periodically updated numerical weather prognosis input from MeteoSwiss as well as complex terrain is used instead. This code runs within the decision support tool JRODOS and aims to provide a best-estimate prognosis as a basis for suggesting protective actions to the decision makers.

3.1.2. Risk characterization:

ENSI: Not applicable

3.2 With regard to the scope of Level 3 PSAs:

3.2.1. What internal and external accident sequence initiating events and hazards are considered in developing a Level 3 PSA?

ENSI: Not applicable

3.2.2. Do you account for correlation between causes of accident sequence initiating events and offsite phenomenological and consequence modelling? If you do, how is this correlation treated?

ENSI: Not applicable

3.3 Are you participating in any ongoing research and development activities related to offsite radiological consequence analyses? If you are, please provide a brief description of these activities.

ENSI: The system used for emergency purposes (JRODOS/LASAT) is used by several other countries as well. Research and development is ongoing with Switzerland participating mainly in steering and funding. The code used for analysing radiological DBA consequences is mature and has experienced few if any developments for years.

Radionuclide Release Characterization

3.4 With regard to the interface between Level 2 and Level 3 PSA:

3.4.1. How do you handle the transition from Level 2 PSA analyses to Level 3 PSA analyses? What (if any) difficulties have you encountered in this area?

ENSI: Not applicable

3.4.2. If you use representative release categories or source term groups, what criteria do you use to assign radiological release sequences in the Level 2 PSA model to release categories or source term groups? How do you address potential inter-sequence variability within each representative release category or source term group?

ENSI: Similarities regarding accident progression and source term, at least containment status (open, vented, isolated, non-isolated, bypassed, ruptured, base mat penetrated), time of release (early or late), mode of ex-vessel releases (dry or submerged core concrete interaction), and containment fission product removal mechanisms (scrubbing by containment spray or by an overlaying water pool). The source terms are usually calculated by using MELCOR. Sometimes there are different calculations available for a release category, allowing the definition of a mean source term for this release category. Generally, no uncertainty analysis regarding source terms is required.

3.4.3. What process do you use to define release fraction timing (time and duration of release) and release truncation time? Do you consider onsite severe accident mitigation actions or offsite emergency response actions in this process? If you do, how are they treated?

ENSI: Release fraction timing is done by using the corresponding information of the MELCOR calculations. The PSAs consider accident mitigation actions, among them, filtered containment venting and containment spray. For example, filtered venting is activated by rupture of a burst disc at a certain containment pressure.

3.5 What radionuclides do you use to characterize the offsite radiological consequences attributed to accidental releases of radiological materials? What is your basis for selecting these radionuclides?

ENSI: In the Level 2 PSA, at least the following radiological groups shall be considered: Xe, I including CsI, Cs, Te, Ba, Mo, Ru, Ce, and La.

For the computation of radiological consequences for emergency purposes up to 100 nuclides are used as set by an ENSI directive. The selection of a set of nuclides depends on the goal aimed at; in an emergency case fewer nuclides are considered, depending on knowledge of the situation and available simulation run time.

3.6 Do you consider releases from multiple units or radiological sources co-located at the same site? If you do, how are these treated?

ENSI: There is only one twin unit site in Switzerland. In an emergency, the tool used for atmospheric dispersion calculations JRODOS is able to simulate the effects of multiple sources at different heights released in parallel and with different time structure.

3.7 Do you perform sensitivity or uncertainty analyses on release categories or release fractions? If you do, what methods are used and what is the basis for using them?

ENSI: No.

Meteorological Data

3.8 How do you account for temporal and spatial variability in meteorological conditions? If you use a subset (sampling) of available meteorological data to serve as a representative data set, what criteria do you use for selecting these representative data?

ENSI: As no PSA-L3 are performed in Switzerland, no such variations are performed.

3.9 Do you perform sensitivity or uncertainty analyses on meteorological conditions? If you do, what methods are used and what is the basis for using them?

ENSI: No.

Environmental Transport and Dispersion

3.10 What atmospheric transport and dispersion (ATD) model(s) do you use? What process and criteria do you use to select: (1) an ATD model; and (2) a time scale for updating calculations to account for time dependence?

ENSI: (1) see 3.1.1. (2) In an emergency, as soon as new and/or more detailed source term information is available (potentially having an impact on the dose prognosis) a new simulation will be performed. As for time dependence, our numerical weather prognosis (NWP) data has a time resolution of 10 min and dispersion simulations are calculated with this time step by default. Every three hours we receive a new set of NWP data covering the next 24 hrs. Thus, we have a sound data basis for performing dispersion calculations and simulate to sufficient degree any time variability of a source term.

3.11 What process and criteria do you use to define the boundaries and intervals of the spatial grid or domain used for performing offsite consequence calculations?

ENSI: The NWP data (see 3.10) obtained from MeteoSwiss covers an area of 300 km x 300 km. In case of an emergency, the first simulations will inevitably be performed on the entire data area. Subsequent dispersion calculations may be restricted to the area of interest, depending on the meteorological situation and the results of the previous simulations. On the technical side, the simulation grid can be configured to use up to fivefold nesting with horizontal resolution reduced by half from one nesting step to the next (obviously, the innermost nesting step has highest resolution). The grid cell size of the innermost nesting can be chosen from a series of values, between 50 m and 1 km with default being 250 m. Vertical level spacing is taken from input NWP data.

3.12 With regard to the spatial modelling around a nuclear power plant for Level 3 PSA:

3.12.1. What information sources do you use to develop geographical or topographical parameters? Do you use generic or site-specific data?

ENSI: Not applicable

3.12.2. Does the ATD response vary spatially with respect to varying topographic parameters? What is the spatial resolution of the ATD and do topographic parameters of the ATD vary within this resolution?

ENSI: Not applicable

3.13 Do you consider radiological releases to water sources and aqueous transport and dispersion phenomena? If you do, how are these treated? If you have previously considered modelling radiological releases to water sources and aqueous transport and dispersion phenomena, but decided not to do so, what was the basis for your decision?

ENSI: For emergency purposes releases to water sources are only considered within the framework of verification of compliance with maximum releases allowed for normal operations. In emergency situations, aqueous transport can only be estimated grossly based on rules of thumb.

Protective Action (Countermeasure) Modelling

3.14 How do you define, and what protective actions (countermeasures) do you model for each of the nuclear or radiological incident or accident response phases listed below?

3.14.1. Early/Emergency Phase:
3.14.2. Intermediate Phase:
3.14.3. Late/Recovery Phase:

ENSI: Not applicable

3.15 Do you model population groups with different protective action (countermeasure) behaviours? If you do, how do you define them? To what extent do you account for population density or age and gender distribution effects?

ENSI: Not applicable

3.16 Do you use probabilistic models of protective action (countermeasure) behaviours that model the probabilities of success or failure for protective actions (countermeasures)? If you do, please describe the models and their bases.

ENSI: Not applicable

3.17 What information sources do you use as a technical basis for protective action (countermeasure) modelling? Do you use generic or site-specific data?

ENSI: Not applicable

Radiological Exposure and Dose Assessment

3.18 How do you define potentially exposed populations or cohorts? Do you consider onsite (e.g., worker) populations or potentially sensitive, vulnerable, or critical groups? If you do, what is your basis for including these groups, and how are they modelled?

ENSI: Not applicable

3.19 What exposure pathways do you model? What is your basis for selecting these pathways? What exposure duration is assumed in your models?

ENSI: Not applicable

3.20 What information sources do you use to develop exposure and dose conversion factors? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

ENSI: Not applicable

Radiological Health Effects

3.21 Do you model and estimate radiological health effects in your offsite radiological consequence analyses? If you do, please respond to questions 3.22 through 3.24. If you do not, please describe any other methods you use to address the offsite radiological health effects attributable to possible accident scenarios involving nuclear installations and proceed to question 3.25.

ENSI: Not applicable

3.22 What types of early and latent health effects (including fatal and non-fatal effects) do you model? What target organs do you use for the associated health effects models?

ENSI: Not applicable

3.23 What dose-response model(s) do you use to estimate the numbers of latent health effects attributable to radiological doses caused by accidental releases from nuclear installations, and what is the basis for their use? What (if any) methods do you use to account for uncertainty about the true dose-response relationship for exposures to low levels of ionizing radiation?

ENSI: Not applicable

3.24 What information sources do you use to develop input parameters for the health effect models? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

ENSI: Not applicable

Economic Consequences

3.25 Do you model and estimate economic consequences in your offsite radiological consequence analyses? If you do, please respond to questions 3.26 and 3.27. If you do not, please describe any other methods you use to address the offsite economic consequences attributable to possible accident scenarios involving nuclear installations and proceed to question 3.28.

ENSI: No, economic consequences are not considered.

3.26 With regard to potential economic consequences considered within a Level 3 PSA:

3.26.1. What types of economic consequences do you model? What process do you use to select cost categories and models for estimating economic consequences?
3.26.2. If you translate radiological health effects into economic consequences, please describe your process for doing so.
3.26.3. If you model land contamination, please describe your process for estimating the health, environmental, or economic consequences attributed to land contamination.

ENSI: Not applicable

3.27 What information sources do you use to develop cost parameters for economic consequence models? Do you use generic, region-specific, or site-specific data?

ENSI: Not applicable

Consequence Quantification and Reporting

3.28 What process and criteria do you use for selecting consequence metrics for quantification and reporting, including spatial intervals or distances from release points?

ENSI: Not applicable

Risk Characterization

3.29 What process and criteria do you use for selecting risk metrics for effectively communicating Level 3 PSA results, including metrics designed to measure the effects of radiological releases involving multiple units or radiological sources co-located at the same site, if applicable?

ENSI: Not applicable

3.30 For each of the groups listed below, what methods do you use for presenting and communicating risk results and the uncertainty in risk results? How do you present the results from low-probability/high-consequence events in an understandable context for each group? Who is responsible for communicating the results to each group?

(1) Decision makers:
(2) General Public:

ENSI: Not applicable

3.31 With regard to the treatment of uncertainties:

3.31.1. Do you perform sensitivity or uncertainty analyses as part of risk estimation? If you do, what process and methods do you use?
3.31.2. How do you determine which parameters will be varied or will have uncertainty distributions specified for propagating uncertainty? What parameters do you evaluate?
3.31.3. To what extent do you consider the effect of correlation on parameters?

ENSI: Not applicable

Other

3.32 Are there any other questions that you believe should have been asked in this survey? Is there any other information or are there any other technical challenges or notable practices you would like to share with the international community?

ENSI: No

United States: USNRC

PART 1: RESPONDENT INFORMATION

Name:	**Keith L. Compton***
Country:	**United States**
Organization:	**Office of Nuclear Regulatory Research, USNRC**
Type of Organization:	**x** Regulatory Authority ☐ Utility ☐ Vendor ☐ Academic/Research ☐ Other (please specify): _____
Mailing Address:	**U.S. Nuclear Regulatory Commission** **Mail Stop T10-B58** **Washington, DC 20555-0001**
E-mail Address:	Keith.Compton@nrc.gov
Telephone Number:	**+1-301-415-0651**

*** The following members of the USNRC technical staff reviewed and supplemented this example survey response:**

Elijah Dickson, Office of Nuclear Reactor Regulation
Daniel Hudson, Office of Nuclear Regulatory Research
Tony Nakanishi, Office of New Reactors
Donald Palmrose, Office of New Reactors
Jason Schaperow, Office of New Reactors
Nathan Siu, Office of Nuclear Regulatory Research

PART 2: APPLICATION OF LEVEL 3 PSA

2.1 Does your country require the performance of Level 3 PSAs for nuclear installations? If yes, for what types of applications are Level 3 PSAs required?

