ASAMPSA_E SEVENTH FRAMEWORK PROGRAMME	Advanced Safety Assessment Methodologies: extended PSA	EURATOM

"NUCLEAR FISSION"

Safety of Existing Nuclear Installations

Contract 605001

Risk Metrics and Measures for an Extended PSA

Reference ASAMPSA_E

Technical report ASAMPSA_E / WP30 / D30.7 / 2017-31 volume 3

Reference IRSN PSN/RES/SAG/2017-0018

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Period covered: from 01/01/2015 to 31/12/2016	Actual submission date: 31-12-2016	
Start date of ASAMPSA_E: 01/07/2013	Duration:42 months	
WP No: 30	Lead topical coordinator : A. Wielenberg	His organization name : GRS

Project co-funded by the European Commission Within the Seventh Framework Programme (2013-2016)		
Dissemination Level		
PU	Public	Yes
RE	Restricted to a group specified by the partners of the ASAMPSA_E project	No
CO	Confidential, only for partners of the ASAMPSA_E project	No

ASAMPSA Quality Assurance page

Partners responsible of the document :	GRS, AREXIS, IRSN
Nature of document	Technical Report
Reference(s)	Technical report ASAMPSA_E/ WP 30 / D30.7 / 2017-31 volume 3 Rapport IRSN-PSN-RES/ SAG/2017-0018
Title	Risk Metrics for an Extended PSA
Author(s)	A. Wielenberg (GRS), C. Hasnaoui (Arexis), L. Burgazzi (ENEA), E. Cazzoli (CCA), P. Jan (EDF), H. Löffler (GRS), T. Siklóssy (NUBIKI), J. Vitazkova (CCA), E. Raimond (IRSN)
Delivery date	31/12/2016
Topical area	Risk Measures, Risk Metrics
For Journal & Conf. papers	No

Summary :

This report provides a review of the main used risk measures for Level 1 and Level 2 PSA. It depicts their advantages, limitations and disadvantages and develops some more precise risk measures relevant for extended PSAs and helpful for decision-making. This report does not recommend or suggest any quantitative value for the risk measures. It does not discuss in details decision-making based on PSA results neither.

The choice of one appropriate risk measure or a set of risk measures depends on the decision making approach as well as on the issue to be decided. The general approach for decision making aims at a multi-attribute approach. This can include the use of several risk measures as appropriate.

Section 5 provides some recommendations on the main risk metrics to be used for an extended PSA. For Level 1 PSA, Fuel Damage Frequency and Radionuclide Mobilization Frequency are recommended. For Level 2 PSA, the characterization of loss of containment function and a total risk measure based on the aggregated activity releases of all sequences rated by their frequencies is proposed.

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Date	15/12/2016	19/12/2016	09/02/2017

MODIFICATIONS OF THE DOCUMENT

Version	Date	Authors	Pages or paragraphs modified	Description or comments
Rev. D0		A.Wielenberg	All	Initial version
Rev. D1		A.Wielenberg, C.Hasnaoui, H.Löffler, T.Siklossy, L. Burgazzi, P.Jan, A.Prosek,		
Rev. D2	24/11/2015	C.Hasnaoui (ed), T.Siklossy, J.Vitazkova, L.Burgazzi	Several	Additional contributions to section 2, appendix B, Integration of comments
Rev. D3	15/05/2016	A.Wielenberg, C.Hasnaoui, H.Loffler, E.Raimond	most	Major revision of section 2 and section 3, integration of comments, restructuring, conclusions.
Rev. D4	01/06/2016	E. Raimond	All	Approval review. Editorial modifications. Conclusions et recommendations of this version of the report need to be discussed further.
Rev. D5	03/11/2016	C. Hasnaoui, H. Loeffler	All	Takes into account end-user's workshop and review
Rev. 0	05/12/2016	C. Hasnaoui, H. Loeffler	A.Prosek, J.Vitazkova, P. Jan	Issue of first revision
Rev. D6	14/12/2016	H. Loeffler	all	Editorial: acceptance of modifications, comments deleted
Rev 7	09/02/2017	E. Raimond	few	Approval review.

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EXECUTIVE SUMMARY

This report provides a review of the main used risk measures for Level 1 and Level 2 PSA. It depicts their advantages, limitations and disadvantages and develops some more precise risk measures relevant for extended PSAs and helpful for decision-making. This report does not recommend or suggest any quantitative value for the risk measures. It does not discuss in details decision-making based on PSA results either.

The risk measures investigated in this report are related to the Level 1 and Level 2 PSA for NPP and the properties and characteristics of risk actually included into these models. Level 3 PSA risk measures and risk metrics are not discussed in this report but Level 2+ risk measures is covered. Level 2+ PSA is understood as a Level 2 PSA with a simple model extension for releases to the environment of the plant (Level 3 PSA).

The choice of one appropriate risk measure or a set of risk measures depends on the decision making approach as well as on the issue to be decided.

The general approach for decision making, aims at a multi-attribute decision making approach. This can include the use of several risk measures as appropriate.

There is not necessarily a need to aggregate all different risk measures into one overall risk measure. Nonetheless, the issue of suitable risk measures for aggregating risk from similar risk measures (e.g. Level 2 PSA release categories) is relevant for decision-making and comparison.

Section 5 provides some recommendations on risk metrics to be used for an extended PSA. For Level 1 PSA, Fuel Damage Frequency and Radionuclide Mobilization Frequency are recommended. For Level 2 PSA, the characterization of loss of containment function and a total risk measure based on the aggregated activity releases of all sequences rated by their frequencies is proposed.

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GLOSSARY

APET	Accident Progression Event Tree
CCA	ASAMPSA_E partner organisation
CCDF	Conditional Core Damage Frequency
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CDP	Core Damage Probability
CDS	Core Damage State
CFDP	Conditional Fuel Damage Probability
CFF	Containment Failure Frequency
CHRS	Containment Heat Removal System
CLRP	Conditional Large Release Probability
DBA	Design Basis Accident
DiD	Defense in Depth
DIM	Differential Importance Measure
ECIS	Emergency Coolant Injection System
EOP	Emergency Operating Procedure
ΔCDF	Change in CDF
EDF	ASAMPSA_E partner organisation
EPR	Evolutionary Power Reactor
EPR FA3	Evolutionary Power Reactor Flamanville 3
FD	Fuel damage at any location and at any operating condition of the plant
FDF	Fuel Damage Frequency
FDP	Fuel Damage Probability
FV	Fussell-Vesely Importance
HSF	Hazard State Frequency
HDMR	High Dimensional Model Representation
HP	High Pressure
HT	Heat Transport
HTS	Heat Transport System
I&C	Instrumentation & Control
ICCDP	Incremental Conditional Core Damage Probability
ICDF	Interface Core Damage Frequency
ICRP	International Commission on Radiological Protection

INES	The International Nuclear and Radiological Event Scale
IRIDM	(Integrated) Risk-Informed Decision Making
IVR	In-Vessel Retention
LP	Low Pressure
LRF	Large Release Frequency
LERF	Large Early Release Frequency
LOCA	Loos of Coolant Accident
LOOP	Loss Of Offsite Power
LTO	Long Term Operation
NPP	Nuclear Power Plant
PDCA	Process Approach for Management Systems
PDS	Plant Damage State
PDSF	Plant Damage State Frequency
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSR	Plant Safety Review
RC	Release Category, Release Class
RCF	Release Category Frequency
(I)RIDM	(Integrated) Risk-informed decision making
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RMF	Radionuclide Mobilization Frequency
RPV	Reactor Pressure Vessel
RR	Research Reactor
RPS	Reactor Protection System
S….	Site….
SARNET	Severe Accident Research NETwork of Excellence
SBO	Station Blackout
SCCI	Site Common Cause Initiators
SCDF	Seismic Core-Damage Frequency
SFDF	Spent Fuel Damage Frequency
SFP	Spent Fuel Pool
SFPDF	Spent Fuel Pool Damage Frequency
SRF	Small Release Frequency
SUI	Single-Unit Initiators
OAT	One-At-Time
UCRT	Universal Common Risk Target

LIST OF SYMBOLS

Symbol	Definition
$\varphi(l_{ij})$	Frequency (or probability) distribution of the sequence in the risk model (likelihood function)
l_{ij}	Likelihood
L_{ij}	Sequence for the "i" scenario" with "j" consequence (e.g. L_{CDF})
s_i	Scenario "i"
c_j	Consequence "j"
$\mu(s_i, c_j)$	Risk measure (Point value)
$E\left(\mu(c_j)\right)$	Mean value
r	Source term
FV	Fussel-Vesely
T_{av}	Reference time average
t, T, τ	Time
p_n	Baseline point of time
$\tilde{p}_n$	Point of time after a change to the plant (observed degradation, design change, procedure change, change in test, maintenance or inspection practice, change in performance of an SSC, changes to the PSA model, etc.) with respect to the baseline
e_k	Plant intermediate state
$t(c_j)\|e_k$.	Probability of transition to consequence cj conditional to plant intermediary state e_k

1 INTRODUCTION

1.1 Background

Nuclear power plant operation is a human activity that comes with its own risk and operation history has shown that a zero risk is not possible. PSA is one of the tools that is used to assess nuclear power plants risks[1]. This report focuses on risk measures for PSA Level 1 and Level 2.

As stated in ASAMPSA_E DoW [1], global results of PSAs are mainly expressed in terms of core damage frequency, or large (early) release frequency. They can be associated to safety objectives (for example INSAG 12 proposes, for existing reactors, CDF<10^{-4} /reactor year and LRF < 10^{-5} /reactor year) even if not all countries apply numerical targets associated to PSA.

The objective of this report is to further the understanding on advantage, limitations and disadvantages for risk measures used in PSA and to develop some more precise risk measures relevant for extended PSAs and helpful for decision-making.

Furthermore, this report intends to contribute to harmonize the understanding of PSA Level 2 risk measures in the PSA community as well as provide guidance on how non-experts can better understand and interpret PSA Level risk measures. A subsidiary aim is that these metrics should contribute to a common basis for discussion with the Off Site Emergency Planning community on the use of PSA Level 2 results.

With regard to risk metrics, it has first to be noted that they are directly and intimately connected to the understanding of risk and the approach to and intended area of application for any decision-making. In this respect, some initial remarks are needed. These will serve as the background against which risk metrics are evaluated and on which recommendations for risk metrics that are suitable for (extended) PSA are derived.

1.2 Report Objectives

The objectives of this report are to develop a common understanding of the terms and underlying principles related to risk assessment using PSA and to provide an overview over risk metrics and risk measures used in current PSA (Level 1 and Level 2) and to derive initial recommendations for risk metrics and risk measures suitable for extended PSA. These issues will be further developed in the ASAMPSA_E guidance on the use of extended PSA in decision making as part of the final ASAMPSA_E guidance.

[1] Appendix B (Section 9) provides a review of some major accidents, mainly Chernobyl and Fukushima, and their consequences as additional material to illustrate the aspects of risk to be considered when discussing risk measures for PSA.

There are multiple aspects of risk. This applies to nuclear power plants and other nuclear facilities. The objective of this report is limited to the specific aspect of risk as described by the fundamental safety objective in IAEA SF-1:

> “The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation.” [3], p. 4

Thus, the risk investigated in this report is the risk of failing to meet this objective. The report will restrict itself to risk metrics and risk measures, which either describe this risk or the risk of reaching an intermediate state, which is seen as a leading indicator of failing to meet the fundamental objective. More specifically, the report will focus on the risk of significant damages outside of the plant boundary, i.e. accidental releases with potential of affecting a large number of people and a significant part of the vicinity of the plant for an extended period of time.

It needs to be acknowledged that the risk of NPP is firmly placed in the Level 3 PSA domain according to the accepted definitions [4], [5]. As the ASAMPSA_E project and consequently this report is investigating issues of Level 1 and Level 2 PSA while Level 3 is not addressed, most of the risk metrics of this report will actually be related to intermediary states and consequences. This limitation has to be recognized.

1.3 Definitions

Risk (ASAMPSA2, Reference [2]):

Risk is defined relative to hazards or accidents. A hazard is something that presents a potential for health, economical or environmental harm. Risk associated with the hazard is a combination of **the** probability (or frequency) of the hazardous event and the magnitude of the consequences. The consequences can be represented in several dimensions. A usual engineering definition of risk associated with an event i is:

Risk(event i) = “the probability of an event i” x “the consequences of an event i”. [2], p. 69 after [66].