The term Level 3 PSA can refer to both an integrated Level 1/2/3 PSA[1] conducted in accordance with industry standards, or simply to the Level 3 offsite radiological probabilistic consequence analysis (PCA) element of such an assessment. Some of the applications involving PCA that are cited in this example survey response were not conducted as part of an integrated Level 1/2/3 PSA, but the offsite radiological consequence analysis was performed using methodologies substantially similar to those of a Level 3 PSA. In this example survey response, the acronym PCA will refer to either the Level 3 offsite radiological PCA element of an integrated Level 1/2/3 PSA or to an offsite radiological PCA performed for other purposes; the term Level 3 PSA will refer to an integrated Level 1/2/3 PSA.

Level 3 PSA is not a regulatory requirement for licensing of U.S. commercial nuclear power plants. However, information about the offsite consequences of severe accidents is required for compliance with environmental assessment requirements specified in national-level legislative statutes,[2] and this information is often generated using PCA. For more information, see the response to item 2.2.

2.2 With regard to the use of Level 3 PSA:

2.2.1. Do you perform, or are you considering performing Level 3 PSAs for nuclear installations? If you do, for what types of applications do you perform them?

The USNRC is currently performing a full-scope integrated site Level 3 PSA for a U.S. commercial nuclear power plant site as part of a research project that is described in more detail in the response to item 3.3.[3,4]

In addition to this and other USNRC-sponsored Level 3 PSAs that have been performed for as part of research projects (e.g., WASH-1400[5] and NUREG-1150[6]), information

[1] An integrated Level 1/2/3 PSA is an assessment of the offsite public risks attributable to a spectrum of possible accident scenarios involving a nuclear installation. In the traditional PSA framework for commercial nuclear power plants, such a Level 1/2/3 PSA includes three progressive levels of analysis: (1) core damage accident or plant damage state frequency analysis; (2) accident progression, containment performance, and radiological release analysis; and (3) offsite radiological consequence analysis. In this context, the term "integrated" means the three levels of analysis are linked through the use of logic or mathematical models to support development of qualitative insights and quantitative estimates of selected risk metrics.

[2] Title 42 United States Code, §§4321-4370h, *"National Environmental Policy Act."* Available at: https://elr.info/sites/default/files/docs/statutes/full/nepa.pdf.

[3] Hudson DW, Stutzke M. *Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities. SECY-11-0089.* Washington, DC: U.S. Nuclear Regulatory Commission; 2011. Available at: http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2011/2011-0089scy.pdf.

[4] Vietti-Cook, AL. *Staff Requirements – SECY-11-0089 – Options for Proceeding with Future Level 3 Probabilistic Risk Assessment (PRA) Activities.* Washington, DC: U.S. Nuclear Regulatory Commission; 2011. Available at: http://pbadupws.nrc.gov/docs/ML1126/ML112640419.pdf.

[5] U.S. Nuclear Regulatory Commission. *Reactor Safety Study. An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants. WASH-1400 (NUREG 75/014).* Washington, DC: U.S. Nuclear Regulatory Commission; 1975.

[6] Office of Nuclear Regulatory Research. *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants. NUREG-1150.* Washington, DC: U.S. Nuclear Regulatory Commission; 1990. Available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1150/.

from PCA is typically used to assess the offsite radiological consequences of severe or beyond-design-basis accidents (BDBAs).[1] Applications of PCA at the USNRC include:

- **Regulatory[2,3] and backbit analyses[4,5,6] to support decisions regarding proposed regulatory actions;**
- **Environmental assessment reviews with respect to severe accidents and Severe Accident Mitigation Alternatives (SAMA) analyses for operating power reactor license renewal[7] or Severe Accident Mitigation Design Alternatives (SAMDA) analyses for new power reactor design stage applications;[8] and**
- **Supporting applied research studies. Recent examples of such studies include: (1) the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project;[9] (2) a consequence study of a beyond-design-basis earthquake affecting the spent fuel pool for a U.S. Mark I boiling water reactor;[10] and (3) an evaluation of alternative protective action strategies for emergency response to severe accidents.[11]**

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[1] The USNRC defines a design-basis accident (DBA) as: "a postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety" (http://www.nrc.gov/reading-rm/basic-ref/glossary/design-basis-accident.html). Beyond-design-basis accidents (BDBAs) are defined as: "accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely." BDBAs are thus considered to be beyond the scope of DBAs that a nuclear facility must be designed and built to withstand (http://www.nrc.gov/reading-rm/basic-ref/glossary/beyond-design-basis-accidents.html).

[2] Division of Systems Analysis and Operational Effectiveness, Office of Nuclear Regulatory Research. *Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission. NUREG/BR-0058, Revision 4*. Washington, DC: U.S. Nuclear Regulatory Commission; 2004. Available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0058/br0058r4.pdf.

[3] Office of Nuclear Regulatory Research. *Regulatory Analysis Technical Evaluation Handbook. NUREG/BR-0184.* Washington, DC: U.S. Nuclear Regulatory Commission; 1997. Available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0058/br0058r4.pdf.

[4] A back fit is a generic or plant-specific modification that becomes effective after specified dates. Examples of back fits include modification of or addition to: (1) facility structures, systems, components, or design; (2) the design approval or manufacturing license for a facility; or (3) the procedures or organization required to design, construct or operate a facility. Any of these modifications or additions may result from a new or amended provision in USNRC regulations or the imposition of a regulatory staff position interpreting USNRC regulations that is either new or different from a previously applicable staff position.[11]

[5] U.S. Nuclear Regulatory Commission. *Title 10 Code of Federal Regulations Part 50, §50.109, "Back fitting."* Available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0109.html.

[6] Allison DP, Conran JM, Trottier CA. *Back fitting Guidelines. NUREG-1409.* Washington, DC: U.S. Nuclear Regulatory Commission; 1990. Available at: http://pbadupws.nrc.gov/docs/ML0322/ML032230247.pdf.

[7] U.S. Nuclear Regulatory Commission. *Standard Review Plans for Environmental Reviews for Nuclear Power Plants: Environmental Standard Review Plan (with Supplement 1 for Operating Reactor License Renewal). NUREG-1555.* Washington, DC: U.S. Nuclear Regulatory Commission; 2000. Available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1555/.

[8] U.S. Nuclear Regulatory Commission. *Title 10 Code of Federal Regulations Part 51, §51.30, "Environmental Assessment."* Available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part051/part051-0030.html.

[9] Chang R, Schaperow J, Ghosh T, Barr J, Tinkler C, Stutzke M. *State-of-the-Art Reactor Consequence Analyses (SOARCA) Report. NUREG-1935.* Washington, DC: U.S. Nuclear Regulatory Commission; 2012. Available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1935/.

[10] Barto A, Chang Y J, Compton K, Esmaili H, Helton D, Murphy A et al. *Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor. NUREG-2161*. Washington, DC: U.S. Nuclear Regulatory Commission; 2014. Available at: http://pbadupws.nrc.gov/docs/ML1425/ML14255A365.pdf.

[11] Sullivan R, Jones J, Schelling FJ, Bixler N, Walton F. *Review of NUREG-0654, Supplement 3, "Criteria for Protective Action Recommendations for Severe Accidents": Technical Basis for Protective Action Strategies. NUREG/CR-6953, Vol. 3*. Washington, DC: U.S. Nuclear Regulatory Commission; 2010. Available at: http://pbadupws.nrc.gov/docs/ML1023/ML102380087.pdf.

PCAs for design basis accidents (DBAs) are more prescriptive than PCAs for BDBAs. For example, U.S. regulations require offsite radiological consequence analyses for DBAs to assess doses to individuals located at any point along the boundaries of specified areas around a nuclear power plant for specified periods of time following postulated fission product releases.[1,2]

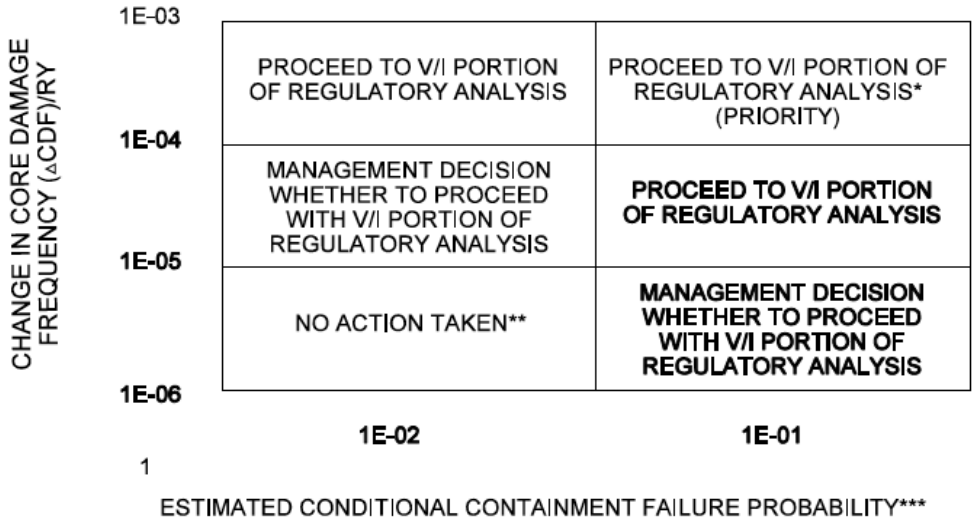
2.2.2. What calculated metrics or results from Level 3 PSAs are used in these applications and what (if any) requirements, goals, or criteria are used to evaluate these results? What is the basis for these requirements, goals, or decision criteria?

A high-level summary response to this survey item is provided below in Table 1. Additional information about application-specific metrics, criteria, and their bases is provided in a more detailed response after the table and figure.

Table 1. Application-Specific Level 3 PSA Metrics and Criteria

Application	Level 3 PSA Metric	Criteria
Safety Goal Screening Evaluation for Regulatory and Backfit Analyses	Change in Core Damage Frequency[a] Conditional Containment Failure Probability[b]	See Figure 1
Regulatory and Backfit Analyses SAMA and SAMDA Analyses	Averted Population Dose (0-50 miles) Averted Offsite Property Damage (0-50 miles)	Net value
[a] The USNRC uses a subsidiary numerical objective related to Level 1 PSA core damage frequency (CDF) results as a surrogate for a quantitative health [effects] objective related to average individual latent cancer fatality risk within 10 miles of a commercial nuclear power plant.		
[b] The USNRC uses a subsidiary numerical objective related to Level 2 PSA large early release frequency (LERF) results as a surrogate for a quantitative health [effects] objective related to average individual early fatality risk within 1 mile of a commercial nuclear power plant.		

Figure 1. Safety Goal Screening Criteria for Regulatory and Back fit Analyses[8]



[1] U.S. Nuclear Regulatory Commission. *Title 10 Code of Federal Regulations Part 50, §50.34, "Contents of Applications; Technical Information."* Available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0034.html.

[2] U.S. Nuclear Regulatory Commission. *Title 10 Code of Federal Regulations Part 50, §50.67, "Accident Source Term."* Available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0067.html.

The primary analytical technique used to evaluate alternatives in regulatory, SAMA, and SAMDA analyses is cost-benefit analysis. In these analyses, the principal metric calculated to support decision making is the net present value of net benefits (net value), which represents the difference between the sum of monetized and discounted benefits and the sum of monetized and discounted costs. However, a number of decision criteria can be used to select and support an appropriate course of action. For example, decision criteria that can be used in regulatory analyses include: (1) calculated net value; (2) relative importance of attributes quantified in other than monetary terms; (3) relative importance of qualitative attributes (attributes that are not quantified); (4) relationship and consistency of proposed alternatives with existing USNRC legislative mandate, policies, and processes; and (5) impacts of proposed alternatives on existing or planned USNRC programs or requirements.[8]

As part of regulatory, SAMA, or SAMDA analyses, results from Level 3 PSAs can be used to evaluate and screen proposed alternatives based on the magnitude of the estimated benefit in terms of enhancing safety or reducing risk. For example, regulatory analysis guidelines8 include guidance for performing an evaluation of proposed regulatory actions with respect to the USNRC safety goals for nuclear power plant operations.[1] This safety goal evaluation is designed to identify when a regulatory requirement should not be imposed generically on nuclear power plants because the residual risk is already acceptably low; it is intended to eliminate some proposed regulatory actions from further consideration, regardless of whether they could be justified on the basis of their net value. This safety goal evaluation can also be used to determine whether a proposed generic safety enhancement back fit that does not meet certain exemption criteria provides a substantial increase in the overall protection of public health and safety[11] to warrant further evaluation of the benefits and costs to determine whether they are justified on their net value basis.[8]

Regulatory analysis guidelines include explicit safety goal screening criteria related to: (1) changes in the frequency of core damage accidents; and (2) conditional containment failure probabilities. These criteria - which are intended to provide a balanced consideration of measures to prevent and mitigate core damage accidents - can be used to evaluate results from Level 1 and Level 2 PSAs to determine conformity with subsidiary safety goal objectives based on core damage frequency (CDF) and large early release frequency (LERF).[8,9] Although these guidelines do not include explicit screening criteria related to the safety goal quantitative health [effects] objectives (QHOs) for average individual early fatality risk and average individual latent cancer fatality risk, results from Level 3 PSAs for these metrics can be used to evaluate proposed regulatory actions with respect to the safety goals. Additional information about the basis for the safety goals and the QHOs is provided in the response to item 2.5.

For those proposed regulatory actions that pass the safety goal screening evaluation, a detailed cost-benefit analysis is performed to estimate the net value. The principal outputs from a PCA or Level 3 PSA that serve as inputs to the cost-benefit analysis are the averted population dose (collective effective dose) and averted economic costs, including offsite property damage. Although these metrics are typically calculated for the offsite region within 50 miles of the nuclear installation, other spatial intervals can be used on a case-by-case basis.[8,9] The averted population dose is then monetized using a dollar per person-rem conversion factor that combines the value of a statistical life with a nominal risk coefficient for stochastic health effects per person-rem. The USNRC is

[1] U.S. Nuclear Regulatory Commission. *Safety Goals for the Operations of Nuclear Power Plants; Policy Statement; Republication. 51 FR 30028.* Washington, DC: U.S. Nuclear Regulatory Commission; 1986. Available at: http://www.nrc.gov/reading-rm/doc-collections/commission/policy/51fr30028.pdf.

currently revising the value of this dollar per person-rem conversion factor and is proposing a process for routine updates to ensure its value is maintained current.[1]

2.2.3. If you are not currently performing or considering performing Level 3 PSAs, have you previously considered performing them? If you have, what was the basis for your decision to not apply Level 3 PSA? Please list any barriers you perceive to performing and applying Level 3 PSA in your country.

Not applicable.

2.2.4. What (if any) alternative methods do you use to estimate offsite public risks attributable to accidental releases of radiological materials from nuclear installations, and for what types of applications are they used?

Not applicable.

2.3 One application that could potentially benefit from Level 3 PSAs is the siting of nuclear installations, including establishing the size and boundary (shape) for each emergency planning or protective action zone. Are Level 3 PSAs used to support the siting of nuclear installations in your country? If they are, how are they used?

A Level 3 PSA is not required to support siting of nuclear power plants. However, U.S. regulations require results from prescriptive DBA dose assessments to demonstrate that doses to individuals located at any point along the boundaries of specified areas around a nuclear power plant for specified periods of time following postulated fission product releases are within prescribed limits.[18,19] Additional regulations require development of an environmental impact statement (EIS) to support siting decisions for specified types of nuclear installations.[2] The EIS includes an assessment of the offsite radiological consequences for BDBAs; either a PCA or Level 3 PSA can be used to provide such an assessment.

2.4 With regard to nuclear power plant emergency planning zones:

2.4.1. What process is used to establish sizes and boundaries for emergency planning zones around a nuclear installation in which arrangements shall be made at the preparedness stage for effectively taking protective and other response actions?

A task force comprised of representatives from the USNRC and the U.S. Environmental Protection Agency (USEPA) developed the process and technical basis for establishing the sizes of generic emergency planning zones (EPZs) around U.S. nuclear power plants.[3] U.S. regulations require nuclear power plant licensees to develop an emergency response plan that specifies a range of preplanned protective actions that can be taken within these EPZs to reduce dose to emergency workers and to the public in the event of an accident.[4]

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[1] Office of Nuclear Reactor Regulation. *Reassessment of NRC's Dollar Per Person-Rem Conversion Factor Policy: Draft Report for Comment. NUREG-1530, Revision 1*. Washington, DC: U.S. Nuclear Regulatory Commission; 2015. Available at: http://pbadupws.nrc.gov/docs/ML1523/ML15237A211.pdf.

[2] U.S. Nuclear Regulatory Commission. *Title 10 Code of Federal Regulations Part 51, §51.20, "Criteria for and Identification of Licensing and Regulatory Actions Requiring Environmental Impact Statements."* Available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part051/part051-0020.html.

[3] Collins HE, Grimes BK, Galpin F. *Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants. NUREG-0396*. Washington, DC: U.S. Nuclear Regulatory Commission; 1978. Available at: http://pbadupws.nrc.gov/docs/ML0513/ML051390356.pdf.

[4] U.S. Nuclear Regulatory Commission. *Title 10 Code of Federal Regulations Part 50, §50.47, "Emergency Plans"* Available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0047.html.

These regulations define two generic EPZs for nuclear power plants that are based on the recommendations of the USNRC and USEPA task force. In general, the plume exposure pathway EPZ for large light-water reactor (LWR) nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. However, the exact size and configuration of the EPZs surrounding specific nuclear power plants are determined based on consideration of local emergency response needs and capabilities, which can be affected by many factors, including: demography, topography, land characteristics, access routes, and jurisdictional boundaries. The sizes of EPZs for gas-cooled reactors and for reactors with an authorized power level less than 250 MW thermal may be determined on a case-by-case basis.[24]

The USNRC and USEPA task force considered several possible rationales in developing the sizes for the existing generic EPZs, including: risk, probability, cost-effectiveness, and accident consequence spectrum. After evaluation of each approach, the task force decided to use the spectrum of possible accident consequences for different classes of nuclear accidents as the principal technical basis for emergency planning. The task force also chose to augment this principal technical basis with consideration of the probabilities of accidents that could result in offsite doses that exceed USEPA Protective Action Guide (PAG) levels[1,2] and that therefore could require emergency response.[23]

2.4.2. What stakeholder groups are involved in the process and what are their respective responsibilities?

The stakeholder groups involved in the process of establishing sizes and boundaries for EPZs around a nuclear installation, as well as their respective responsibilities, are summarized below in Table 2.

Table 2. Stakeholder Groups and Responsibilities in Establishing EPZ Sizes and Boundaries

[1] Office of Radiation Programs. *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents. EPA 400-R-92-001. Second Printing.* Washington, DC: U.S. Environmental Protection Agency; 1992. Available at: https://www.epa.gov/sites/production/files/2014-11/documents/00000173.pdf.

[2] U.S. Environmental Protection Agency. *PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents. Draft for Interim Use and Public Comment.* Washington, DC: U.S. Environmental Protection Agency; 2013. Available at: https://www.epa.gov/sites/production/files/2015-06/documents/pag-manual-interim-public-comment-4-2-2013.pdf.

Stakeholder Group	Responsibilities
U.S. Environmental Protection Agency (USEPA)	· Establishes federal radiation protection standards. · Establishes PAG levels used in developing EPZ basis. · Participated in joint USEPA/USNRC task force that established technical basis for existing EPZs.
U.S. Nuclear Regulatory Commission (USNRC)	· Establishes regulatory requirements for onsite radiological emergency preparedness and emergency plans. · Ensures adequacy of radiological emergency preparedness and emergency plans and issues operating licenses. · Participated in joint USEPA/USNRC task force that established technical basis for existing EPZs.
Federal Emergency Management Agency (FEMA)	· Establishes regulatory requirements for offsite radiological emergency preparedness and emergency plans.
Nuclear Power Plant Licensees	· Develop and implement onsite radiological emergency preparedness and emergency plans.
State, Local, and Tribal Governments	· Develop and implement offsite radiological emergency preparedness and emergency plans.
Other[a]	· Represent respective interests and perspectives and provide valuable input to the process.
[a] Examples of other stakeholder groups include, but are not limited to: (1) industry organizations; (2) public interest groups; (3) research organizations; (4) local emergency response organizations; and (5) public citizens.	

2.4.3. Under what conditions (if any) can emergency planning zones be reduced in size? Can a Level 3 PSA be used to establish a probabilistic cut off for events that need to be considered for emergency planning purposes?

The USNRC has not established criteria for reducing the size of EPZs for large LWR nuclear power plants. However, USNRC regulations specify that the sizes of EPZs for gas-cooled reactors and for reactors with an authorized power level less than 250 MW thermal may be determined on a case-by-case basis.[24]

Results and insights from the WASH-1400[5] Level 3 PSAs were used to provide the probabilistic perspective that informed the task force's recommendations with respect to the sizes of the existing generic EPZs. Although a probabilistic cut off was not explicitly specified for accidents that need to be considered for emergency planning purposes, the task force concluded that: (1) DBAs and less severe but more likely core damage accidents should be considered when selecting the technical basis; and (2) certain features of more severe but less likely core damage accidents should be considered to assure sufficient response capability exists for even the most severe accidents.[23]

2.5 Another application that could potentially benefit from Level 3 PSAs is the development of safety goals or risk acceptance criteria. Are Level 3 PSAs used to support this type of application in your country? If they are, how are they used?

The USNRC safety goals and corresponding QHOs were developed in the 1980's and fundamentally reflect a value judgment about the level of incremental (additional) risk to the public from nuclear power plant operations relative to background risk that was judged to be acceptable. Although results and insights from available Level 3 PSAs were not explicitly used to derive the safety goals and QHOs, they were used to determine: (1) the practicability of

measuring attainment of the QHOs; and (2) the need for consideration of uncertainty in evaluating the results from Level 3 PSAs in safety goal evaluations.

Additional information about the USNRC safety goals and QHOs, as well as the role of Level 3 PSAs in their development and implementation, is provided below. For a more comprehensive treatment, please refer to the 2009 WGRISK report on probabilistic risk criteria and safety goals.[1]

The USNRC safety goal policy statement[20] broadly defines an acceptable level of radiological risk to public health and safety from potential releases of radiological materials from commercial nuclear power plants.[2] This policy essentially addresses the question of "how safe is safe enough?" for regulatory decisions regarding nuclear power plant safety and guides agency evaluations of whether additional safety enhancements beyond those needed to ensure adequate protection are no longer justified because the level of residual risk to the public is acceptably low and limited resources could be better applied to other areas.

The USNRC used a hierarchical framework in establishing its safety goal policy. This framework is comprised of two high-level qualitative safety goals; one goal encompasses risks to individual members of the public, while the other addresses societal risk. Each qualitative safety goal is supported by a QHO that can be used to determine whether and to what extent the qualitative safety goals have been achieved; one QHO is for individual early (prompt) fatality risk, while the other is for individual latent cancer fatality risk.