A more formal definition with the theoretical background is provided in Appendix A (Section 8.2).

Risk Model

A risk model is a logical model, which describes the risk relative to hazards (see above) and provides the means to quantify the risk with appropriate risk metrics and risk measures. Risk models usually gather models over numerous events. The PSA for a NPP is a salient example.

Risk Measure and Risk Metrics:

“In the context of risk measurement, a risk metric is the concept quantified by a risk measure.” [68]. The risk metric is a feature or property of the risk model like e.g. a consequence, a transition between two states of the risk model, or an indicator derived from another risk measure. The risk measure includes in addition the quantification procedure for the risk metric. Risk measures are used for the representation, discussion, and interpretation of PSA results. For risk measures like core damage frequency, conditional failure probability of a system, or basic event importance for CDF to be used, the risk model has to support the respective risk metrics. However, under the ASAMPSA_E project the two terms risk metrics and risk measures have been used without distinction. For this reason, in this report, the term risk measure will be used as a more comprehensive term even if only the risk metric is meant. The term risk metric will be used if specifically the metric aspect is addressed or if there would otherwise be ambiguities.

Quantitative Risk Criteria, Risk Limits and Risk Objectives:

A quantitative risk criterion is a threshold for a risk measure, usually applied for decision making. It is expected that the risk threshold is not exceeded.

A risk criterion is termed a risk limit, if the threshold **shall** not be exceeded (and otherwise remedial actions are expected).

A risk criterion is termed a risk objective, if the threshold **should** not be exceeded (and otherwise remedial actions are considered).

Qualitative Risk Criteria

A qualitative risk criterion is associated to general safety objectives without any numerical threshold. Typical examples for the nuclear industry are the following:

- the ALARA approach : the reduction of risks as far as reasonably achievable ; the background is in general the risk identification, the available technology for its reduction and the costs for risk reduction implementation,
- the practical elimination of accidents with consequences that would not be limited in space and/or in time.

1.4 Risk Metric Attributes

The following discussion largely follows the arguments of Johansen and Rausand [31], [32]. This section defines the desirable attributes for the PSA risk metrics and risk measures.

There are quite a lot of risk measures (and metrics), which can be used for PSA of NPP, see e.g. IAEA-TECDOC-1511 [65], Appendix I, as well as for PSA applications, see e.g. IAEA-TECDOC-1200 [63]. Risk analysts always consider the risk metric (e.g. a consequence c_j) and the quantification procedure (i.e. the measure function μ).

For the purpose of this report, evaluation criteria of risk measures are used as defined in [31], [32] and their application are discussed as follows:

1. Validity

   Validity describes whether the risk measure is in line with the assumptions made and the calculation approach applied in the risk model (predictive validity), and if the risk metric adequately reflects an aspect of the analysed risk and provides relevant information for decisions on risk (content validity). For the latter, an agreement of decision makers and stakeholders would be necessary [32]. Obviously, this cannot be achieved within this report. Instead, the report will provide an opinion on the validity of investigated PSA risk measures for certain purposes (cf. contextuality and acceptability).

2. Reliability

   Reliability describes if the risk measure (risk metric) is clearly defined and if its relation to the risk analysis is explicit and adequate. Moreover, reliability entails that the risk metric and risk measure allow for reproducible results (in the sense that two analysts with the same objectives, methodology, data, and assumptions will be able to come up with the consistent results [32]).

3. Transparency Transparency according to [32] means that the basis and rationale of a risk measure is clear and traceable for decision makers and stakeholders, if it is justified, and if the risk measure can contribute to the decision (cf. validity). Particularly, traceability entails the inclusion of judgements related to risk aversion or to risk acceptance (value judgement). For this report, investigations of all the aforementioned aspects of transparency are clearly out of scope. Instead, the report will give an opinion on the rationale and justification of a risk measure from a technical point of view. Moreover, risk measures will be evaluated whether they are risk-neutral, risk-averse or risk-accepting. In line with the assumptions of this report, risk measures that are judged to be risk-neutral will be recommended. For this report, transparency is an aspect of reliability, whereas risk aggregation properties of risk measures are discussed separately.

4. Unambiguity

   Unambiguity entails according to [32] the precise definition and delimitation of a risk measure, a clear interpretation for the risk measure results as well as an adequate approach for risk aggregation regarding the risk measure. For the purpose of this report, former aspects are included into the aspects of validity and reliability. With regard to risk aggregation, the aggregation of risk over e.g. consequences necessitates the definition of a new (aggregate) risk metric and the selection of a suitable risk measure. Therefore, risk measures will be checked for their risk aggregation properties; suitable risk aggregation metrics will be recommended, which are judged to be risk-neutral.

5. Contextuality

   Contextuality is defined in [32] as suitability for decision support. For the purpose of this report, this criterion is an aspect of the validity of a risk metric (see above).

6. Communicability (Out of the scope of the present report)

   Communicability is understood in [32] as the understandability of a risk measure for non-experts. As explained above, risk communication issues are out of scope for this report. The discussion in this report will be limited to the understandability of risk measures to the PSA community. This is already covered with the aspects of validity and reliability.

7. Consistency

   Consistency is interpreted in [32] as a requirement that the risk measure does not give rise to contradictions in its application for different analyses and for decision making, if it is suitable for defining a ranking of scenarios, if it is sensitive to specific (discretionary) assumptions on the modelling, and if there are different versions of a risk measure. For this report, the clear definition of a risk measure, its sensitivity on discretionary assumptions, and its ability to support decisions will be treated under headings of validity and reliability. Therefore, the check on consistency is limited to possible contradictions in relevant decision making scenarios.

8. Comparability and specificity (Not developed further, only risk aggregation is discussed)

   Comparability and specificity according to [32] are antipodes. A risk metric is considered (more) specific if it is restricted to (increasingly smaller) subsets of scenarios or consequences. Conversely, a risk metric is considered to be (more) comparable, if it can be used to aggregate risks over (increasingly larger) subsets of scenarios and consequences. This also applies to secondary risk measures via the underlying direct ones. As a rule of thumb, the use of specific risk measures requires expert level knowledge about the risk model (PSA) and the modelled system (NPP) experts, whereas comparable risk measures (e.g. core damage frequency) can be used also by non-experts.

   As already pointed out, the issues of risk aggregation and suitable aggregate measures will need to be discussed in this report. The other aspects mentioned in [32] cannot be treated in this report.

9. Rationality (Not developed)

   Rationality according to [32] is the requirement that risk measures are justified without inconsistencies. It includes a theoretical framework for the definition. Since this report is not restricted to a decision making approach based on expected utility, we follow [32] in not requiring that rationality of a risk measure includes that it must be compatible with expected utility theory.

   Since the remainder of the report is an attempt at checking the rationality of risk measures for (extended) PSA of NPPs with respect to a group of PSA experts, this aspect needs no further explicit consideration.

10. Acceptability (Not developed)

    Acceptability according to [32] summarizes whether the risk measure is considered adequate, informative and justified by stakeholders (i.e. fulfills the criteria given above). For this report, this cannot be investigated. Instead, the report provides an opinion on the merit of the different investigated risk measures and recommendations on the use of certain risk measures for determined purposes, which substitutes for acceptability.

In summary, risk measures are systematically evaluated regarding the following attributes:

1. Validity
2. Reliability
3. Consistency
4. Risk aggregation properties including judgments on appropriate risk-neutral aggregation approaches.
5. Understandability to the PSA community

The properties and implications of an extended PSA will be considered in all cases. This specifically relates to risk measures for risk aggregation.

1.5 Risk Consideration for Decision Making

There is no common understanding on the correct (or even appropriate) approach to decision making regarding risk in the scientific community as well as with actual end-users [52]. Depending on the subject matter to decide and the role and the interest of the decision maker or stakeholder, different approaches to decision making are advocated or rejected [22], [25], [46], [47], [52], [54], [7]. Moreover, the acceptability of these approaches to the stakeholders or the society obviously depends on the culture of the society in question and the specific values and believes on risk acceptance on a personal and societal level [58]. For the purpose of the ASAMPSA_E project, work on the ethical or legal or theoretical foundations of decision-making [17], [49], [50], [51], [52] is clearly out of scope, as is a discussion of cultural influences.

The present report focuses on risk measures based on PSA, thus an operational definition of the basic decision making approach is needed. The approach propagated by INSAG on (integrated) risk-informed decision making (IRIDM) in INSAG-25 is identified as this foundation [6]. It is in general terms consistent with approaches by regulatory authorities on decisions for nuclear facilities in using information from Level 1 and Level 2 PSA [7], [8], [10], [11], [13], [55] and is in line with WENRA [72], [73] and IAEA requirements on the use of PSA information in safety assessment and decision making [1], [4], [5], [56].

In summary, INSAG defines IRIDM as a process (broadly following a PDCA[2] approach [57]) where for an issue first decision options are defined. For those, a systematic assessment of potentially relevant aspects (mostly: safety assessments) is performed. The results are evaluated and used for an “integrated decision” i.e. taking into account all relevant factors. Thereafter, the decision is implemented, the implementation is monitored and corrective actions are derived if needed, thus closing the PDCA loop (cf. Table 1)

[2] **PDCA** (plan-do-check-act or plan-do-check-adjust) is an iterative four-step management method used in business for the control and continuous improvement of processes and products.

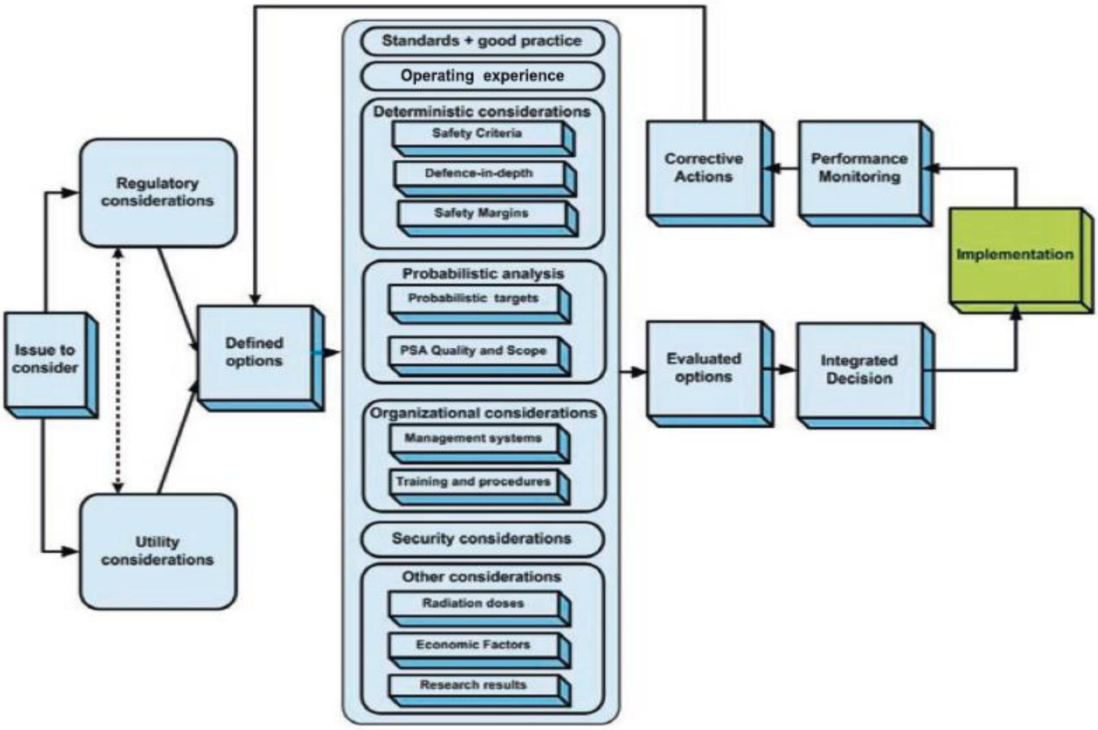

Table 1: Key elements of IRIDM approach from INSAG-25 [6] , p. 6

Moreover, INSAG-25 recommends using a risk-informed approach for all safety related decisions on nuclear installations, if such an approach is merited. Following GSR Part 4 [56], this is consistent with a graded approach to safety assessment. If a decision can be made using a less systematic and less onerous approach, it is not necessary to do (lots of) unnecessary assessments and investigations. Nonetheless, for any organizations following a quality and safety management approach, the generic decision process would contain these elements.