Although results and insights from the WASH-1400 study[5] and from other industry-sponsored Level 3 PSAs were available during the development and trial evaluation of the safety goals, this information was not explicitly used to derive the qualitative safety goals or their corresponding QHOs. Instead, the safety goals and QHOs reflect a value judgment about the level of incremental (additional) risk to the public from nuclear power plant operations relative to background risk that was judged to be acceptable. Through stakeholder engagement and a deliberative process, the USNRC determined that the risk to an average individual from living within specified distances of a nuclear power plant should be no more than 0.1% (1/1000) of the other major risks to which they are exposed.[3]

Results and insights from WASH-1400[5] and from other available Level 3 PSAs were primarily used to determine the practicability of measuring attainment of the QHOs and the need for consideration of uncertainty in evaluating the results from Level 3 PSAs. However, results and insights from the NUREG-1150 study[6] were later used to demonstrate that the subsidiary

[1] Working Group on Risk Assessment, Committee on the Safety of Nuclear Installations. *Probabilistic Risk Criteria and Safety Goals. NEA/CSNI/R(2009)16.* Paris, France: Nuclear Energy Agency; 2009. Available at: https://www.oecd-nea.org/nsd/docs/2009/csni-r2009-16.pdf.

[2] The scope of the safety goal policy includes risks to the public arising from both routine and accidental radiological releases from commercial nuclear reactors. Public risks imposed by the nuclear fuel cycle were excluded based on earlier assessments that suggested fuel cycle risks were relatively small in comparison to operating reactor risks. In addition, EISs performed before existing nuclear power plants had been licensed to operate indicated there would be no measurable radiological impact on members of the public from routine operations. Moreover, since compliance with national radiation protection guidance and USNRC regulations was believed to ensure that public risks arising from routine emissions were comparatively small, the Commission expressed its belief that such risks need not be routinely analysed on a case-specific basis to demonstrate conformance with the safety goal policy. The USNRC safety goals therefore apply primarily to public risks arising from potential accidental releases of radiological materials from operating commercial nuclear reactors.

[3] Office of Policy Evaluation. *Safety Goals for Nuclear Power Plant Operation. NUREG-0880, Revision 1. For Comment.* Washington, DC: U.S. Nuclear Regulatory Commission; 1983. Available at: http://pbadupws.nrc.gov/docs/ML0717/ML071770230.pdf.

objectives for CDF and LERF are acceptable surrogates for the average individual latent cancer and early fatality risk QHOs, respectively.[1]

2.6 Offsite radiological consequence analyses may be performed to support applications that are not related to Level 3 PSA. Please list any additional applications supported by offsite radiological consequence analyses that you support or perform.

Offsite radiological consequence analyses are also performed to support incident response applications. In these applications, dose assessments are performed in real-time to inform or evaluate offsite protective action recommendations. As described in the response to item 3.1.1, different computer codes are used to perform offsite radiological consequence analyses or dose assessments for such non-PSA applications.

In addition to accidental releases of radiological materials, U.S. regulations require dose assessments for certain routine or off-normal releases to demonstrate compliance with specified dose limits for members of the public.[2] **However, the scope of this survey is assumed to be limited to offsite radiological consequence analyses for accidental releases of radiological materials.**

[1] Drouin, M. *Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing, Volumes 1 and 2. NUREG-1860.* Washington, DC: U.S. Nuclear Regulatory Commission; 2007. Available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1860/.

[2] U.S. Nuclear Regulatory Commission. *Title 10 Code of Federal Regulations Part 20, §20.1302, "Compliance with Dose Limits for Individual Members of the Public."* Available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part020/part020-1302.html.

PART 3: LEVEL 3 PSA MODELING ISSUES AND TECHNICAL CHALLENGES

General

3.1 What computer code(s) do you use, or are you developing, to perform the tasks listed below?

3.1.1. Offsite radiological consequence analyses:

The USNRC uses the MELCOR Accident Consequence Code System (MACCS)[1] to perform PCA or the offsite radiological consequence analysis element of a Level 3 PSA. Other codes are used to perform offsite radiological consequence or dose assessments for non-PSA applications. For example, the Radiological Assessment System for Consequence AnaLysis (RASCAL) code[2] and the HotSpot Health Physics codes[3] support incident response applications, while the RADionuclide, Transport, Removal, And Dose estimation (RADTRAD)[4] code supports DBA dose assessment.

3.1.2. Risk characterization:

The USNRC does not currently maintain a code specifically designed for performing risk estimation or risk integration by combining frequency estimates from Level 1 and Level 2 PSA analyses with conditional consequence metric estimates from Level 3 PSA analyses to calculate risk metrics. Although such a code was developed to support the NUREG-1150 Level 3 PSAs,[5] the USNRC is planning to use available commercial off-the-shelf software (MATLAB)[6] to perform the risk estimation task as part of its ongoing Full-Scope Integrated Site Level 3 Probabilistic Risk Assessment (PRA) Project;[3,4] a brief description of this project is provided in the response to item 3.3.

3.2 With regard to the scope of Level 3 PSAs:

3.2.1. What internal and external accident sequence initiating event hazards are considered in developing a Level 3 PSA?

The scope of accident sequence initiating event hazards considered in developing a Level 3 PSA is determined as part of the initiating-event analysis technical element of the Level 1 (core damage accident frequency analysis) PSA. Hazard groups that are typically considered in developing Level 1 PSA models for U.S. nuclear power plants are summarized below in Table 3.

1 http://maccs.sandia.gov/.

2 https://www.usnrc-ramp.com/RASCAL Overview.

3 https://narac.llnl.gov/hotspot.

4 https://www.usnrc-ramp.com/content/snapradtrad-overview.

5 Iman RL, Johnson JD, Helton JC. *PRAMIS: Probabilistic Risk Assessment Model Integration System. NUREG/CR-5262.* Washington, DC: U.S. Nuclear Regulatory Commission; 1990.

6 http://www.mathworks.com/products/matlab/?requestedDomain=www.mathworks.com.

Table 3. Hazard Groups Typically Considered in Level 1 PSA Models for U.S. Nuclear Power Plants[1]

Hazard Type	Hazard Group	Example Hazard or Initiating Events
Internal	Internal Events	Loss of Offsite Power (LOOP¾an internal event by convention)
		Transients
		Loss of Coolant Accidents (LOCAs)
		Steam Generator Tube Ruptures (SGTRs)
		High Energy Line Breaks (HELBs)
		Interfacing Systems LOCAs (ISLOCAs)
		Special Initiators (e.g. support system failures)
	Internal Floods	
	Internal Fires	
External	Seismic Events	
	High Winds	
	External Floods	
	Other Hazards	External Fires
		Transportation Accidents
		Other Industrial Accidents
		Extreme Temperatures
		Turbine-Generated Missiles

3.2.2. Do you account for correlation between causes of accident sequence initiating events and offsite phenomenological and consequence modelling? If you do, how is this correlation treated?

For some PCA studies, accident sequences are selected by technical judgment. For example, accident sequences were selected based on professional judgment using insights from site-specific Level 1/2 PSA models for sites analysed as part of the SOARCA Project.[15]

Some correlations between the causes of initiating events and offsite phenomenological and consequence modelling are typically considered. For example, the impact of seismic events on evacuation networks, evacuation speeds, and shielding parameters is typically explicitly evaluated and modelled for seismically initiated accident sequences.

3.3 Are you participating in any ongoing research and development activities related to offsite radiological consequence analyses? If you are, please provide a brief description of these activities.

Full-Scope Integrated Site Level 3 PRA Project[3,4]
The USNRC is performing a full-scope integrated site Level 3 PSA that aims to achieve the following high-level objectives:

- **Develop a Level 3 PSA, generally based on current state-of-practice methods, tools, and data, that: (1) reflects technical advances since completion of previous USNRC-sponsored Level 3 PSA studies; and (2) addresses limitations in the scope of these previous studies - including consideration of accidents involving multiple site radiological sources;**

[1] The American Society of Mechanical Engineers (ASME), American Nuclear Society (ANS). *Addenda to ASME/ANS RA-S-2008: Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications. ASME/ANS RA-Sb-2013.* New York, NY: The American Society of Mechanical Engineers; 2013.

- **Extract new risk insights to enhance regulatory decision-making and help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety;**
- **Enhance PRA staff capability and expertise and improve documentation practices to make PRA information more accessible, retrievable, and understandable; and**
- **Obtain insight into the technical feasibility and cost of developing new and more comprehensive Level 3 PSAs.**

The scope of this Level 3 PSA project includes: (1) all major site radiological sources, including all operating reactor units, spent fuel pools, and dry cask storage installations - with accidents involving individual sources and accidents involving different combinations of more than one source included within the scope; (2) all modes of reactor operation; and (3) all internal and external hazards, excluding deliberate malevolent acts.

Consistent with the high-level project objectives, the Level 3 PSA is generally being developed using current state-of-practice methods, tools, and data. However, new research and methodological development is being pursued to address a limited set of technical issues and challenges (e.g., development of methods for modelling, quantifying, and analysing multi-source accident scenarios - including treatment of dependent failures, human reliability analysis, accident management, multi-source radiological releases, multi-source offsite radiological consequence analysis, and candidate site-level risk metrics for characterizing integrated site risk).

Additional SOARCA Studies

SOARCA Uncertainty Analysis

Uncertainty analyses are being performed for specific accident scenarios that were evaluated as part of the original SOARCA study. These analyses are conditioned on the assumed occurrence of specified conditions in the progression of the modelled accident scenarios. The high-level objectives of these conditional uncertainty analyses are to:

- **Develop insights into the overall sensitivity of SOARCA results to uncertainty in inputs;**
- **Identify the most influential input parameters for accidental radiological releases and accident consequences; and**
- **Demonstrate the application of an uncertainty analysis methodology that could be used in future source term, PCA, or Level 3 PSA studies.**

The uncertainty analyses involve varying multiple uncertain model parameters using Monte Carlo sampling of parameter probability distributions. Subject matter experts were consulted to determine the most important uncertain parameters in accident progression, radiological release, and offsite radiological consequence models for variation. Multiple statistical regression techniques are then used to quantify uncertainty and to determine which parameters have the greatest influence on the results.

The results of the uncertainty analysis for the long-term station blackout accident scenario at the Peach Bottom Atomic Power Station are available in a draft report.[1] The uncertainty analysis for the Surry Power Station is still in progress.

[1] Sandia National Laboratories. *State-of-the-Art Reactor Consequence Analysis Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station. Draft Report. NUREG/CR-7155.* Washington, DC: U.S. Nuclear Regulatory Commission; 2016. Available at: http://pbadupws.nrc.gov/docs/ML1318/ML13189A145.pdf.

Sequoyah Nuclear Plant
A follow-on SOARCA study has been initiated to develop best estimates of the offsite radiological health consequences for select accident scenarios involving a pressurized water reactor (PWR) nuclear power plant with an ice condenser containment. Accident scenarios are being chosen to challenge this style of containment, which is smaller than the large dry containment used with other PWR nuclear power plants, including the Surry Power Station - which was evaluated as part of the original SOARCA Project.

The Sequoyah SOARCA study is applying lessons learned and best practices[1,2] from the original SOARCA Project. In addition, the effects of using diverse and flexible coping strategies involving portable equipment - which the U.S. nuclear industry implemented in response to challenges identified by the 2011 accident at the Fukushima Daiichi Nuclear Power Station - are also being modelled as part of this study to characterize their benefits.