With regard to this report, the INSAG approach defines the scope of the applications for which results of an extended PSA for nuclear power plants could be used in decisions related to the safety of an NPP.

Importantly, INSAG-25 does not recommend a specific approach for arriving at a decision, i.e. on how different aspects should be balanced against each other. From the theoretical background on decision making, there are several approaches for this task like “value at risk” [47], [52], “loss of life” [33], [34], expected utility or multi-attribute utility theory (MAUT) [59], [45], [51], prospect theory [51], risk matrices [27], [23], [45], etc. The choice of one appropriate risk measure or a set of risk measures depends on the decision making approach [32] as well as on the issue to be decided. For the purpose of this report, certain assumptions on risk measures for decision making have to be made.

First, the general approach by INSAG-25 clearly aims at a multi-attribute decision making approach. This can include the use of several risk measures as appropriate. Consequently, this report should discuss risk measures regarding their suitability.

Second, since we assume a multi-attribute approach, there is no need to aggregate all different risk measures into one overall risk measure (i.e. effectively a utility or disutility function). Thus, there is no need for recommending one overarching, consistent risk measure, which aggregates over different risk measures. Nonetheless, the issue of suitable risk measures for aggregating risk from similar risk measures (e.g. Level 2 PSA release categories) should be investigated.

Third, the risk measures investigated in this report should be closely related to the Level 1 and Level 2 PSA for NPP and the properties and characteristics of risk actually included into these models. Level 3 PSA risk measures and risk metrics will not be discussed systematically in this report[3] but Level 2+ risk measures will be covered. Level 2+ PSA is understood as a Level 2 PSA with a simple model extension for releases to the environment of the plant (Level 3 PSA).

Extended definitions of risk (i.e. “stakeholder acceptance” and similar non-technical approaches) are out of scope of this report. It focuses on those risk metrics and measures that are used by practitioners and reviewers of PSA for NPP for evaluating PSA results and for communication with the PSA community and with regulators. Still, the suitability of risk metrics/measures for communicating with non-expert stakeholders and the general public should be addressed as appropriate.

Fourth, the issue of risk aversion and risk-taking during decision making is out of scope of this report. The decision making process shall be assumed to be “risk-neutral”. However, since we do not require that a unique utility function exists and has specific properties (von Neumann-Morgenstern axioms) [14], [32], a working interpretation of risk neutrality will be derived. With respect to risk metrics/risk measures this at least includes the requirement that risk metrics/risk measures recommended in this report should not be defined to be explicitly risk averse or risk accepting.

1.6 Structure of the Report

Section 2 provides an overview of the current status of risk metrics/measures for Level 1 PSA. Section 3 provides an overview of the risk metrics/measures for Level 2 PSA. Section 4 discusses multi-sources risk metrics and Section 5 presents some recommendations on risk measures for an extended PSA and Section 6 presents the main conclusions of the report.

2 RISK METRICS FOR LEVEL 1 PSA

In this section, risk measures for PSA Level 1 are presented and discussed. The basic approach is to present first direct risk metrics and the related risk measure. Then, secondary[4] risk measures related to the direct risk measures are investigated.

Since the same concepts presented in the following subsections can be applied to all direct (and some secondary) risk measures, the basic definition of the measure and its typical fields of application will be given at some length

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[3] For the subdivision of PSA in levels, the common definitions will be used, cf. e.g., SSG-3 [4] and SSG-4 [5]. Specifically, Level 2 PSA stops at the releases from the plant to the environment, i.e. effectively at the plant fence.

[4] For more information on direct and secondary risk measures/risk metrics as well as other technical concepts referenced in this section cf. section 8.

for the case of CDF (Section 2.1). For other direct risk measures, only relevant differences or issues for special consideration will be mentioned.

All risk measures may be used as time average and/or time dependent variants. Specifics and differences of these variants are illustrated on Core Damage Frequency in Section 2.1 (time average) and Section 2.2 (time dependent).

For each risk measure, the following issues are considered.

- Definition(s) of the risk measure.
- Areas of application in PSA for NPPs.
- Discussion of validity, reliability, consistency of the risk measures, its risk aggregation properties, and its understandability to the PSA community (cf. section 1.4).
- Limitations as per the risk assessment.
- Recommendations on a harmonized definition of the risk measure will be given, if applicable.

The following sections on individual risk metrics take into account numerous publications related to that matter. The following are some of these references: [65], [63], [66], [69], [41], [42], [13].

2.1 Core Damage Frequency (CDF), time average

2.1.1 Definition of Risk Measure

<u>Risk metric</u>:

Core damage for PSA Level 1 is commonly understood to occur if there is a significant degradation of reactor core components (like fuel rod (cladding) or control rod). The core damage metric is constitutive for the definition of PSA Level 1 [4], [5], because the dividing line between Level 1 and Level 2 is usually set at the onset of a significant degradation of reactor core components (with release of fission products outside of the reactor core). There is, however, a wide range of specific definitions for the core damage metric depending of the PSA Level 1 objectives and the NPP design [69], [13], e.g.

- Maximum fuel element cladding temperature above 1204 °C,
- Changes in core geometry are such that core cooling is no longer deemed successful,
- Uncovery of the top of the core except for short-term reflooding,
- Uncovery of top of active fuel,
- Onset of heat-up of the reactor core due to anticipated prolonged oxidation involving a part of the core sufficient for causing a significant release,
- Onset of melting of core material (fuel elements, control rods) in the reactor core or the fuel storage pool, e.g. [77],
- “Uncovery and heatup of the reactor core and leading to a significant release of radioactive material from the core” [74], p. 49., if the initiating event occurs during power operation,
- Loss of structural integrity of more than one fuel channel (due to molten fuel) for CANDU reactors,
- etc.

The risk metric is usually applied to (end-) states in the risk model (i.e. a consequence). Core damage is one of the constitutive attributes for the (uncontrolled) end-states in PSA Level 1.

There are different practices as to whether “core damage” applies only to fuel elements present in a reactor core or if it can, especially for aggregation, also include damages to fuel elements outside of a reactor core, in particular fuel losing overall cladding integrity in the spent fuel pool. For more discussion see below; for fuel damage frequency see section 2.12.

Risk measure:

The quantification for the risk metric “core damage” is always the direct frequency (or probability) of the sequence in the risk model, i.e. it assigns $\varphi(l_{ij})$ to the sequence $s_i \xrightarrow{\varphi(l_{ij})} c_j$ (Where s_i is the “i” scenario with the “j” consequence c_j). For aggregating risks over sequences, the probabilities for all sequences with core damage are “summed up”. Formally, the frequency distribution $\varphi(l_{CDF}) = \varphi\left(\cup\, s_i \xrightarrow{\varphi(l_{ij})} c_j \mid \forall i \text{ and } c_j \in CD\right)$ is computed. If all sequences are independent (i.e. there are no common minimal cuts between the sequences), then frequencies can simply be added up.

The time averaging for the risk measure is usually done over one reactor year of full power operation (/ ry), or for the duration of the operating state per year. See also Core Damage Frequency, time dependent (section 2.2). The time-averaging is often based on approximations, e.g. by using respective estimations for basic event failure probabilities when quantifying minimum cut sets. Then, results for different reference times (per year, per reactor year, duration of operating state per year) can be converted into each other by multiplying with the relation between the respective time durations. Certain time-dependent effects are however neglected but are captured in the time-dependent CDF measure. If these effects are essential, then the time-dependent CDF would have to be integrated over the reference time T_{av} (cf. section 2.2).

$$l_{CDF,T_{av}} = \frac{1}{T_{av}} \int_0^{T_{av}} l_{CDF}(t)dt$$

Use of uncertainty distributions:

The CDF measure l_{CDF} is initially computed with point values for likelihoods. These results are typically used in the heuristic determination of minimal cuts in PSA tools like CAFTA®, FinPSA®, or RiskSpectrum® in cut-off algorithms. Point value CDF figures are then complemented with the uncertainty distribution $\varphi(l_{CDF})$ for the CDF with a Monte Carlo analysis based on the distributions for basic events $\varphi(l(X_i))$. The mean value as well as quantiles (5%, median, 95%) is often stated for PSA results.

Simultaneous averaging over time and over parameter uncertainties is often not supported by PSA programs and thus not performed for PSA results.

2.1.2 Areas of application

Core damage frequency is the most common measure of risk since most nuclear power plants have undergone at least a Level 1 PSA and the methodology is well established. In many countries, numerical values of this type are used either formally or informally as probabilistic safety goals or criteria [4]. CDF has been used for PSA for licensing, submissions to the regulator, oversight, design alternatives, risk gap analysis, Risk management and Risk-informed decision making.

2.1.3 Discussion

Validity:

Core damage is – once defined – a clearly described state in the risk model. Core damage states (and comparable other losses of fuel integrity) are a precondition for releases from a NPP that can challenge the fundamental safety objective. In that respect, core damage is a valid leading indicator and can provide relevant information to PSA specialists and decision makers. However, the CDF is unable to discriminate between scenarios likely leading to very large releases and scenarios likely leading only to small releases. Assigning the likelihood (distribution) to the sequence(s) ending in core damage is a clear and traceable quantification procedure. Averaging risk model results over time is a sensible and consistent way of defining a risk measure. In this regard, CDF is a valid risk measure for most purposes.

Reliability:

There is no unique definition available for a core damage state. It is a well-acknowledged fact that core damage states need to be defined separately for different fundamental reactor designs [71] (e.g. LWR reactors, CANDU type reactors, fast breeder reactors, etc.). Even for LWR reactors, several slightly different definitions are in use. However, these differences for LWR reactor CDF measures are widely regarded as having only a minor impact on PSA results. The issue is further complicated by the problem that "core damage" in some models encompasses fuel element damage in e.g. the fuel pool, or that "core damage" is restricted only to fuel damage during power operation.

For some non-LWR reactor types like some Gen IV designs, a core damage metric is hard to define in a meaningful way e.g. Gen IV reactors like a molten salt reactor, "core damage" in the sense of "fuel starts to melt and leads to a severe accident" is not applicable.

However, if the core damage metric has been clearly established, it allows for – in principle – reproducible PSA modelling of the accident sequence analysis. Observed differences between models are usually due to analysts' choices on the scope, level detail, and degree of conservatism in accident sequence modelling rather than due to different understanding of the core damage metric. In that sense, CDF is a reliable measure.

Consistency:

Core damage frequency induces an order relation satisfying rationality criteria, if risk aggregation properties are properly considered. Basically, the measure is consistent. However, the issues related to reliability and risk aggregation properties should not be overlooked.

Risk aggregation properties:

Aggregating CDF values over multiple scenarios (i.e. frequency values) is a well-defined operation, if performed on a minimal cut set basis as described above, resulting in a consistent risk measure. With respect to risk aggregation, PSA specialists and decision makers have to acknowledge the limitations of the CDF measure. As there is no distinction between core damage states that likely lead to large releases and those that likely lead only to limited releases, simply aggregating the risk from this scenarios can (and often will) obfuscate the risk profile of the plant with regard to the fundamental safety objective.

Understandability to the PSA community:

CDF is a widely used risk measure. It is well understood in the PSA community as the risk at the end-point of PSA Level 1. Existing ambiguities and differences in the definition of "core damage" do not significantly impede the interpretation of results stated as CDF, they do however hamper comparisons between plants and designs. Another issue are advanced (planned) reactor concepts, e.g. "core damage" in the sense of "fuel starts to melt and leads to a severe accident" is not applicable for Gen IV reactors like a molten salt . The CDF measure is so entrenched in the PSA community for NPP that there are attempts to define a CDF measure for these reactor types as well.

2.1.4 Limitation

There are several limitations to the CDF (time average) risk measure.

Importantly, the CDF metric does not distinguish between severities of core damage (extent of damage to fuel rods) beyond the defining threshold for core damage. In this respect, the CDF measure is likely inappropriate for investigating workplace risk (irradiation of on-site staff in case of minor fuel damages during operation). Such scenarios, which sometimes are analyzed with PSA models, require dedicated risk measures.

Similarly, certain kinds of scenarios (e.g. mechanical damage to fuel rods during refueling operation, fuel pool accidents…) are not covered by the CDF measure. Moreover, the limitations arising from the different definitions of the CDF measure do apply (e.g. restriction to fuel elements in the core, no consideration of shutdown states, etc.)

Another limitation, which has already been mentioned above, is that the CDF metric does not preserve (or provide) information on core damage characteristics in light of expected releases (e.g. time of core damage onset, pressure in the RPV at core damage, status of barriers and safety systems, etc.).

Conceptually, the core damage metric defines the transition from PSA Level 1 to PSA Level 2. Because of the limitations of the CDF measure, the interface between Level 1 and Level 2 is usually based on more detailed characterizations of the plant damage state reached. For more detail, see section 2.13.

Risk profile of the plant is another limitation related to CDF metric since it represents frequency of core damage only and thus do not cover the IAEA requirements for risk assessment where risk is defined as multiplication of frequency and consequences. Since the IAEA definition of safety is based on control over sources which is limited by the fourth barrier of Defence in Depth – containment – the consequences with respect to general safety objective are releases with their potential to impact significantly population health, environment and economy. Ultimately, it is alleged in the previous paragraph, that no distinction between core damage states can obfuscate the risk profile of the plant, and even the significant reduction of CDF not necessarily means significant reduction of large releases.

Another limitation which is worth to mention is that PSA results are interpreted in fact as “per reactor year” even though all the data in PSA models are only time related: per hour, per month, on demand etc. So, the PSA result is indeed per year and does not take into account different plant operation states during the year. The results should be integrated over lifetime of the plant (taking into account all reactor states and all sources), or e.g. over the time period 10000 years. Here it should be also noted, that the IAEA CDF limit is not plant specific and therefore is to be applied for all types of reactors[5].

2.1.5 ASAMPSA_E recommendation on CDF (Time Average)

From the discussion above, no specific and simultaneously universal definition of the core damage metric can be attained. However, the PSA community should agree on a common understanding of the core damage metric pertaining to a specific reactor type. That definition should be chosen so that the CDF measure is placed at the interface between PSA Level 1 and Level 2, i.e. that CDF integrates accidental scenarios with the potential for severe off-site releases related to the core of the reactor. To this end, the CDF measure needs to be consistent with the plant damage state frequency measure(s) (PDSF) it shall aggregate, cf. section 2.13.

For historical reasons, the final results of PSA Level 1 have often been given as CDF. Several regulators have set Level 1 objectives based on the CDF measure. Semantically, the core damage metric needs to be restricted to the “reactor core”, i.e. the fuel in the reactor that is used for maintaining the chain reaction. Reinforced by the Fukushima Dai-ichi, other risk measures have been defined for fuel outside of the reactor core (e.g. the SFP). Conceptually, the fuel damage metric and measure should be a more general measure, combining all sorts of scenarios with damage to reactor fuel (irrespective of its location on the site or the operating status of the plant), cf. section 2.12.

Therefore, the recommendation by the ASAMPSA_E is that CDF should be defined as a subset of the FDF measure, specifically covering accidental scenarios with the potential for severe off-site releases related to the core of the reactor. Moreover, the CDF measure shall be consistent with the PDSF measure(s), which are assigned to accidental scenarios with the potential of severe releases related to the reactor core.

With regard to the relationship between CDF, time averaged, and CDF, time dependent, the ASAMPSA_E project recommends raising awareness about the limitations of the respective calculation methods. To the extent practicable, CDF quantification should be done based on CDF, time-dependent. Thereby, explicitly time-dependent effects, like e.g. staggered testing schemes, will be adequately considered in PSA results. In

[5] It is worth to mention, that some authors assert that calculated results are not consistent with operational experience. The operating experience and statistics show much higher CDF than PSAs models show. If one takes into account 6 CDF of large extent (more than 25%) in history – Bohunice A1, Slovakia, in 1977 with 25% officially reported core melt, TMI2 – Surry, USA, 1979 with 50% officially reported core melt, Chernobyl, Russia, 1986 with 100% of core melt and Fukushima, Japan, in 2011 3 cores of various extent of melting in 14.500 reactor years this results in the frequency about 4E-4/Ry which exceeds the IAEA CDF limit E-4/Ry. Other authors assert that this argument is based on faulty premises and does not provide valid insights on PSA in general.

uncertainty analysis, time averaging should be done before calculation of mean values[6] (See Appendix A for more details).

$$E(\varphi(l_{CDF,T_{\text{av}}})) = E\left(\varphi\left(\frac{1}{T_{\text{av}}}\int_{0}^{T_{\text{av}}} l_{CDF}(t)dt\right)\right)$$

This results in good consistency with commonly used quantification approaches. PSA programs should provide the functionality needed for such computations of $\varphi(l_{CDF,T_{\text{av}}})$.

2.2 Core Damage Frequency, time dependent

2.2.1 Definition of Risk Measure

The definition of the risk measure is the same as for core damage frequency (section 2.1), the only difference being that the risk model is quantified at a specific point in time T with the particular plant status at this point in time ("picture" of the NPP at a certain time t taking into account actual unavailabilities, etc.). Fundamentally, the time-averaged CDF value can be obtained by integrating the time dependent CDF likelihood over the interval T_{av}.

$$l_{CDF,T_{\text{av}}} = \frac{1}{T_{\text{av}}}\int_{0}^{T_{\text{av}}} l_{CDF}(t)dt$$

<u>Use of uncertainty distributions:</u>

The CDF measure $l_{CDF}(t)$ is initially computed with point values for likelihoods, i.e. minimum cuts are quantified with basic event models quantified at time t with the nominal values (mean values) of uncertain parameters. The uncertainty distributions $\varphi(l_{CDF}(t))$ can be determined with Monte Carlo analysis for each point in time. Usually, the uncertainty distribution $\varphi(l_{CDF})$ is not determined via $\frac{1}{T_{\text{av}}}\int_{0}^{T_{\text{av}}}\varphi(l_{CDF}(t))dt$. Consequently, a simultaneous time-averaging and uncertainty evaluation for the time-averaged CDF values is not done in current PSA, as already mentioned in section 2.1.

2.2.2 Areas of application

See Section 2.1.2

2.2.3 Discussion

<u>Validity</u>:

The same considerations already given in section 2.1 apply. For risk monitors and other PSA applications where the risk value at a certain point in time needs to be known, the time dependent version of the CDF measure needs to be chosen.

<u>Reliability</u>:

[6] The E( ) in the formula below denotes the expected value (i.e. mean) of the probability distribution $\varphi(l_{CDF,T_{\text{av}}})$. See also in section 2.2 on CDF, time dependent.

The same considerations already given in section 2.1 apply.

Consistency:

The same considerations already given in section 2.1 apply.

Risk aggregation properties:

The same considerations already given in section 2.1 apply, if risk is aggregated for at a certain point in time t. For risk aggregation over a time period, the respective formula has been given above, which is a strait forward and consistent operation. The differences to the common application of CDF, time-average, measure should be noted, as explained above.

Understandability to the PSA community:

CDF, time dependent, is a widely used risk measure for risk monitors and other PSA applications, where the time-dependent behavior of the CDF measure is of importance, like for instance risk budgeting for a plant considering planned changes in operating states and (“unfortunate”) operating events. It is well understood in the PSA community as the risk at the end-point of PSA Level 1. Existing ambiguities in the definition of “core damage” do not significantly impede the interpretation of results stated as CDF; they do however hamper comparisons between plants and designs.

2.2.4 Limitation

The discussion under section 2.1 applies. In addition, CDF, time dependent, risk measure analyses particular plant states existing at the point in time of interest.

2.2.5 ASAMPSA_E recommendation on CDF (Time Dependent)

For the underlying issues on core damage frequency, see section 2.1. The same, consistent definitions of CDF and FDF should be applied. With regard to CDF, time averaged, the ASAMPSA_E project recommends raising awareness about the limitations of the respective calculation methods. To the extent practicable, the time-averaged value should be computed based on the time-dependent version, cf. section 2.1:

$$\varphi(l_{CDF,T_{\mathrm{av}}}) = \varphi\left(\frac{1}{T_{\mathrm{av}}}\int_0^{T_{\mathrm{av}}} l_{CDF}(t)dt\right)$$

In risk monitors, if $l_{CDF}(t)$ considers the current status of the plant including current unavailabilities of components or systems, then $\frac{1}{T_{\mathrm{av}}}\int_0^{T_{av}} l_{CDF}(t)\,dt$ can be used to calculate the risk challenge for a period.

2.3 Change in CDF (Time Average and Time Dependent)

In the following sections 2.3 to 2.10, risk measures (secondary risk measures) derived from the CDF measure are discussed exemplarily. Since the same concepts presented in the following subsections can be applied to all direct (and some secondary) risk measures, the basic definition of the measure and its typical fields of application will be given at some length for the case of CDF. For other direct risk measures, only relevant differences or issues for special consideration will be mentioned.

2.3.1 Definition of Risk Measure

Evidently, risk measures reflecting changes in core damage frequency are the most commonly applied secondary relative risk measures originated by the direct risk measure of core damage frequency. Time average and time dependent changes of core damage frequency can also be interpreted.

The assessment related to time average changes of core damage frequency is usually based on the impact due to a modification being evaluated from p_n to $\tilde{p}_n$ relative to a "baseline" value. The change may be due to an observed degradation, design change, procedure change, change in test, maintenance or inspection practice, change in performance of an SSC, or changes to any input or assumption associated with the PSA model, etc. Therefore the change in the risk measure associated with the measure for significant degradation of the reactor core (*CD*) is:

$$\Delta CDF, T_{av} = CDF, T_{av}|\tilde{p}_n - CDF, T_{av}|p_n.$$

Furthermore, time average change in core damage frequency needs to be evaluated at a specific point in time, *T*. In this manner time-averaging can be performed by using the following formula:

$$\Delta CDF, T_{av} = \frac{1}{T_{av}} \int_0^{T_{av}} \Delta CDF(t)dt = \frac{1}{T_{av}} \int_0^{T_{av}} (CDF(t)|\tilde{p}_n - CDF(t)|p_n)dt$$

Time dependent changes in the core damage frequency reflects the difference between the core damage frequency relevant for two certain points of time with the associated particular plant states. Obviously, for the calculation of this time dependent relative risk measure, time dependent core damage frequency as a direct risk measure needs to be taken into consideration. The secondary risk measure is simply defined by

$$\Delta CDF(t, t_0) = CDF(t) - CDF(t_0)$$

2.3.2 Areas of application:

The change in CDF is a widely applied secondary risk measure in several PSA application areas, since it usually reflects some modification on the plant (e.g. change in the design, procedures, test, maintenance or inspection practice) or refinement to any input or assumption associated with the PSA model. In this manner it can be used, amongst others, for planning and prioritizing purposes. Hereby we list some examples of PSA applications that have relevance concerning the change of CDF risk measures (for a more comprehensive presentation see [65]):

- NPP upgrades, backfitting activities and plant modifications,
- risk-informed support to plant ageing management programs,
- risk monitor,
- periodic safety review,
- development and improvement of the emergency operating procedures,
- improvement of operator / maintenance personnel training program,
- maintenance program optimization,

- exemptions to technical specifications and justification for continued operation,
- determination and evaluation of changes to allowed outage times and changes to required technical specification actions,
- determination and evaluation of changes to surveillance test intervals,
- risk-informed in-service inspections / testing,
- planning and prioritization of inspection activities,
- risk evaluation of corrective measures,
- long-term regulatory decisions.

2.3.3 Discussion

Validity:

The risk measure of change in CDF compares two scenarios (before and after a change) with respect to their impact on plant safety. In this manner, change in CDF is a valid secondary risk measure for most purposes providing relevant information to PSA specialists and decision makers. Furthermore, similarly to the risk measure of CDF, the change in CDF cannot reflect risks associated to very large release or only a small release.

Reliability:

As it was already presented in section 2.1.3, there is no unique definition available for a core damage state. However, if the core damage metric has been clearly established, it allows for – in principle – reproducible PSA modelling of the accident sequences. Apart from the specificities of the CDF risk metrics itself, PSA analysts have the same understanding on the change in CDF as a secondary risk measure, hence it can be considered as a reliable measure.

Consistency:

The change in CDF shows the increase or the decrease of the plant risk with respect to significant degradation of the reactor core. In this respect – not taking into consideration the characteristics of the direct risk measure of CDF – the evaluation of the change in CDF is a suitable tool to help decision makers, not giving rise to contradiction in relevant decision making scenarios.