Other Research Activities
The USNRC Office of Nuclear Regulatory Research is also performing or supporting a vast array of other research activities related to offsite radiological consequence analysis and Level 3 PSA. High-level summary information about the objectives, research approach, and status pertaining to notable research activities is provided in NUREG-1925.[3] Where appropriate, additional information about specific research activities that directly relate to items in this survey is provided in the response to the relevant survey items.

Radionuclide Release Characterization

3.4 With regard to the interface between Level 2 and Level 3 PSA:

3.4.1. How do you handle the transition from Level 2 PSA analyses to Level 3 PSA analyses? What (if any) difficulties have you encountered in this area?

See response to item 3.4.2.

3.4.2. If you use representative release categories or source term groups, what criteria do you use to assign radiological release sequences in the Level 2 PSA model to release categories or source term groups? How do you address potential inter-sequence variability within each representative release category or source term group?

The Level 2 PSA analyst selects radiological release categories in consultation with the Level 3 PSA analyst. The USNRC uses the MELCOR code[4] for modelling severe accident progression and estimation of source terms. Source term information from the Level 2 element is contained in MELCOR plot files (*.ptf) that are processed using the MELMACCS interface software utility[5] to extract the data needed for the offsite radiological consequence analysis and to generate the corresponding inputs for the MACCS code.

[1] Ross K, Phillips J, Gauntt RO, Wagner KC. *MELCOR Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project. NUREG/CR-7008.* Washington, DC: U.S. Nuclear Regulatory Commission; 2014. Available at: http://pbadupws.nrc.gov/docs/ML1423/ML14234A136.pdf.

[2] Bixler N, Jones J, Osborn D, Weber S. *MACCS Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project. NUREG/CR-7009.* Washington, DC: U.S. Nuclear Regulatory Commission; 2014. Available at: http://pbadupws.nrc.gov/docs/ML1423/ML14234A148.pdf.

[3] Office of Nuclear Regulatory Research. *Research Activities: FY2015-FY2017. NUREG-1925, Rev. 3.* Washington, DC: U.S. Nuclear Regulatory Commission; 2016. Available at: http://pbadupws.nrc.gov/docs/ML1606/ML16060A414.pdf.

[4] http://energy.sandia.gov/energy/nuclear-energy/nuclear-energy-safety-technologies/melcor/.

[5] http://maccs.sandia.gov/melmaccs.aspx.

The approach to release category binning can vary by application, but may include consideration of factors such as: (1) core damage status, (2) containment status, (3) scrubbing or filtration of the release, and (4) timing of the release. Release category binning guidance is provided in various information sources, including PSA procedures guides,[1] standards,[2] and international documents.[3]

Some particularly difficult modelling challenges or technical issues that would benefit from information exchange and sharing of best practices among the international PSA community include modelling of: (1) operator actions prompted by radiation measurements; (2) effective size and location of release(s); (3) initial conditions of releases associated with containment/confinement failure caused by hydrogen explosion; and (4) effects of offsite releases or other conditions on onsite actions.

3.4.3. What process do you use to define release fraction timing (time and duration of release) and release truncation time? Do you consider onsite severe accident mitigation actions or offsite emergency response actions in this process? If you do, how are they treated?

Releases estimated from MELCOR are usually segmented into approximately hourly plume segments to match the resolution of typical meteorological data. The MACCS code allows the analyst to model up to 200 distinct plume segments for each source term.

Development of release termination criteria is typically part of the Level 2 element. It is common practice to use release truncation times like 24, 48, or 72 hours after event initiation. Truncation times can also be anchored to the onset of fuel damage - when severe accident management guidance would typically be invoked - and can include consideration of onsite or offsite mitigation resources and reliability. Alternatively, release termination can be implemented by developing a source term that is analysed through the time of containment depressurization by either base mat melt-through or containment failure. For example, the original SOARCA analyses involved the use of technical judgment to select a truncation time of 48 hours after accident initiation; this was judged to be appropriate based on the belief that sufficient onsite or offsite resources would be mobilized to flood containment and halt accident progression within this time period.[15]

Research is ongoing to: (1) understand precedents for selecting release truncation times; (2) identify factors the analyst should consider in making this decision; and (3) how the choice impacts results for different Level 2 PSA metrics. This research suggests the analyst should consider multiple factors in selecting a release truncation time, including: (1) the modelling uncertainty associated with the severe accident analysis; (2) the availability of accepted human reliability analysis methods for the domains of interest; (3) the role and maturity of accident management at the facility being studied; (4) the expected impact of truncation time on deterministic and probabilistic results; and (5) study objectives.

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[1] Brookhaven National Laboratory. *Kalinin VVER-1000 Nuclear Power Station Unit 1 PRA: Procedures Guide for a Probabilistic Risk Assessment (English Version). NUREG/CR-6572, Rev. 1.* Washington, DC: U.S. Nuclear Regulatory Commission; 2005. Available at: http://pbadupws.nrc.gov/docs/ML0604/ML060450618.pdf.

[2] The American Society of Mechanical Engineers (ASME), American Nuclear Society (ANS). *Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs): Trial-Use Standard. ASME/ANS RA-S-1.2-2014.* New York, NY: The American Society of Mechanical Engineers; 2014.

[3] International Atomic Energy Agency. *Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants. Specific Safety Guide No. SSG-4.* Vienna, Austria: International Atomic Energy Agency; 2010. Available at: http://www-pub.iaea.org/mtcd/publications/pdf/pub1443_web.pdf.

3.5 What radionuclides do you use to characterize the offsite radiological consequences attributed to accidental releases of radiological materials? What is your basis for selecting these radionuclides?

For LWR offsite radiological consequence analysis applications, current state of practice is to use results from previous LWR radionuclide screening analyses that used the likelihood of an isotope being a significant contributor to radiological dose as the screening criterion; factors considered in these screening analyses included radionuclide inventory, half-life, chemical composition, release fraction (typically related to volatility for core-melt releases), and dose coefficients. The process for identifying radionuclides for inclusion in the analysis is described in Section 3.3 and 8.2.1 of Appendix VI of WASH-1400.[5]

A set of 54 radionuclides was used to assess offsite radiological consequences using the Calculation of Reactor Accident Consequences (CRAC) code for WASH-1400.[5] The default input files for the MACCS code include data for a set of 60 radionuclides - the original 54 considered in WASH-1400 and 6 additional radionuclides (Strontium-92, Yttrium-92, Yttrium-93, Barium-139, Lanthanum-141, and Lanthanum-142). In addition to these 60 radionuclides, the MACCS input file data implicitly include 11 daughter products by summing the dose conversion factors (DCFs) for both parent and daughter radionuclides to allow for consideration of the contribution to radiological doses from progeny.

Based on previous studies, it is common practice to present results for Iodine-131 and Cesium-137 release fractions to further characterize the short-term and long-term offsite radiological consequences, respectively.

3.6 Do you consider releases from multiple units or radiological sources co-located at the same site? If you do, how are these treated?

The current state-of-practice in the U.S. is to perform offsite radiological consequence analyses for releases from single reactor units or radiological sources - even if they are co-located with multiple radiological sources at a shared site.

However, the MACCS code was recently enhanced to include the capability to model releases from multiple, co-located radiological sources with potentially different accident progression timelines. This capability was introduced with the release of version 3.10. In practice, the analyst implements the new multi-source model by specifying multiple MELMACCS-generated source term input files to be combined, as well as the timing offset for each of the source terms.

As part of its ongoing Full-Scope Integrated Site Level 3 PRA Project,[3,4] the USNRC is exploring the use of this added capability to assess the contribution to total site risk from accidental releases involving different combinations of two or more major radiological source at the modelled site.

3.7 Do you perform sensitivity or uncertainty analyses on release categories or release fractions? If you do, what methods are used and what is the basis for using them?

Although more formal approaches are available (e.g., phenomena identification and ranking technique (PIRT), global sensitivity analysis), sensitivity analyses are typically performed with variations in source term parameters identified by professional judgment to develop a set of alternative source terms. For example, alternative source terms reflecting different levels of containment filter effectiveness were used in sensitivity analysis cases for the analyses supporting

an evaluation of containment protection and release reduction for Mark I and Mark II boiling water reactor rulemaking activities.[1]

As described in the response to item 3.3, uncertainty analyses on source terms for specific accident scenarios have been performed as part of follow-on studies for the SOARCA Project. In these analyses, Monte Carlo sampling methods are used to generate multiple MELCOR outputs that are then coupled with sampled MACCS input parameters to evaluate the distribution of accident consequences arising from uncertainties in selected MELCOR and MACCS input parameters.

Meteorological Data

3.8 How do you account for temporal and spatial variability in meteorological conditions? If you use a subset (sampling) of available meteorological data to serve as a representative data set, what criteria do you use for selecting these representative data?

Temporal variability in meteorological conditions is typically addressed by sampling from one year of hourly weather data. The base year is selected by the analyst based on professional judgment and typically is intended to reflect a representative year rather than a year with extreme weather conditions. The sampling scheme employed is a form of importance sampling that assigns each hourly weather sequence to a meteorological bin based on the wind speed, stability class, magnitude of precipitation, and onset of precipitation. Sufficient samples are drawn from each bin to ensure that lower frequency, higher consequence meteorological conditions (i.e., stable low wind speed conditions or periods of precipitation) are adequately represented in the overall sample.

Spatial variability in meteorological conditions is typically addressed by specifying a limiting spatial interval for the use of recorded meteorological data. All spatial intervals beyond this interval will use analyst-specified boundary weather conditions. These boundary weather conditions will also be applied if the duration of recorded meteorological data is not enough to transport the last plume through the limiting spatial interval for measured weather conditions.

3.9 Do you perform sensitivity or uncertainty analyses on meteorological conditions? If you do, what methods are used and what is the basis for using them?

Sensitivity analyses are sometimes performed with alternate years of weather data or by varying the number of samples drawn for the selected weather year. As described in the response to item 3.8, the analyst uses professional judgment to select one year of hourly weather data to serve as a representative year. Sensitivity analyses may therefore be used to evaluate the impact on results of using alternative years of hourly weather data.

Environmental Transport and Dispersion

3.10 What atmospheric transport and dispersion (ATD) model(s) do you use? What process and criteria do you use to select: (1) an ATD model; and (2) a time scale for updating calculations to account for time dependence?

The MACCS code uses a Gaussian straight-line trajectory plume segment model that can handle changes in wind speed, wind direction, stability, and precipitation, but not changes in plume trajectory. Earlier analyses typically employed a limited number of plume segments that could represent multiple hours of releases, but more recent analyses include plume segments of

[1] Beall, R H. *Evaluation of the Containment Protection & Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities (10 CFR Part 50) (RIN-3150-AJ26). SECY-15-0085*. Washington, DC: U.S. Nuclear Regulatory Commission; 2015. Available at: http://pbadupws.nrc.gov/docs/ML1502/ML15022A218.html.

approximately hourly duration to match the hourly resolution of the underlying meteorological data.

A new particle tracking ATD model based on the HYbrid Single Particle Lagrangian Integrated Trajectory (HYSPLIT) model[1] is under development for addition to MACCS and will be released in a future version. This particle tracking ATD model is being developed to provide an alternative ATD model that addresses known limitations of the Gaussian straight-line trajectory plume segment model.

As more ATD model options become available, a process and criteria will be needed to select which model(s) will be used for a particular study. Some factors to consider in making the selection may include: (1) study objectives; (2) computational efficiency; (3) site characteristics; and (4) data availability.

3.11 What process and criteria do you use to define the boundaries and intervals of the spatial grid or domain used for performing offsite consequence calculations?

In the MACCS code, the region potentially affected by an accidental release is represented with an (r, θ) polar coordinate spatial grid system centred on the location of the release. The radius r represents downwind distance. The angle θ represents the angular offset from north in the clockwise direction. The analyst specifies the number of radial divisions as well as their endpoint distances. Up to 35 divisions may be defined, extending out to a maximum distance of 9999 km. However, it is common practice to model the region surrounding a nuclear installation out to a radial distance of 1000 miles. The angular divisions used to define the spatial grid are fixed in the code, and therefore cannot be modified by the analyst. All of the calculations of MACCS are stored on the basis of this polar coordinate spatial grid.

USNRC guidelines specify that regulatory analyses should typically calculate offsite radiological consequences within 50 miles of the nuclear installation. However, other spatial intervals can be considered on a case-by-case basis.[8,9]

The USNRC safety goal policy statement specifies that risk metrics for comparison with the QHOs are to be calculated over defined spatial intervals:

- ***Average Individual Early Fatality Risk QHO*: 0-1 mile beyond the nuclear power plant site boundary.**
- ***Average Individual Latent Cancer Fatality Risk QHO*: 0-10 miles beyond the nuclear power plant site boundary.[20]**

PCAs or Level 3 PSAs performed to support research studies may calculate offsite radiological consequences over a broader range of spatial intervals. Although Level 3 PSAs performed as part of previous research studies[5,6] calculated some consequence metrics over the entire modelled spatial domain out to 1000 miles around each of the included nuclear power plant sites, current USNRC-sponsored PCAs and Level 3 PSAs typically calculate offsite radiological consequences within 50-100 miles to reflect the limitations of the Gaussian plume segment ATD model. Results may be calculated out to longer distances to determine whether the defined spatial domain of interest sufficiently captures the full range of consequences, but these longer distance results are typically not used for risk estimation or evaluation purposes.

[1] http://www.arl.noaa.gov/HYSPLIT_info.php.

3.12 With regard to the spatial modelling around a nuclear power plant for Level 3 PSA:

3.12.1. What information sources do you use to develop geographical or topographical parameters? Do you use generic or site-specific data?

The MACCS code uses site-specific meteorological data as well as site-specific population, agricultural, and economic data estimated on a polar grid, typically using the SECPOP code.[1]

3.12.2. Does the ATD response vary spatially with respect to varying topographic parameters? What is the spatial resolution of the ATD and do topographic parameters of the ATD vary within this resolution?

Because MACCS uses a simple terrain model that does not account for terrain elevation differences or topographic effects on winds, the use of topographic data is used only qualitatively to evaluate the applicability of the Gaussian plume segment model to a particular site.

3.13 Do you consider radiological releases to water sources and aqueous transport and dispersion phenomena? If you do, how are these treated? If you have previously considered modelling radiological releases to water sources and aqueous transport and dispersion phenomena, but decided not to do so, what was the basis for your decision?

The MACCS code includes a simplified model for deposition of atmospheric radioactivity onto water bodies and runoff of deposited material on land into water bodies to support estimation of the contribution to radiological doses from ingestion of contaminated water.[2]

However, aqueous releases directly to water bodies (e.g., runoff, pipe discharges, or discharges to aquifers from base mat melt-through) are typically not addressed. There a two principal justifications for excluding aqueous releases: (1) the airborne pathway is expected to be dominant for health risks because movement of radionuclides to the accessible environment is expected to be slow relative to atmospheric transport; and (2) releases to groundwater or surface water are considered to be easier to interdict.[3]

After the 2011 Fukushima nuclear accident, the USNRC initiated a research project to assess potential offsite consequences of losing control of highly contaminated water in a severe accident in which the contaminated water flows to a body of water. The final project report presents state-of-the-art hydrologic transport modelling results for leakage of waterborne radionuclides directly into a freshwater body. The study postulated releases to three types of freshwater settings: (1) a large river, (2) a small river, and (3) a small lake. Two- and three-dimensional modelling was used to explore the concentrations of radionuclides as they are transported through the three freshwater bodies. The approach determined how advection, dilution/dispersion, radioactive decay, and adsorption/desorption processes affect concentrations

[1] http://maccs.sandia.gov/secpop.aspx.

[2] Helton JC, Muller AB, Bayer A. Contamination of Surface-Water Bodies after Reactor Accidents by the Erosion of Atmospherically Deposited Radionuclides. *Health Physics*; 1985; 48(6): 757-771.

[3] Niemczyk SJ, Adams KG, Murfin WB, Ritchie LT, Eppel EW, Johnson JD. *The Consequences from Liquid Pathways after a Reactor Meltdown Accident. NUREG/CR-1596.* Washington, DC: U.S. Nuclear Regulatory Commission; 1981. Available at: http://www.iaea.org/inis/collection/NCLCollectionStore/_Public/35/047/35047210.pdf.

of each of the transported radionuclides under the hypothetical conditions in each freshwater setting.[1]

Protective Action (Countermeasure) Modelling

3.14 How do you define, and what protective actions (countermeasures) do you model for each of the nuclear or radiological incident or accident response phases listed below?

3.14.1. Early/Emergency Phase:
3.14.2. Intermediate Phase:
3.14.3. Late/Recovery Phase:

USEPA guidance[25,26] uses the following definitions for each radiological incident phase:

(1) *Early Phase*: The beginning of a radiological incident when immediate decisions for effective use of protective actions are required and must therefore be based primarily on the status of the radiological incident and the prognosis for worsening conditions. This phase may last from hours to days.
(2) *Intermediate Phase*: The period beginning after the source and releases have been brought under control (has not necessarily stopped but is no longer growing) and reliable environmental measurements are available for use as a basis for decisions on protective actions and extending until these additional protective actions are no longer needed. This phase may overlap the early phase and late phase and may last from weeks to months.
(3) *Late Phase*: The period beginning when recovery actions designed to reduce radiation levels in the environment to acceptable levels are commenced and ending when all recovery actions have been completed. This phase may extend from months to years.

The MACCS code utilizes a similar framework. The early phase is modelled using the EARLY module, and the intermediate and late phases are modelled using the CHRONC module. However, MACCS imposes constraints that prohibit the overlap of phases in consequence models; each phase must begin at the end of the preceding phase.

Early phase protective actions modelled in the MACCS EARLY module include: sheltering, evacuation, and dose-dependent relocation. MACCS also includes a capability to implement a potassium iodide (KI) model that accounts for use of KI as a supplementary protective action to reduce the radiological dose to the thyroid gland by blocking the uptake of radioiodine. Intermediate phase protective actions modelled in the MACCS CHRONC module include: dose-dependent relocation or interdiction and dose-dependent bans on agricultural products. Late phase protective actions or recovery actions modelled in the MACCS CHRONC module include: dose-dependent relocation or interdiction (whether temporary or permanent - which is referred to as condemnation), decontamination, and dose-dependent bans on agricultural products or condemnation of farmland.

3.15 Do you model population groups with different protective action (countermeasure) behaviours? If you do, how do you define them? To what extent do you account for population density or age and gender distribution effects?

Multiple cohorts are typically used to model population groups with different protective action behaviours. The MACCS code allows the analyst to define up to twenty cohorts. At a minimum, two cohorts are used to model and estimate offsite consequences for two distinct groups: (1) an

[1] Yabusaki SB, Napier BA, Perkins WA, Richmond MC, Rakowski CL, Snyder SF et al. *Modeling of Radionuclide Transport in Freshwater Systems Associated with Nuclear Power Plants. NUREG/CR-7231*. Washington, DC: U.S. Nuclear Regulatory Commission; 2017. Available at: https://www.nrc.gov/docs/ML1711/ML17111A578.pdf.

evacuating cohort that models the fraction of the offsite population that begins evacuation when a general emergency is declared at the nuclear installation; and (2) a non-evacuating cohort that models the fraction of the offsite population that is assumed to maintain normal activity unless their projected dose is predicted to exceed USEPA PAG levels.

A number of factors are typically used to select and define cohorts, including special populations. Examples of these factors include: (1) location or spatial interval around the nuclear installation; (2) delay times for implementing protective actions; (3) evacuation speeds; and (4) exposure factors and shielding parameters. Age- and gender-specific variability in protective action behaviours are generally not considered; however, school populations may be assigned to their own cohort if they constitute a distinct population within or near the EPZ.

3.16 Do you use probabilistic models of protective action (countermeasure) behaviours that model the probabilities of success or failure for protective actions (countermeasures)? If you do, please describe the models and their bases.

As stated in the response to item 3.15, the current state-of-practice is to specify multiple cohorts to model population groups with different protective action behaviours, including those that take a longer period of time to evacuate or that choose to not evacuate at all. For each cohort, protective actions are modelled using deterministic models that implicitly assume protective actions (if taken) are 100% effective for the entire cohort and result in specified exposure and shielding factors for each exposure pathway. Although the MACCS code has the capability to specify probability distributions for each of the parameters used to define these cohort-specific protective action models, these do not provide a quantitative measure of the probability of success or failure of protective actions.

3.17 What information sources do you use as a technical basis for protective action (countermeasure) modelling? Do you use generic or site-specific data?

Parameters for modelling protective actions are typically based on site-specific data. For example, parameters for evacuation modelling are typically derived from information in site-specific evacuation time estimate (ETE) analyses. U.S. regulations require that an analysis of the time required to evacuate be provided for various sectors and distances within the plume exposure pathway EPZ for transient and permanent residents.[1] This ETE analysis includes consideration of multiple factors, including: (1) evacuation demand estimation for multiple population groups (e.g., permanent residents, transient populations, transit-dependent populations, special facility residents, and schools); (2) roadway characteristics, capacity analysis, and traffic control; and (3) alternative scenarios comprised of different combinations of season, day of the week, time of day, weather conditions, special events, roadway impact, or other circumstances that should be assessed. Multiple scenarios are evaluated to ensure ETE results encompass a reasonable range of potential evacuation situations for the specific site. Criteria for development of ETE studies are available for further information.[2]

U.S. regulations require nuclear power plant licensees to update their ETE analyses on a periodic basis; in practice, these analyses are updated approximately every 10 years to coincide with the release of decennial population census data from the U.S. Census Bureau.[24,54]

[1] US Nuclear Regulatory Commission. *Title 10 Code of Federal Regulations Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities"* Available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-appe.html.

[2] Jones J, Walton F, Wolshon B, Laughlin J. *Criteria for Development of Evacuation Time Estimate Studies. NUREG/CR-7002.* Washington, DC: U.S. Nuclear Regulatory Commission; 2011. Available at: http://www.nrc.gov/docs/ML1130/ML113010515.pdf.

Radiological Exposure and Dose Assessment

3.18 How do you define potentially exposed populations or cohorts? Do you consider onsite (e.g., worker) populations or potentially sensitive, vulnerable, or critical groups? If you do, what is your basis for including these groups, and how are they modelled?

In general, PCAs or Level 3 PSAs are performed to assess the offsite public consequences or risks attributable to a spectrum of possible accident scenarios involving a nuclear installation. Therefore, onsite populations are generally not considered.

As described in the response to item 3.15, cohorts are generally selected and defined based on a number of factors. Although sensitive, vulnerable, or critical groups are generally not differentiated from members of the general public within MACCS, such populations may be modelled as distinct cohorts if they have unique exposure factors or shielding parameters.

3.19 What exposure pathways do you model? What is your basis for selecting these pathways? What exposure duration is assumed in your models?