Risk aggregation properties:

As change in CDF is a derivative measure of the CDF, both risk measures have the same fundamental risk aggregation properties including their limitations too (see also section 2.1.3). However, as a secondary risk measure, aggregating $\Delta\mathrm{CDF}$ values needs to be done by applying the set of all changes $\tilde{\mathrm{p}}_{\mathrm{n}}$ (and in principle the set of all baseline values) to the CDF model. This operation is well-defined if the set of all changes can be defined consistently. Then, the overall $\Delta\mathrm{CDF}$ value can be computed. Analysts (and decision makers) have to be aware that the respective result can deviate significantly from any sum of single $\Delta\mathrm{CDF}$ values, for example: If there are two changes, each increasing CDF by 10, then having them simultaneously could result in 100 instead of 20. What actually happens is hard to predict a priori. If, as an extreme case, changes trigger a two element minimum cut, then CDF is 1.

Understandability to the PSA community:

The change in CDF is a widely used secondary risk measure, which is well understood by the PSA community (besides the difficulties arising from the lack of general definition on core damage, see section 2.1.3).

2.3.4 Limitation

Amongst others the most important limitation of the change in CDF risk measure is – similarly to CDF direct risk measure - that it cannot distinguish between severity of core damage beyond the defining threshold for core damage. In this manner it cannot be identified by the risk measure of change in CDF, that the difference concerns risk contributions related to very large releases, only small releases or a certain combination of thereof. On the other hand change in CDF does not reflect any information on core damage characteristics in light of expected releases (e.g. time of core damage onset, pressure in the RPV at core damage, status of barriers and safety systems, etc.) with respect to the difference indicated by the change in CDF. For more details on the limitations of CDF, hence on the change in CDF, see section 2.1.4.

2.3.5 ASAMPSA_E recommendation on Change in CDF

Besides the recommendations related to the risk measures of CDF time average (see section 2.1.5) and CDF time dependent (see section 2.2.5), the definition presented in section 2.3.1 is widely used and accepted by the PSA community. Therefore no additional harmonized definition is recommended for the risk measure of change in CDF.

2.4 Conditional Core Damage Probability (CCDP)

2.4.1 Definition of Risk Measure

Conditional core damage probability is a secondary conditional consequence risk measure originated by the direct risk measure of core damage frequency. Depending on the boundary conditions of the assessment, this secondary risk measure can be derived from the CDF: independently of any duration of time, or on the basis of a certain time interval.

Conditional core damage probability irrespective of time duration can be derived from the risk model by including intermediate states (besides core damage) e_k, i.e.

$$s_i \xrightarrow{\varphi(l_{ij})} c_j |\, c_j \in CD \Leftrightarrow s_i \xrightarrow{\varphi(l_{ik})} e_k \xrightarrow{\varphi(t(c_j)|e_k)} c_j |\, c_j \in CD$$

with conditional transition probabilities $t(c_j)|e_k$. In this case the intermediate state (e_k) may represent the occurrence of an initiating event (with or without other SSC failures) or degradation of mitigation systems denoted by *DC* (Degraded plant Conditions) hereinafter. With this definition, the secondary risk measure of conditional core damage probability can be defined as

$$\mu(c_j|e_k, c_j \in CD \text{ and } e_k \in DC) = \varphi(t(c_j)|e_k, c_j \in CD \text{ and } e_k \in DC)$$

in a natural way. In other words, conditional core damage probability is the probability of significant degradation of the reactor core (CD) upon the condition that an initiating event occurs. Accompanying the occurrence of an initiating event, degradation of mitigation systems can also be taken into consideration as properties of the intermediate state. Time average CDF risk measure as well as time dependent CDF risk measure at a certain point of time can be taken into consideration during the assessment of conditional core damage probability.
If the change in risk due to the occurrence of an initiating event is in the focus of the analysis, then the risk model is processed by setting the corresponding initiating event to TRUE and making adjustments as seen necessary to model the effect of any additional failure events that may also have occurred. The relevant event tree(s) is evaluated by quantifying the probability of core damage given the occurrence of the initiating event in question.

If there are failures in mitigation systems without the occurrence of an initiating event, then duration dependent conditional core damage probability can be assessed by utilizing the time dependent version of the conditional core damage frequency (for details see section 2.5).
A CDF based secondary conditional consequence risk measure is the cumulative conditional core damage probability (*CumCCDP*) over a certain time interval *T*. This risk measure can be obtained by time integration of the conditional core damage frequency as follows:

$$CumCCDP(T) = \int_0^T CCDF(t)dt$$

Another interpretation of the time dependent, CDF based secondary conditional consequence risk measure is the incremental conditional core damage probability (*ICCDP*). This risk measure is the increase in risk of the plant for a specific configuration *i*, for example the unavailability of a component, with the duration *T*. $ICCDP_i$ can be sensibly computed as:

$$ICCDP_i = \int_0^T (CCDF_i(t) - CDF_{baseline}(t))\, dt$$

This risk measure is used world-wide for probabilistic evaluation of operational experience. For example the Swiss regulator recommends in [76] to use the following formula to estimate $ICCDP_i$ (Δt_i is the duration of component unavailability configuration in hours) given $CCDF_i$ and $CDF_{baseline}$ are constant within time Δt_i:

$$ICCDP_i = (CCDF_i - CDF_{baseline})\frac{\Delta t_i}{8760\ h/year}$$

They are referred to as a conditional probability because they are conditioned on being in a specific plant configuration. The definition references a so-called baseline CDF, which corresponds to a zero-maintenance model of the plant [76].

It can be implied by their definition, that cumulative conditional core damage probability and incremental conditional core damage probability are derived risk measures based on other secondary risk measures, i.e. on time dependent conditional core damage frequency *(CCDF(t))*.

2.4.2 Areas of application:

Conditional core damage probability is also a widely applied secondary risk measure in several PSA application areas, since it reflects, amongst others, the level of risk in a certain condition of the plant (at a certain time point or for a time period). In this manner it can be used for screening purposes, e.g. an external event has a mean occurrence frequency <10-5/yr, and the mean value of the conditional core damage probability is assessed to be <10-1. Also as a significant application area, this risk measure can be a tool to calculate the usage of a predefined risk budget for a given time period, e.g. for a year. Hereby we list some examples of PSA applications that have relevance concerning the risk measure of conditional core damage probability (for a more comprehensive presentation see [65]):

- risk informed (PSA based) evaluation and rating of operational events,
- real time configuration assessment and control (response to emerging conditions),
- risk monitor,
- dynamic risk-informed technical specifications,
- determination and evaluation of changes to allowed outage times and changes to required technical specification actions,
- risk-informed in-service inspections,
- configuration planning (e.g. support to plant maintenance and test activities),
- exemptions to technical specifications and justification for continued operation.

2.4.3 Discussion

Validity:

This risk measure implies the level of risk on an NPP having a special plant configuration at a certain point of time or for a time period. In this manner, conditional core damage probability is a valid risk measure for several purposes providing relevant information to PSA specialists and decision makers. However, conditional core damage probability (similarly to the risk measure of CDF) is unable to discriminate between scenarios likely leading to very large releases and scenarios likely leading only to small releases.

Reliability:

With respect to reliability, the same applies to conditional core damage probability as to change in CDF (see section 2.3.3).

Consistency:

Conditional core damage probability shows the probability of significant degradation of the reactor core (*CD*) upon the condition that a specific plant configuration is present. In this respect – not taking into consideration the characteristics of the direct risk measure of CDF – the evaluation of the conditional core damage probability is a suitable tool to help decision makers, not giving rise to contradiction in relevant decision making scenarios.

Risk aggregation properties:

As conditional core damage probability is a derived measure of the CDF, it inherits the basic limitations on risk aggregation properties (see also section 2.1.3). With respect to aggregating CCDP results, simply adding these figures is incorrect in most cases. Instead, Bayes' law has to be respected. In practice, the aggregation of the conditional core damage probabilities for different intermediate states should be performed by implementing model rearrangements and/or special boundary conditions (house events, flags) that are relevant to all intermediate states in question. Then the modified model should be evaluated by an appropriate quantification approach. It is often not appropriate to separately model each intermediate state and aggregate the risk measures by summing them up one by one. If there is some dependence among the different intermediate states, then the summation of measures obtained from the separate models can yield misleading results.

Understandability to the PSA community:

The conditional core damage probability is a widely used secondary risk measure, which is well understood by the PSA community (besides the difficulties arising from the lack of general definition on core damage, see section 2.1.3).

2.4.4 Limitation

With respect to limitation, the same applies to conditional core damage probability as to change in CDF (see section 2.3.4).

2.4.5 ASAMPSA_E recommendation on CCDP

Besides the recommendations related to the risk measures of CDF time dependent (see section 2.1.5), the definition presented in section 2.4.1 is widely used and accepted by the PSA community. Therefore no additional harmonized definition is recommended for the risk measure of conditional core damage probability.

2.5 Conditional Core Damage Frequency (CCDF)

2.5.1 Definition of Risk Measure

Conditional core damage frequency is a secondary conditional consequence risk measure originated by the direct risk measure of core damage frequency. Depending on the boundary conditions of the assessment, this secondary risk measure can be derived from the CDF in a time average and time dependent manner.

Conditional core damage frequency by imposing a set of conditions m_k leading to changes in input parameters $\tilde{p}_n$ (cf. section 2.3). Then

$$CCDF_{m_k}(t) = CDF(t)|\tilde{p}_n$$

Importantly, there is still an initiating event, although its value might be changed. This is the main difference to CCDP discussed above, as this explicitly covers transition probabilities from intermediary states to the consequence (here: core damage).

In several cases, CCDF can be understood by inserting an intermediate state e_k, representing the conditions m_k into the sequence, i.e.

$$s_i \xrightarrow{\varphi(l_{ij})} c_j |\ c_j \in CD \Longrightarrow s_i \xrightarrow{\varphi(l_{ik})|\tilde{p}_n} e_k \xrightarrow{\varphi(t(c_j)|e_k)} c_j |\ c_j \in CD$$

Such a representation is helpful if the condition relates to specific sequences (e.g. event tree sequences with failures of specific safety functions). As with every conditional measure, risk aggregation has to be made with care and often using Bayes' theorem.

In other words, conditional core damage frequency is usually meant by the frequency of significant degradation of the reactor core *(CD)* upon the condition of some system, structure or component unavailability. Besides the unavailability of SSCs, special operating status of the plant can be taken into consideration. Time average CDF risk measure can be used to obtain the time average conditional core damage frequency, while the use of time dependent (instantaneous) CDF risk measure yields the time dependent conditional core damage frequency.

As it can be implied by their definition, that cumulative conditional core damage probability and incremental conditional core damage probability are derived risk measures based on time dependent conditional core damage frequency *(CCDF(t))*. Moreover, the time average risk measure of change in core damage frequency is based on time average conditional core damage frequency (for details see section 2.3).

2.5.2 Areas of application:

Conditional core damage frequency is applied in several PSA application areas, since it reflects, amongst others, the level of risk at a certain time point in a certain condition of the plant. The risk measure of conditional core damage frequency is the typical output of risk monitors, which entails the utilization of this risk measure for other risk measures, e.g.:

- configuration planning (e.g. support for plant maintenance and test activities)
- real time configuration assessment and control (response to emerging conditions)
- dynamic risk-informed technical specifications
- short term risk based performance indicators

2.5.3 Discussion

With respect to all risk measure attributes discussed in similar subsections, the same applies to conditional core damage frequency as to conditional core damage probability (see section 2.4.3).

2.5.4 Limitation

With respect to limitation of the risk measure, the same applies to conditional core damage frequency as to conditional core damage probability (see section 2.4.4).

2.5.5 ASAMPSA_E recommendation on CCDF

With respect to recommending additional harmonized definition for the risk measure, the same applies to conditional core damage frequency as to conditional core damage probability (see section 2.4.5).

2.6 Importance Risk Measures

2.6.1 Definition of Risk Measures

One of the principal activities within a risk-informed regulatory process is the ranking of Structures, Systems and Components (SSCs). It can be performed through the estimation of Importance (and Sensitivity) measures.
In the following, we refer to “traditional” importance measures, including the following ones [28]:

- Fussell-Vesely measure;
- Risk Reduction Worth;
- Risk Achievement Worth/Risk Increase Factor;
- Birnbaum measure;
- Criticality importance measure.

The above measures were originally defined with reference to the probability of the top event of a Fault tree φ, for the individual basic events, but are more generally applicable as secondary risk measures in relation to an underlying risk measure. Therefore, their definitions can be:

- applied to an Event tree-Fault tree model, with reference to the probability of defined undesired consequence $\left(\mu(c_j)\right)$, considering all sequences leading to it $\left(\mu(c_j) = \mu(\cup_i(s_i, c_j))\right)$;
- specified in the general terms of system failure function $(f_{c_j}[x_1, \dots, x_n] = 1)$ when the consequence occurs; $x_1, \dots, x_n$ are the states of the basic events) and by its specific representation through minimal cut sets $(\cup_i(s_i, c_j) \sim \cup_{i=1}^{m} MC_i)$;
- generalized with reference to a direct risk measure different than the probability $\left(\mu(c_j) = \mu\left(\varphi(c_j)\right)\right)$.

It is useful to represent the probability of the undesired consequence as linear function of the basic events probability: $f = a \cdot P_i + b, \text{where } P_i = f(x_i = 1)$. This formulation is strictly correct when basic events are independent [104].

<u>Fussell-Vesely Importance</u>

The Fussell-Vesely importance measure (FV) is the fractional contribution of a given basic event to the probability of the undesired consequence when the basic event probability is changed from its base value to zero (i.e. the basic event never occurs) or equivalently the (conditional) probability that at least one “minimal cut set” containing the basic event occurs (given that the undesired consequence is occurred) [28].
Referring to an individual basic event, the Fussell-Vesely Importance measure is defined as:

$$FV_i = \frac{f - f(P_i = 0)}{f} \sim \frac{f\left(MC_{including\ i}\right)}{f} = \frac{a \cdot P_i}{a \cdot P_i + b}$$

where $f(P_i = 0)$ is the probability of the undesired consequence when the basic event probability is zero.

Risk Achievement Worth / Risk Increase Factor

The Risk Achievement Worth (RAW) measures the "worth" of a given basic event in achieving the present risk level (probability of the undesired consequence in the following), by considering its maximum that is when the basic event always occurred. It indicates the importance of maintaining the current level of reliability for the basic event *i*.

Referring to an individual basic event, the Risk Achievement Worth is defined as:

$$RAW_i = \frac{f(x_i = 1)}{f} = \frac{f(P_i = 1)}{f} = \frac{a + b}{a \cdot P_i + b}$$

where $f(x_i = 0)$ is the probability of the undesired consequence when $x_i = 1$ (i.e. the basic event always occurs).

Risk Reduction Worth / Risk Decrease Factor

The Risk Reduction Worth (RRW) measures the "worth" of a given basic event in reducing the risk level (probability of the undesired consequence in the following), by considering its maximum decrease that is when the basic event never occurs. It indicates the importance of reducing the current level of unreliability for the basic event *i*.

Referring to an individual basic event, the Risk Achievement Worth is defined as:

$$RRW_i = \frac{f}{f(x_i = 0)} = \frac{f}{f(P_i = 0)} = \frac{a \cdot P_i + b}{b} = \frac{1}{1 - FV_i}$$

Birnbaum Importance

The Birnbaum Importance measure (B) is the rate of change in the risk (probability of the undesired consequence in the following) as result of the change in the probability of a given basic event, or equivalently the difference in the probability of the undesired consequence when the basic events always occurs and never occurs, or equivalently the probability to be in a "critical" status for the particular basic event (i.e. the undesired consequence occurs only if the basic event occurs).

Referring to an individual basic event, the Birnbaum Importance is defined as:

$$B_i = \varphi(x_i = 1) - \varphi(x_i = 0) = \varphi(P_i = 1) - \varphi(P_i = 0) = \frac{\partial \varphi}{\partial P_i} = a = RAW_i + RRW_i$$

Criticality Importance

The criticality importance (C) measure is the (conditional) probability that the undesired consequence occurs because of the occurrence of a particular basic event (given that the undesired consequence occurs):

$$C_i = \frac{\big(f(P_i = 1) - f(P_i = 0)\big) \cdot P_i}{f} = \frac{a \cdot P_i}{a \cdot P_i + b} = FV_i$$

2.6.2 Areas of application

Generally speaking, SSCs can be ranked with respect to their "risk-significance" and "safety-significance", providing complementary ways to identifying their role [28]. Conceptually, a risk-significant ranking is related to the role that the SSC plays in the current level of risk and the prevention of the occurrence of the undesired consequence.

Even if relationships exist among the above traditional importance measures, they provide some complementary information. It is commonly recognized that the Risk Achievement Worth produces a safety-significant ranking, while all the remaining ones produce risk-significant ones.
In many applications, only one risk-significance importance measure could be sufficient. To describe the influence of the SSCs exhaustively, the relevant basic events can be ranked through a “two-dimensional” criterion, by estimating a risk-significant measure (e.g. FV) and a safety-significant one (RAW). The concurrent use of two measures is advisable, even if the obtained results – in terms of SSCs ranking – could be less obvious.

2.6.3 Discussion

Validity:

The traditional importance measures are introduced with reference to the probability φ of a defined undesired consequence, as basic direct risk measure. The same definitions apply to a generic direct risk measure

$$\mu = \mu\left(\varphi(c_j)\right) = \mu\left(\bigcup_i (s_i, c_j)\right) = \mu\left(f_{c_j}[x_1, \dots, x_n] = 1\right) \sim \mu\left(\bigcup_{i=1}^{m} MC_i\right).$$

Traditional importance measures are addressed by a number of scientific publications and guidelines and are widely used in the existing PSA of NPPs. Their estimation is supported by a number of software tools, typically based on minimal cut-sets to solve the probabilistic model.

The Fussell-Vesely and the Risk Achievement Worth (RAW) are the most widely used importance measure. Their contextual use could provide complementary insights, as previously indicated.

About the use of RAW, because of its extreme nature, it is likely that the safety-significant SSCs would be a large set. About the Birnbaum measure, it’s useful to remark its relations with the Differential Importance Measures and with the linear regression method for sensitivity analysis (introduced in the following sections).

Reliability and Consistency:

The traditional Importance measures are clearly defined. Different formulations are possible, maintaining consistency and assuring their reproducibility. Simple mathematical relationships hold among these importance measures at the individual basic event, allowing their indirect computation. These relationships also allow computing different measures (e.g. the Differential Importance measure introduced in the following) without additional evaluations of the model.

Although the basic philosophy is consistent and mathematical formulas are defined coherently, some inconsistency could be introduced in the calculation of the Importance measures. Indeed, the values obtained for the measure by setting to “true“ or “false“ the variables (binary state of the basic events) and solving the probabilistic model could not coincide with the values obtained by setting the basic event probabilities equal to their extreme values (0, 1).

Risk aggregation properties:

The SSCs ranking may require being able to consider many basic events as a part of a group. For instance: a particular SSC may be represented in the model by several basic events, which represent different failure modes; the analyst is interested in the ranking of different typologies of SSCs, whose basic events are in different “parts” of the model. In this regard, as main limitation of the traditional Importance measures (see the following paragraph), they are not “additive”: the measure for a group of input variables cannot be computed as the sum of the measures estimated for each single variable.

Understandability to the PSA community:

Being proposed and reviewed by a number of scientific publications and used in a number of PSA applications, the understandability of the traditional importance measure is not considered a major concern. Anyway, some limitations discussed in the following paragraph, if not well understood, could lead to some misunderstandings about the interpretation of the ranking produced by the measures.

Additional difficulties in the interpretation of results could exist in the concurrent use of risk-significance and safety-significance measures.

It is useful to remark that the above importance measure shall be considered as “relative” ones. As consequence, the comparison of results coming from Importance analyses developed for different plants shall be performed carefully or avoided. The use of a single value for the adopted measure as a “universal” criterion to screen for significance means, i.e. to establish group membership for SSCs (significant or non-significant), can lead to inconsistent SSCs ranking for different plants. Indeed, setting a fixed threshold for risk/safety-significance, the contributions of the same basic events are different for plants having different direct risk measure (e.g. CDFs or LERFs) [28].

2.6.4 Limitation

As above remarked, the traditional Importance measures are not “additive”. Their estimation for a group of variables requires new evaluations of the model (e.g. new selections among the minimal cut-sets).

The traditional Importance measures strictly apply to binary coherent systems/models. For non-coherent systems, whose non-monotonic system failure function is represented by the “prime implicant” sets (minimal combinations of basic event - in normal and negated forms - leading to the undesired consequence), some generalizations of the importance measures can be defined in order to account for the criticality of the occurrence and non-occurrence of the event separately. Anyway, the use of minimal cut sets as approximated form obtained by removing negated events from the prime implicant sets of a non-coherent system/model, leads to conservative results, facilitates the interpretation of system failure modes and allows a significant reduction of computation time and working memory space.

The traditional importance measures are “local” ones, meaning that they deal with point values and “small” changes of the input variables. They cannot be used in order to account for their finite changes or, in this case, they do not include the contributions of non-linear terms. These non-linear terms represent the “interactions” among input variables, whose effects are manifested for their simultaneous changes and are not taken into account by the super-imposition of the effects due to the One-At-Time (OAT) change of variables.

The traditional importance measures assume that linear relations exist between the probability of the undesired consequence and the probability of the basic events. Indeed, the measures for the basic events can be computed starting from the extreme values of their probability. This limitation is particularly significant if the measure is referring to the parameters of the model. In this case, the probability distributions (typically exponential ones) for basic events introduce non-linear terms that are not accounted for.

The uncertainty on the input variables of the model (basic events probabilities or relevant parameters) makes it difficult to determine a robust ranking of SSCs through the traditional importance measures. The typical approach is to represent probabilistically this uncertainty and to compute the importance measures in terms of probability distributions, e.g. by means of sampling techniques. It could lead to the impossibility to define a unique raking of SCCs because of overlaps among the probability distributions of the measures for different events. Otherwise, different approaches shall be used within an importance and sensitivity analysis framework (as discussed in the following).

Several tools for the solution of Fault Trees/Event Trees model are based on a common broadly accepted scheme: (i) event tree sequences (and linked fault trees) are transformed into Boolean formulae; (ii) minimal cut-sets of these formulae are determined; (iii) various probabilistic measures are assessed from the cut-sets (including secondary risk measures). However, this approach is based on some hypotheses to be fulfilled and relevant approximations: the “rare event” hypothesis introduces approximations mainly due to the dependences among minimal cut-sets; in order to minimize cut-sets, and therefore avoiding combinatorial explosion, truncation criteria are applied; in order to handle success branches, various procedures more or less mathematically justified are used [103]. The use of the Binary Decision Diagram (BDD), being based on the Shannon decomposition formula, allows overcoming this limitation, providing an exact solution of the model in terms of combination of disjoint “paths” among the variables, in their normal or negated forms (i.e. for coherent and non-coherent systems) [106]. BDD also allows reducing the effort for the computation of the importance measures [105]. Unfortunately, the full conversion of large fault trees into BDDs could remain out of reach in terms of computational resources, because of the size, non-coherency, redundancy, and complexity of the model. A potential solution is to design hybrid algorithms that combine the approximations due to the cut-offs introduced on the minimal cut sets probability (and/or order) and the exact solution through BDD applied to a “simplified” fault tree [103].

2.7 Differential Importance Measures

2.7.1 Definition of Risk Measure

As introduced in the Appendix A (Section 8), the probability of the undesired consequence $f(X): \mathbb{R}^n \rightarrow \mathbb{R}$ (or a different direct risk measure) can be written by its Taylor series representation (§ 9.5.1). Starting from it, the differential importance measure [95], the joint importance measure [96] and the total order differential importance measure [97] are introduced.

<u>First Order Differential Importance measures</u>

The differential importance measure (DIM) is the fraction of the total change of the risk measure due to one-at-time “small change” of the input variables (basic events probability) [95]:

$$DIM_i = \frac{\frac{\partial f}{\partial P_i} dP_i}{\sum_{j=1}^{n} \frac{\partial f}{\partial P_j} dP_j} = \frac{B_i \Delta_i}{\sum_{j=1}^{n} B_j \Delta_j}$$

where:

- B_i is the Birnbaum importance for the basic event i;
- $\Delta_i = 1$ under the hypothesis of "uniform changes" of the basic events probability ($dP_i = dP_j \; \forall i, j$);
  $\Delta_i = Pi$ under the hypothesis of "uniform percentage changes"[7] of the basic events probability ($\frac{dP_i}{P_i} = \frac{dP_j}{P_j} \; \forall i, j$).

Joint and Total Order Differential Importance Measure

Generally, the Taylor series representation requires an infinite number of terms to represent exactly the model output. It can be proved that the failure probability of any (coherent and non-coherent) system, coming from a system failure function represented by a Boolean equation, is a multi-linear function of the failure probability of its components[8].