The MACCS code models the following exposure pathways identified by MACCS module and the corresponding radiological incident phases:

(1) *EARLY Module (Early Phase)*: (1) direct exposure to external radiation from the plume of released radiological materials (cloud shine); (2) inhalation of radioactivity in the plume; (3) direct exposure to external radiation from ground contamination (ground shine); (4) inhalation of resuspended radioactivity; and (5) contamination of skin and clothing.
(2) *CHRONC Module (Intermediate and Late Phases)*: (1) ground shine; (2) inhalation of resuspended radioactivity; and (3) ingestion of contaminated food and water.

Exposure durations are typically based on site-specific ETE analyses and protective action models for the early phase, while a lifetime (50-year) exposure duration is typically assumed for the late phase. Within MACCS, the analyst specifies the duration of the intermediate phase, which can range from 0 to 365 days. Recent PCA studies[15,16] have used an intermediate phase duration of 0 days based on professional judgment. Sensitivity analyses can be performed to evaluate the impact of using alternative intermediate phase durations.

3.20 What information sources do you use to develop exposure and dose conversion factors? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

Exposure factors such as inhalation and intake rates and dose conversion factor (DCFs) are based on generic average values obtained from USEPA guidance documents. DCFs express the relationship between: (1) environmental concentrations or intakes; and (2) resultant human doses or dose rates. They are developed for specific exposure pathways, organs, and radionuclides. The MACCS code requires DCFs for the inhalation, ingestion, cloud shine, and ground shine exposure pathways for each organ for which doses are to be calculated and for each radionuclide included in the analysis. DCF files that were created based on USEPA guidance and dosimetric models[1] are typically used as input to MACCS.

[1] Eckerman KF, Leggett RW, Nelson CB, Puskin JS, Richardson ACB. *Federal Guidance Report No. 13: Cancer Risk Coefficients for Environmental Exposure to Radionuclides. EPA 402-R-99-001.* Washington, DC: U.S. Environmental Protection Agency; 1999. Available at: https://www.epa.gov/sites/production/files/2015-05/documents/402-r-99-001.pdf.

Radiological Health Effects

3.21 Do you model and estimate radiological health effects in your offsite radiological consequence analyses? If you do, please respond to questions 3.22 through 3.24. If you do not, please describe any other methods you use to address the offsite radiological health effects attributable to possible accident scenarios involving nuclear installations and proceed to question 3.25.

Early (acute) fatalities and injuries are typically estimated using deterministic health effects models. Latent (delayed/chronic) fatalities and injuries are typically estimated using stochastic health effects models.

3.22 What types of early and latent health effects (including fatal and non-fatal effects) do you model? What target organs do you use for the associated health effects models?

Radiological health effects models have been developed for multiple types of fatalities and injuries for use in offsite radiological consequence analyses.[1] The health effects evaluated in a specific analysis are typically driven by the needs of the application. Metrics used to support various applications are described in responses to other items. Notable examples include the metrics used to compare results with the safety goal QHOs for average individual risk of early and latent cancer fatalities.

Although early injuries and non-fatal latent cancers are typically modelled and estimated in PCAs or Level 3 PSAs, these results are generally used to provide additional insights and perspective; they are not typically used for risk estimation and evaluation purposes.

The types of early and latent radiological health effects and target organs that are typically modelled and estimated using the MACCS code are summarized below in Table 4.

Table 4. Early and Latent Radiological Health Effects and Target Organs in MACCS

Category	Type	Health Effect	Target Organ[a,b]
Early	Fatality	Hematopoietic Syndrome	Red Bone Marrow
		Gastrointestinal Syndrome	Stomach
		Pulmonary Syndrome	Lungs
		Total Early Fatalities	Not Applicable
	Injury	Prodromal Vomiting	Stomach
		Diarrhea	
		Skin Erythema	Skin
		Transepidermal Injury	
		Thyroiditis	Thyroid Gland
		Hypothyroidism	
		Pneumonitis	Lungs
		Total Early Injuries	Not Applicable
Latent	Fatality or Injury	Leukemia	Red Bone Marrow
		Bone Cancer	Bone
		Breast Cancer	Breasts
		Lung Cancer	Lungs
		Thyroid Cancer	Thyroid Gland
		Liver Cancer	Liver
		Colon Cancer	Lower Large Intestine
		Residual Cancers	Bladder Wall
		Total Cancer Fatalities or Injuries	Not Applicable

[a] Target organs available for specifying early health effects models in MACCS include: (1) lower large intestine, (2) lungs, (3) red bone marrow, (4) skin, (5) stomach, and (6) thyroid gland.

[b] Target organs available for specifying latent health effects models in MACCS include: (1) bladder wall, (2) bone, (3) breasts, (4) liver, (5) lower large intestine, (6) lungs, (7) red bone marrow, (8) thyroid gland, and (9) whole-body lifetime effective dose.

[1] Evans JS, Abrahamson S, Bender MA, Boecker BB, Gilbert ES, Scott BR, Yaniv SS. *Health Effects Models for Nuclear Power Plant Accident Consequence Analysis. Part I: Introduction, Integration, and Summary. NUREG/CR-4214, Rev. 2, Part I.* Washington, DC: U.S. Nuclear Regulatory Commission; 1993. Available at: http://pbadupws.nrc.gov/docs/ML0500/ML050030192.pdf.

3.23 What dose-response model(s) do you use to estimate the numbers of latent health effects attributable to radiological doses caused by accidental releases from nuclear installations, and what is the basis for their use? What (if any) methods do you use to account for uncertainty about the true dose-response relationship for exposures to low levels of ionizing radiation?

In previous Level 3 PSAs performed as part of research studies,[5,6] linear no-threshold (LNT) or linear-quadratic no-threshold dose-response models were used to quantify total and individual latent cancer fatality risk. However, in recent USNRC-sponsored PCAs,[15,16] sensitivity analysis has been used to evaluate the effects of using plausible alternative dose-response models for quantifying latent health effects. In these PCAs, the default LNT model was used as the base case, and two models involving dose truncation levels were used as sensitivity cases. In the first model, the dose truncation was set at 6.2 mSv/year¾a level that corresponds to the background radiation exposure an average individual in the U.S. receives in a given year. In the second model, the dose truncation was set at 50 mSv/year with a 100 mSv lifetime limit¾a level below which the Health Physics Society recommends against quantitative estimation of health risks.[1]

Supplementary Information

The dose-response models used for quantifying the lifetime cancer risk attributable to possible nuclear accidents have been based largely on associations observed in epidemiology studies of well-defined populations exposed to radiation doses that were orders of magnitude greater than what the majority of the affected public would be exposed to following a nuclear accident. Since high-quality dose-response data for the general population are not available for the range of radiation exposures relevant to nuclear accidents, there is considerable uncertainty about the nature of the true dose-response relationship for these exposures; extrapolation from the high-dose range of epidemiologic data to the low-dose range of interest can therefore be performed using a number of plausible alternative mathematical models that provide a reasonable fit to the observed data.

For decades, scientific advisory and governmental bodies have repeatedly endorsed the default use of the LNT dose-response model for radiation protection applications - where precaution and conservatism have been judged to be prudent. Although prevailing knowledge of biological mechanisms and available epidemiologic evidence continue to support the LNT hypothesis, emerging evidence from *in vitro* radiation biology studies has spurred heated debate within the scientific community about the validity of the LNT model. Moreover, due in part to previously mentioned scientific limitations, available epidemiologic studies have not demonstrated the occurrence of radiation-induced cancers among populations exposed to either very low levels of radiation or to radiation delivered over long periods of time. For these reasons, scientific advisory bodies and professional organizations now caution against using the LNT model in risk projections to estimate the number of cancer deaths resulting from low-level radiation exposures to large affected populations in risk assessment applications; instead, they now recommend not quantifying the lifetime cancer risk for radiation exposures below specified levels. In practice, this approach is implemented using a truncation dose-response model. In this type of model, a dose truncation or cut off level is specified, below which lifetime cancer deaths attributable to accidental releases are not quantified.

3.24 What information sources do you use to develop input parameters for the health effect models? Do you use generic or site-specific data? Do you use average or age- and gender-specific data?

Generic average values from multiple information sources are typically used for specifying health effects model parameters. Parameter values for deterministic models used to estimate

[1] Health Physics Society. *Radiation Risk in Perspective: Position Statement of the Health Physics Society.* 2010;(January 1996). Available at: http://hps.org/documents/risk_ps010-2.pdf.

early health effects are typically based on information from previous studies sponsored by the USNRC[57] and by the Commission of the European Communities.[1] Parameter values for stochastic models used to estimate latent health effects are typically based on information contained in USEPA guidance[56] and U.S. National Academy of Sciences reports.[2]

Economic Consequences

3.25 Do you model and estimate economic consequences in your offsite radiological consequence analyses? If you do, please respond to questions 3.26 and 3.27. If you do not, please describe any other methods you use to address the offsite economic consequences attributable to possible accident scenarios involving nuclear installations and proceed to question 3.28.

Economic consequences or offsite property damages are typically estimated to provide input to cost-benefit analyses performed as part of regulatory, back fit, or environmental analyses. Economic consequences and other socio-economic indicators (e.g., the extent of land or population impacted by protective actions) are sometimes estimated in research studies to provide additional insights that can be used to either: (1) check for consistency with other correlated results; or (2) evaluate results from a different perspective.

3.26 With regard to potential economic consequences considered within a Level 3 PSA:

3.26.1. What types of economic consequences do you model? What process do you use to select cost categories and models for estimating economic consequences?

The MACCS code includes economic models for estimating the costs attributed to implementation of modelled protective actions to reduce radiological dose to the offsite public from the accidental release of radiological materials. These costs include: (1) daily costs of compensation for populations subject to evacuation or short-term relocation arising from food, housing, transportation, lost income, or replacement of lost personal property; (2) costs of long-term relocation of populations and businesses in interdicted land areas; (3) depreciation costs that account for loss of value of interdicted property; (4) decontamination costs; and (5) costs arising from implementation of agricultural countermeasures.

MACCS does not include models for estimating some cost categories that could be important for nuclear accident scenarios. Examples of these cost categories include: (1) costs associated with the number of radiation-induced injuries or fatalities; (2) replacement power costs; and (3) costs associated with storage or disposal of contaminated material following decontamination efforts.

An alternative economic model that can estimate national and regional gross domestic product (GDP) losses using regional input-output techniques is under development for addition to MACCS and will be released in a future version.

[1] Haskin FE, Kraan BCP, Harper FT, Grupa JB, Goossens LHJ, Randall J. *Probabilistic Accident Consequence Uncertainty Analysis: Early Health Effects Uncertainty Assessment. NUREG/CR-6545.* Washington, DC: U.S. Nuclear Regulatory Commission; 1997.

[2] Committee on the Biological Effects of Ionizing Radiation (BEIR V); Commission on Life Sciences; Division on Earth and Life Studies; National Research Council. *Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V.* Washington, DC: National Academies Press; 1990. Available at: http://www.nap.edu/catalog/1224/health-effects-of-exposure-to-low-levels-of-ionizing-radiation.

3.26.2. If you translate radiological health effects into economic consequences, please describe your process for doing so.

As described in the response to item 2.2.2, to generate commensurable values for use in cost-benefit analyses, benefits in terms of averted radiological health effects may be monetized using a conversion factor applied to the averted population dose within a prescribed distance (typically 50 miles) from the nuclear installation. The USNRC is currently revising the value of this dollar per person-rem conversion factor and is proposing a process for routine updates to ensure its value is maintained current.[21]

3.26.3. If you model land contamination, please describe your process for estimating the health, environmental, or economic consequences attributed to land contamination.

The health and economic consequences attributed to land contamination are captured in the dose and economic models, but there is no additional consideration of environmental consequences beyond the previously described dose and cost models.

3.27 What information sources do you use to develop cost parameters for economic consequence models? Do you use generic, region-specific, or site-specific data?

There are a variety of inputs to the economic consequence models. Site-specific data pertaining to populations, property values, personal incomes, agricultural sales, and land fraction values come from census data provided by the U.S. Census Bureau and from the U.S. Department of Agriculture.

Generic inflation data used to escalate cost values to common years are derived from the U.S. Bureau of Labour Statistics Consumer Price Index for All Urban Consumers (CPI-U) database. Generic decontamination plan data, depreciation rates, and expected rates of return from property come from literature reviews and updates to previous Level 3 PSAs performed as part of research studies.[5,6]

Consequence Quantification and Reporting

3.28 What process and criteria do you use for selecting consequence metrics for quantification and reporting, including spatial intervals or distances from release points?

There is no specified process or criteria for selecting consequence metrics for quantification and reporting. Factors that are typically considered in selecting consequence metrics and spatial intervals can include: (1) analysis objectives - which are typically linked to application-specific requirements or questions to be addressed; (2) stakeholder interests; (3) standards requirements or state-of-practice; (4) potential future uses of results and insights; (5) schedule and resource constraints; and (6) capabilities and limitations of models and analytical tools. Metrics used to support various applications are described in responses to other items.

Notable examples of consequence metrics that can be quantified and reported using the MACCS code are summarized below in Table 5. Although MACCS enables the analyst to quantify and report a large number of consequence metrics, schedule and resource constraints may impose practical limits on what is quantified and reported for a particular study.

Table 5. Notable Consequence Metrics Available for Quantification and Reporting Using MACCS[1]

MACCS Module	Consequence Metric	Definition
EARLY	Health Effect Cases	Number of cases of specified health effects predicted to occur within specified spatial interval(s).
	Early-Fatality Radius	The greatest distance at which a specified level of early fatality risk is exceeded. Can be used to provide information about the size of the region in which early fatalities are predicted to occur by setting the threshold parameter value to 0.
	Population Exceeding Threshold	Number of people predicted to receive early phase doses to specified target organ(s) that exceed specified threshold level(s).
	Average Individual Risk	Average individual risk of specified health effect(s) for individuals within specified spatial interval(s). Calculated by summing health effect cases predicted to occur in all sectors at specified spatial interval(s) and dividing by the number of sectors.
	Population Dose	The total population dose to specified target organ(s) from modeled exposure pathways within specified spatial intervals.
	Population-Weighted Risk	Average individual risk of specified health effect(s) for individuals within specified spatial interval(s). Calculated by summing health effect cases predicted to occur within specified spatial interval(s) and dividing by the total population within the specified interval(s).
	Land Area Exceeding Dose	Sizes of contaminated land area(s) in which dose(s) to specified target organ(s) exceed specified threshold levels.
	Land Area Exceeding Concentration	Sizes of contaminated land area(s) in which concentration(s) of specified radionuclide(s) exceed specified threshold levels.
	Population Movement	Fraction of population crossing outer boundary of specified spatial interval(s) over specified time interval(s).
CHRONC	Population Dose	The total population dose to specified target organ(s) from modeled exposure pathways in specified spatial intervals.
	Economic Cost	Economic costs associated with implementing protective actions within specified spatial interval(s).
	Impacted Area or Population	Sizes of area(s) or number of people within specified spatial interval(s) that are impacted by interdiction, decontamination, condemnation, or disposal of agricultural products.
	Impacted Population	Number of people within specified spatial interval(s) that are impacted by evacuation or relocation.

Risk Characterization

3.29 What process and criteria do you use for selecting risk metrics for effectively communicating Level 3 PSA results, including metrics designed to measure the effects of radiological releases involving multiple units or radiological sources co-located at the same site, if applicable?

As with consequence metrics, there is no specified process or criteria for selecting risk metrics for quantification and reporting. Factors that are typically considered in selecting risk metrics can include: (1) analysis objectives - which are typically linked to application-specific requirements or questions to be addressed; (2) stakeholder interests; (3) standards requirements or state-of-practice; (4) potential future uses of results and insights; (5) schedule and resource constraints;

[1] This table does not include a comprehensive listing of consequence metrics that are available for quantification and reporting in the MACCS code.

and (6) capabilities and limitations of models and analytical tools. Metrics used to support various applications are described in responses to other items.

As part of its ongoing Full-Scope Integrated Site Level 3 PRA project, the USNRC is exploring candidate integrated site risk metrics that can be used to measure the effects of radiological releases involving multiple radiological sources co-located at the same site. In principle, the risk metrics used to estimate the frequencies of offsite public health, economic, and environmental consequences for accidental releases from single radiological sources can be adapted by adjusting the frequency basis to estimate the same quantities for accidental releases from multiple radiological sources. For example, instead of quantifying risk metrics on a per-reactor-year basis, they can be quantified on a per-site-year or per-calendar-year basis.

3.30 For each of the groups listed below, what methods do you use for presenting and communicating risk results and the uncertainty in risk results? How do you present the results from low-probability/high-consequence events in an understandable context for each group? Who is responsible for communicating the results to each group?

(1)

(2) Decision makers:

(3) General Public:

In general, risk results that characterize variability or aleatory uncertainty arising from inherent randomness or stochastic processes are presented using a variety of formats and graphical displays. Notable examples include: (1) point estimates - especially expected (mean) risk of selected consequence metrics over all-weather trials - which are typically used to inform regulatory decision making; and (2) complementary cumulative distribution function (CCDF) curves (also termed *exceedance frequency curves* or *risk curves*) - which represent the frequencies of exceeding different consequence levels over all modelled radiological release categories and weather trials.

Epistemic uncertainty in risk results arising from imperfect knowledge is typically characterized using random or pseudo-random sampling techniques (e.g., simple Monte Carlo sampling or Latin Hypercube Sampling) to develop empirical probability distributions that characterize the uncertainty in risk results. Notable examples of graphical displays used to illustrate this uncertainty include: (1) empirical probability distributions for selected risk metrics - typically in the form of probability density function (PDF) curves and/or cumulative distribution function (CDF) curves; (2) box plots that illustrate the locations of key summary statistics (e.g., mean value, 50th percentile (median value), 95th percentile, and 5th percentile) for selected risk metrics; and (3) sets of CCDFs (also termed *families of risk curves*) - with the different curves representing different probabilities of frequencies of exceeding different consequence levels.

In general, the USNRC technical staff and managers directly involved with a particular analysis are responsible for communicating results to decision makers and to the general public. However, the USNRC has an Office of Public Affairs with communications professional who can provide advice and consultation to USNRC staff on risk communication efforts. In addition, the USNRC has developed guidance documents that aim to enhance risk communication with both internal stakeholders (including decision makers)[1] and external stakeholders (including the general public).[2]

3.31 With regard to the treatment of uncertainties:

[1] Szabo A, Persensky J, Peterson L, Specht E, Goodman N, Black R. *Effective Risk Communication: The Nuclear Regulatory Commission's Guidelines for Internal Risk Communication. NUREG/BR-0318.* Washington, DC: U.S. Nuclear Regulatory Commission; 2004. Available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0318/.

[2] Persensky J, Browde S, Szabo A, Peterson L, Specht E, Wight E. *Effective Risk Communication: The Nuclear Regulatory Commission's Guidelines for External Risk Communication. NUREG/BR-0308.* Washington, DC: U.S. Nuclear Regulatory Commission; 2004. Available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0308/.

3.31.1. Do you perform sensitivity or uncertainty analyses as part of risk estimation? If you do, what process and methods do you use?

Sensitivity analyses are frequently performed for regulatory analyses to evaluate the sensitivity of net value estimates to changes in cost-benefit analysis inputs. For Level 3 PSAs, sensitivity analyses can be performed to: (1) identify important parameters for treatment in an uncertainty analysis; or (2) assess the impacts of uncertainties related to the existence of plausible alternative models of logic structures and physical phenomena.

Simplified uncertainty analyses using Monte Carlo sampling methods have been performed for some regulatory analyses.[1] In addition, Level 3 PSAs performed as part of research studies[5,6] have included uncertainty analyses. Current state-of-practice is to perform integrated uncertainty analyses using Monte Carlo sampling methods by specifying probability distributions for selected parameters in the Level 1 and Level 2 PSA models; parametric uncertainty analyses are not typically performed for parameters used to specify the Level 3 offsite PCA model.

3.31.2. How do you determine which parameters will be varied or will have uncertainty distributions specified for propagating uncertainty? What parameters do you evaluate?

Professional judgment informed by reviews of previous sensitivity or uncertainty analyses is typically used to select parameters for variation or sampling.

3.31.3. To what extent do you consider the effect of correlation on parameters?

If performed, the extent to which state of knowledge correlation (SOKC) between parameters is accounted for in performing a parametric uncertainty analysis is driven by the needs of a particular PSA application with respect to the degree of site-specificity and model realism.

Other

3.32 Are there any other questions that you believe should have been asked in this survey? Is there any other information or are there any other technical challenges or notable practices you would like to share with the international community?

Two additional questions could have been asked in this survey to add value to this WGRISK activity:

(1) Do you include consideration of non-radiological impacts (e.g., injuries or fatalities caused by taking protective actions to avoid or reduce radiological dose, psychological or mental health impacts, or social disruption) in performing Level 3 PSA? If you do, what types of non-radiological impacts do you consider? What models and sources of information do you use to characterize these impacts? If you do not, what is your basis for excluding non-radiological impacts from these analyses?

As described in the response to item 2.5, the USNRC safety goal policy statement broadly defines an acceptable level of *radiological* risk to public health and safety from potential releases of

[1] Skeen DL. *Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments. SECY-12-0157.* Washington, DC: U.S. Nuclear Regulatory Commission; 2012. Available at: http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2012/2012-0157scy.pdf.

radiological materials from commercial nuclear power plants. The safety goal QHOs for average individual risk of early fatality and latent cancer fatality relate only to the risk of radiological health effects caused by exposure to ionizing radiation from accidental radiological releases. Therefore, in making determinations about whether a proposed generic safety enhancement or back fit provides a substantial enough increase in public health and safety to warrant a detailed evaluation to determine whether it is cost-beneficial, the USNRC focuses on measures of radiological risk or surrogate measures related to CDF or LERF. However, non-radiological impacts could be considered in evaluating the benefits and costs of proposed actions, and therefore could be considered in performing a Level 3 PSA.

The MACCS code is limited in its capability to quantify non-radiological impacts. In particular, as shown in Table 5, the code is able to quantify and report the numbers of people affected by various protective actions; these results can be used to provide additional perspective about the societal impacts of accidental releases. The code does not include models for quantifying other types of non-radiological impacts.

(2) Results from recent studies[15,16] suggest that latent cancer fatality risk is sensitive to assumptions about long-term habitability dose criterion that are used to determine when relocated populations are allowed to return home following a modelled accidental release. What long-term habitability dose criterion do you use in your Level 3 PSAs and what is the basis for this value?

The USEPA intermediate phase PAG level for relocation of the public is 2 rem (20 mSv) projected dose during the first year and 0.5 rem (5 mSv) per year projected dose during subsequent years.[25,26] While many states in the U.S. have adopted this as the long-term habitability dose criterion, some have established more restrictive requirements. In performing a Level 3 PSA or PCA, it is common practice to use the site-specific value for the long-term habitability dose criterion for the base case analysis. Sensitivity analyses may then be performed to evaluate the impact on results of using alternative values.