It follows that its Taylor series representation has a finite number of terms, allowing the introduction of a measure related to the "total change" of the model output [96].

The total order differential Importance measure for the basic event i is the fraction of the total change of the f that is due to the change of P_i, alone and together with the changes of the remaining P_j $(j \neq i)$, in any number and combination:

$$D_i^T = \frac{{}^T\Delta\varphi_i}{{}^T\Delta\varphi} = \frac{B_l \Delta P_i + \sum_{k=2}^{m} \sum_{\substack{l<\cdots<k \\ i \in l,\dots,k}} \left({}^kJ_{l\dots k} \cdot \prod_{s=1}^{k} \Delta P_s \right)}{\sum_{l=1}^{n} (B_l \Delta P_l) + \sum_{k=2}^{m} \sum_{\substack{l<\cdots<k \\ i \in l,\dots,k}} \left({}^kJ_{l\dots k} \cdot \prod_{s=1}^{k} \Delta P_s \right)}$$

where:

- $B_i = \frac{\partial f}{\partial P_i}$ is the Birnbaum measure for the basic event i;
- ${}^kJ_{l\dots k} = \frac{\partial^k f}{\partial P_1 \cdot \ldots \cdot \partial P_k}$ is the "joint importance of k-order" and gives information about how the basic events $l \dots k$ "interact", i.e. how their simultaneous change modify the model output;
- $\sum_{l<\dots<k} \cdots = \sum_{l=1}^{n} \sum_{j=l}^{n} \cdots \sum_{m=\dots}^{n} \sum_{k=m}^{n} \cdots$

2.7.2 *Areas of application*

The DIM is a risk-significance measure, which refers to the first order approximation of the Taylor series representation. It has been introduced by a number of years in the scientific literature and, as previously noted, it could be computed starting from the traditional Importance measures (as post elaboration of the results coming from the available software tools). It provides remarkable improvements with respect, for instance, to the Birnbaum measure, first of all the additivity of the measure and its definition within a framework (Taylor series representation of the primary risk measure) which allows the consistent introduction of further measures able to assess the interactions among variables, which are not accounted by the traditional importance measure.

[7] The uniform percentage changes shall be assumed when the input variables have different measure units.
[8] The system unreliability is not a multi-linear function of the parameters that define the failure (and repair) probability distributions of components, i.e. basic events.

The joint importance measure and the total order differential importance measure are relatively new and probably never used in the existing PSA of NPP. Anyway, scientific papers address their potentialities and limitations and provide a number of examples of applications.

The total order differential importance measure refers to the influence of a basic event as result of its individual effect and of all possible interactions with the other basic events. It combines in a unique measure the information provided by the Birnbaum measure and by the joint importance measures of any order.

For a "small enough" (i.e. differential) change of the input, the total order differential importance measure coincides with the first order differential importance measure ($DIM_i = D_i^T$).

It is remarkable that it opens the possibility to investigate the interactions among the basic events of the PSA model (i.e. among SSCs). For instance, the estimation of the (joint and the) total order differential importance measure(s) could support the identification of potential dependent failures: the higher the significance of interactions among a set of variables, the higher is the potential impact on the risk if credible common root causes exist. This could extend the evaluations beyond the assessment of the common-cause groups identified beforehand (e.g. redundant items performing the same function), allowing for the identification of "latent" dependencies (not obvious in large models, specifically for different typologies of SSCs, e.g. different SSCs implementing different lines of protection but vulnerable to the same cause - e.g. internal flooding).

2.7.3 Discussion

Validity:

The differential importance measures have a very general scope. With reference to PSA applications, they can be referred to the basic events (as in the above definitions), as well as to the parameters of the model (which typically define the probability distributions of the basic events). However, only in the first case there is a simplified procedure to estimate the total order differential importance measures without computing each one of its terms.

The DIM provides information about the "main" (i.e. first order) contribution of each input variable. The joint importance measure provides information on the interactions about a specific group of input variables. The total order differential importance measure provides information that includes the contribution of the interactions between the variable at issue and all the remaining on in any number and combination.

Generally, both the uniform changes and uniform percentage changes assumptions can be adopted. The second one shall be adopted if the parameters have not the same measure unit.

Reliability and Consistency:

The basic philosophy and the mathematical formulas are consistent, as well as the relations with the other Importance measures, specifically with the traditional importance measures (for the first order differential measure) and with the finite change sensitivity measures (which are based on a comparable framework but starting from a different representation of the model, i.e. HDMR (high dimensional model representation) instead of Taylor series).

Risk aggregation properties:

The differential importance measures, being based on a representation of the model output which is a sum of terms depending on an increasing number of variables, are intrinsically additive if related to basic events.

Rigorously, however, only the DIM is an additive measure: the measure for a group of variables (basic events probability or relevant parameters) is equal to the sum of measures computed for each one of them and can be estimated without additional evaluations of the model. For instance, the DIM for the pair of basic events *i* and *j* is:

$$DIM_{i,j} = \frac{\frac{\partial f}{\partial P_i} dP_i + \frac{\partial f}{\partial P_j} dP_j}{\sum_{k=1}^{n} \frac{\partial f}{\partial P_k} dP_k} = DIM_i + DIM_j$$

Conversely, all the higher order measures for a group of variables cannot be estimated as sum of the measures computed for single or subgroups of variables, and requires further computations because new interactions terms are introduced.

Understandability to the PSA community:

The interpretation of the ranking provided by the DIM is substantially the same of the risk-significance traditional importance measures. The ranking produced by the total order differential importance measures, which includes the effects of the interactions among the variables, provides different information. Its correct interpretation requires the understanding of the whole framework.

The knowledge of the first and total order measures provides information on the local and global significance of each input variable (i.e. with reference to the nominal point value and to the whole range of variability) and on the whole effects of its interactions with the remaining variables, in any number and combination. It should be sufficient for PSA applications.

The estimation of all joint importance measures of k-order – if possible despite the required effort - provides an abundance of information which may be difficult to interpret. Alternatively, they could be computed just for a reduced number of (groups of) variables suspected to have significant interactions with the other ones (e.g. having a significant total order differential importance measures although DIM is not so relevant), as second-level of investigation.

2.7.4 Limitation

The DIM, as well as the traditional importance measures, is a "local" importance measure, dealing with point values and "small" changes of the input variables. It cannot be used in order to account for their finite changes or, in this case, they do not include the contributions of non-linear terms.

Without looking at computational cost, a brute force approach could be applied in order to compute all joint importance measures and with them all terms within the total order differential importance measure. For PSA models of a realistic size, the relevant computational effort for PSA applications is too onerous. In fact, the first and higher order partial derivatives of the direct risk measure with respect to all combinations of the input variables have to be computed. Even if the differential importance measures are applied for truncated cut set lists, the resulting combinatorics are prohibitive for current PSA codes. The effort increases if the measures refer to the parameters of the model.

The effort required to compute the total order differential importance measure can be significantly reduced when it refers to the basic events probability (rather than to parameters). In this case, it coincides with the total order finite change sensitivity measure and they can be computed through the same procedure (introduced in the following) by means of $n+2$ evaluations of the model. For a "small enough" (i.e. differential) changes of the input variables, the total order and the first order differential importance measures coincide $\left(D_i^T = DIM\right)$. Therefore, this procedure can be applied for the computation of the (first order) DIM for basic events (as alternative approach to the preliminary estimation of the traditional Importance measures, e.g. Birnbaum). Using truncated cut set lists reduces the number of basic events, which have to be considered for this evaluation, to a certain extent.

2.7.5 ASAMPSA_E recommendation on Differential Importance Measures

The definitions presented above are judged to be standard and state-of-the-art. No specific harmonization is found to be needed.

2.8 Linear Regression Method for Sensitivity Measures

2.8.1 Definition of Risk Measure

The random variable $\mathrm{X} = \mathrm{x}_1, \dots, \mathrm{x}_\mathrm{n}$ of the direct risk measure $f(X)\colon \mathbb{R}^n \to \mathbb{R}$ can be affected by uncertainty. Consequently, the model output will be affected by uncertainty represented by a probability distribution. From a general point of view, the sensitivity analysis aims at quantifying the contributions of the uncertainty on the input variables to the uncertainty on the model output (direct risk measure).

The different approaches for sensitivity analysis can be classified into two main branches [101]:

- Local analysis, which is focused on the point values of the input variables (in the sense previously used for Importance measures);
- Global analysis, which is focused on the entire range of values of the input variables.

A traditional approach for the sensitivity analysis is the development of a linear regression model for the model output (direct risk measure, i.e. probability of undesired consequence in the following):

$$\hat{f} = a_0 + \sum_i a_i x_i$$

The uncertainty on the input variables is represented by probability or frequency distributions.

For linear models (or with reference to the first order approximation of the Taylor series representation) the following "standardized regression coefficients" can be defined and used as importance measure:

$$\beta_i = \bar{a}_i \frac{Var(x_i)^{\frac{1}{2}}}{Var(\varphi)^{\frac{1}{2}}} = \frac{\partial f}{\partial x_i} \cdot \frac{Var(x_i)^{\frac{1}{2}}}{Var(\varphi)^{\frac{1}{2}}} \quad i = 1, \dots, n$$

For linear models $\sum \beta_i = 1$, while for non-linear ones $\sum \beta_i < 1$.

Under the same assumptions, the square of the Standardized regression coefficients can be used as sensitivity measure:

$$\beta_i^2 = \left(\frac{\partial f}{\partial x_i}\right)^2 \cdot \frac{Var(x_i)}{Var(\varphi)}$$

2.8.2 Areas of application

The Regression method provides an algebraic representation of relations between the output of the model (direct risk measure) and (one-at-time) input variables.

Complementing information is provided by the Standardized regression coefficients and its square.

The standardized regression coefficient combines a term focused on the point value of the input variable (i.e. partial derivative, i.e. Birnbaum measure) with a term focused on the whole range of variability (i.e. ratio between the square root of variances on the input variable and the model output).

The square of the standardized regression coefficient provide information on the propagation of the uncertainty through the model, which depends on the square of the partial derivatives.

2.8.3 Discussion

Validity:

The regression method provides measures able to account for the uncertainty associated to the input variables, which is represented by a normal probability distribution and then characterized by the second central moment (variance). It could be the result of the assessment of the (epistemic) uncertainty on the input variables, or just as fictitious uncertainty introduced to calculate the sensitivity measures.

Reliability and Consistency:

The basic philosophy and the mathematical formulas are consistent, simple and easy to be implemented. Typically, sampling techniques are used to generate the sets of values of the input variables; the value of the model output is computed for each input set; the “Not standardized coefficients” parameters a_i can be computed, for instance, by means of the “least square approach”.

The “efficiency” of β_i and β_i^2 as importance and sensitivity measures can be estimated through the so-called “coefficient of determination” of the linear regression. It is the ratio between the variance on the model output explained by the linear regression and the variance on the sampled data.

$$R_U^2 = \frac{\sum_j \left(\hat{f}^j - \bar{f}\right)^2}{\sum_j \left(f^j - \bar{f}\right)^2} = \frac{Var(y_{regr})}{Var(y)}$$

It results $0 \le R_U^2 \le 1$. Specifically, R_U^2 is close to 1 when the linear regression model takes into account most of the uncertainty on the model output.

Risk aggregation properties:

The above measure is not additive. Its estimation for a group of input variables requires the development of a multi-regression analysis, or at least the re-coding of variables into a single fictitious variable singularly considered.

Understandability to the PSA community:

Although the results of a linear regression model is easy to understand in mathematical terms, the information encoded in the importance and sensitivity measures defined above could be difficult to interpret because they mix local (partial derivatives) and global (ratio between variances of input and output) information into single measures.

2.8.4 Limitation

Obviously, the measures defined by the linear regression method assume that linear relations exist between the probability of the undesired consequence and the probabilities of the basic events or anyway neglect non-linear terms. The standardized regression coefficient shows the same limits of the traditional importance measures previously discussed. Nonlinear regression provides an alternative approach but a major challenge is the determination of a suitable form for the regression model. A rank transformation can be used to convert a nonlinear but monotonic relationship between the input and output variables into a linear one, but will not provide information on the original nonlinear aspect.

The method requires the assignment of uncertainty for each input variable. The propagation of this uncertainty through the model provides insights on its structure which are not accounted for by the traditional importance measures (e.g. it depends on the square of the partial derivatives of the model output). Nevertheless, the measure is not able to account for and does not provide insights on the interactions among variables, which are manifested when variables change at the same time in their range of variability.

A general limitation concerns the use of normal distribution to represent – through the second central moment (variance) - the uncertainty on the input variables, which could be not the optimal one. Indeed, it is generally recognized that log-normal distribution better represents the (epistemic) uncertainty, which should be associated to the basic parameters of the model.

Moreover, it is always useful to remark that when limited information is available to characterize uncertainty, probabilistic characterizations can give the appearance of more knowledge than is really present. Alternative representations for uncertainty such as Evidence theory and Possibility theory merit consideration. In order to investigate the “structure” of the probabilistic model by the propagation of uncertainty, from the input variable to the model output, the same variance could be assigned to all the input variables.

2.8.5 ASAMPSA_E recommendation on harmonized definition

The definitions presented above are judged to be standard and state-of-the-art. No specific harmonization is found to be needed.

2.9 “Finite Change” approach for Linear Regression Method for Sensitivity Measures

2.9.1 Definition of Risk Measure

Considering a finite change of the input variables, from an initial value $x^0 \in X$ to a final value $x^1 \in X$, the corresponding change in the output can be written as $\Delta f = f(x^1) - f(x^0)$ where $f(x_i^0) = f(x_1^0, \ldots, x_i^0, \ldots, x_n^0)$ and $f(x_i^1) = f(x_1^0, \ldots, x_i^1, \ldots, x_n^0)$.

It is obvious from this definition, that this secondary risk measure is closely related to the ΔCDF risk measure discussed in section 2.3.

Starting from HDMR representation, the change of the model output (probability of undesired consequence or different primary risk measure) can be written as:

$$\Delta f = \sum_{i=1}^{n} \Delta_i f + \sum_{i=1}^{n} \sum_{j=i+1}^{n} \Delta_{ij} f \; + \ldots + \Delta_{1\ldots n} f = \sum_{k=1}^{n} \sum_{j_1 < \ldots < j_k}^{n} \Delta_{j_1 \ldots j_k} f$$

where:

$$\begin{cases} \Delta_i f = f\left(x_1^0, \ldots, x_i^1, \ldots, x_n^0\right) - f(x^0) \\ \Delta_{ij} f = f\left(x_1^0, \ldots, x_i^1, x_j^1, \ldots, x_n^0\right) - \Delta_i f - \Delta_j f - f(x^0) \\ \ldots \end{cases}$$

The output change Δf can thus be decomposed into 2^n-1 terms depending on an increasing number of variables: the first order terms $\Delta_i f$ consider the contributions due to the one at time change of the input variables, the second order terms $\Delta_{ij} f$ consider the additional contributions due to the interaction between all variables pairs (i.e. due to their concurrent changes), and so on.

Starting from the above decomposition of the finite change of the model output, the following measures can be defined.

<u>First Order Finite Change Sensitivity measure</u>

The "first order finite change sensitivity measure" is the contribution to the change $\Delta\varphi$ of the finite change of a single variable, its normalized version being the corresponding fraction of the change:

$$\rho_i^1 = \Delta_i f \qquad\qquad \Gamma_i^T = \frac{\Delta_i f}{\Delta f} = \frac{f\left(x_i^1\right) - f\left(x^0\right)}{f\left(x^1\right) - f\left(x^0\right)}$$

For a model with n input variables (i.e. $X \in \mathbb{R}^n$), the number of model evaluations required to compute the first order finite changes sensitivity index is $n + 2$, being $f(x^1)$, $f(x^0)$ and $f\left(x_i^1\right)$ to be estimated.

<u>Order k Finite Change Sensitivity measure</u>

The "order k finite change sensitivity measure" is the contribution to the (finite) change $\Delta\varphi$ of the interactions among (the first) k variables $x_1, x_2, \ldots, x_k$, its normalized version being the corresponding fraction of the change:

$$\rho_{i_1,\ldots,i_k}^k = \Delta_{j_1\ldots j_k} f \qquad\qquad \Gamma_{i_1,\ldots,i_k}^k = \frac{\Delta_{j_1\ldots j_k} f}{\Delta f}$$

<u>Total Order Finite Change Sensitivity measure</u>

The "total order finite change sensitivity measure" is the contribution to the (finite) change $\Delta\varphi$ of the (finite) change of the variable at issue, alone and together with the changes of all remaining variables in any number and combination, its normalized version being the corresponding fraction of the change:

$$\rho_i^T = \Delta_i f + \sum_{j \neq i} \Delta_{ij} f + \ldots + \Delta_{1\ldots n} f = \sum_{k=1}^{n} \sum_{i \in j_1 \ldots j_k} f_{j_1,\ldots,j_k}^k$$

$$\Gamma_i^T = \frac{\Delta_i f + \sum_{j \neq i} \Delta_{ij} f + \ldots + \Delta_{1\ldots n} f}{\Delta f} = \frac{\sum_{k=1}^{n} \sum_{i \in j_1 \ldots j_k} f_{j_1,\ldots,j_k}^k}{\Delta f}$$

2.9.2 Areas of application

The importance and sensitivity analysis could (and should) be considered as a unique task that includes the computation of different measures, which provide complementary information to the decision maker, concerning the contributions of each single variable to the value of the model output (importance analysis) and to the relevant uncertainty (sensitivity analysis). In this regard, the importance analysis has significant overlap with the local sensitivity analysis.

As pointed out above, different approaches for sensitivity analysis can be classified local and global ones.

The finite change sensitivity measures (as well as the variance based approach introduced in the following) allow a global importance and sensitivity analysis, being considered the entire range of values of the input variables, and providing information on their contributions to the model output and to the relevant uncertainty. Specifically, the finite change sensitivity measures allow ranking input variables through a non-parametric approach, i.e. without the need to specify probability distributions for the relevant uncertainty, but just their ranges of variability.

2.9.3 Discussion

Validity:

The “finite change” approach for the importance and sensitivity analysis allows the apportionment of the (finite) change of the model output into the contributions due to the individual and simultaneous (finite) changes of input variables.

Although it has been only recently proposed and probably never used in NPP PSA, the consistency with other secondary measures, the possibility to overcome the computational limits of other approaches for global sensitivity analysis (as for the variance-based approach introduced in the following) and to avoid the specifications of a probability distributions representing the uncertainty on the input variables, make this approach very attractive.

The total order finite changes sensitivity measure, when referring to basic events probability, coincides with the total order differential importance measure and both coincide with the first order differential importance measure when the finite changes become “small enough” (i.e. differential ones). This “reconciles” the changes sensitivity measures with the traditional importance measure already used in PSA applications.

Reliability & Consistency:

The finite changes sensitivity measures are introduced consistently with the HDMR representation of the model output (direct risk measure) and then intrinsically recognize the presence of terms depending on a number of variables interacting among themselves.

Although based on a “sophisticated” representation of the direct risk measure (HDMR), the formulas to be used for the computation of the first order (to be used for a local perspective) and total order (to be used for a global perspective) finite changes sensitivity measures for basic events are very simple and require $2n + 2$ evaluations of the model (according to the procedure introduced in the following).

Risk aggregation properties:

The finite change sensitivity measures, being based on a representation of the model output which is a sum of terms depending on an increasing number of variables, are intrinsically additive.

Rigorously, however, only the first order finite change sensitivity measure is an additive measure: the measure for a group of variables (basic events probability or relevant parameters) is equal to the sum of measures computed for each one of them and can be estimated without additional evaluations of the model.

Differently, all the higher order measures for a group of variables cannot be estimated as sum of the measures computed for single or subgroups of variables, and requires further computations because new interactions terms are introduced.

Understandability to the PSA community:

The interpretation of the ranking produced by the first order finite change sensitivity measures is substantially the same of the DIM and of the risk-significance traditional importance measures. Considering the whole range of the basic events probability $[0; 1]$, the first order finite change sensitivity measure coincides with the Birnbaum measure, being neglected non-linear terms of the model output.

From a general point of view, sensitivity indices give information about the direction of change of the model output due to individual or simultaneous changes of the input variables (not interested for coherent systems), the key-drivers of the change of the model output (direct risk measure) and the structure of the model (i.e. the relevance of interactions).

The magnitudes of the Γ_i^T allow the identification of the key drivers of the model output change, i.e. the variables whose change – alone and together with the changes of the remaining variables – determines the larger contributions to the change of the model output.

Information about the structure of the model is provided by the magnitudes of Γ_i^1 and all $\Gamma_{i_1,\dots,i_k}^k$ with any other variables, in any combination. If the complete decomposition is not achievable due to the required computational effort, the differences $\Gamma_i^T - \Gamma_i^1$ can be taken as indicators of the relevance of interactions among variables:

- if $\Gamma_i^T - \Gamma_i^1 \approx 0$ the effects of the interactions involving the variable i are irrelevant;
- if $|\Gamma_i^T - \Gamma_i^1| \gg |\Gamma_i^1|$ the relevance of the input variable is mainly attributable to its “cooperation” with the others, rather than to its individual effect.

2.9.4 Limitation

The use of a non-parametric approach for the representation of the uncertainty on the input variables can be a limitation when a uniform probability distribution over the entire range of variability introduces an inappropriate bias for the extreme values (if reasonably less probable).

For a model with n input variables (i.e. $X \in \mathbb{R}^n$), the computation of all order finite changes sensitivity measures can be performed directly from the definitions provided above, requiring $\sum_{i=1}^{n}\binom{n}{i} = 2^n$ model evaluations. Frequently for PSA applications it is computationally too onerous.

As already remarked for the differential importance measures, the knowledge of the first and total order measures provides information on the local and global significance of single input variables and on the whole effects of its interactions with the remaining variables, in any number and combination.

Starting from the evidence that the direct risk measure is a multi-linear function, being the system failure function represented by a Boolean equation, the total order finite changes sensitivity measure for the basic events (which coincide with the total order differential importance measure) can be computed through the equation [96]:

$$\Gamma_i^T = \frac{f(x^1) - f(x_{-i}^1)}{f(x^1) - f(x^0)}$$

where:

- x^1 and $x^0$are the initial and final values of the variable;
- $x_{(-i)}^1$ is the point obtained by shifting all parameters at their final value but x_l, which is at its initial value, i.e. $\left(x_{(-l)}^1 = x_1^1, x_2^1, \ldots, x_{l-1}^1, x_l^0, x_{l+1}^1, \ldots, x_n^1\right)$.

Through this procedure, the number of model evaluations required to compute the first and total order finite changes sensitivity measures is $2n + 2$, requiring the evaluation of $f(x^1)$, $f(x^0)$ $f(x_i^1)$ and $f(x_{(-i)}^1)$.

2.9.5 ASAMPSA_E recommendation on "Finite Change" approach for Linear Regression Method for Sensitivity Measures

The definitions presented above are judged to be standard and state-of-the-art. No specific harmonization is found to be needed.

2.10 Variance Based approach for Sensitivity Measures

2.10.1 Definition of Risk Measure

The "variance based" approach for sensitivity analysis is based on the HDMR representation of the model output (i.e. probability of undesired consequences or different direct risk measure).

This parametric approach is based on the use of the normal probability distribution to represent – by the second central moment - the uncertainty on the input variables and the model output. The variance on the model output is apportioned into the contributions due to the variance on the input variables.

Sensitivity indices based on the "variance-decomposition" are introduced in the following.

<u>Sobol Sensitivity Indices</u>

Starting from the HDMR representation of the direct risk measure, the related variance can be written as the sum of terms (partial variances) depending on an increasing number of variables:

$$V = \sum_{i=1}^{n} V_i + \sum_{i=1}^{n} \sum_{j=i+1}^{n} V_{ij} + \ldots + V_{1\ldots n}$$

where:

- $V = Var(f(X)) = \int f^2 \cdot p(X)dX - f_0^2$
- $V_{1\ldots m} = Var(f_{1\ldots m}(x_1, \ldots, x_m)) = \int f_{1\ldots m}^2(x_1, \ldots, x_m) \cdot \prod_{k=1}^{m} p_k(x_k)\, dx_k$

Sobol Sensitivity indices are defined as the ratios between the partial variance due to the variables at issue and the total variance on the model output [101]:

$$S_{1\ldots m} = \frac{V_{1\ldots m}}{V}$$

All the terms $S_{1\dots m}$ are non negative and their sum is equal to one $\sum_{k=1}^{n}\sum_{j_1<\dots<j_k}^{n} S_{j_1\dots j_k} = 1$.

For each basic event, the term $\mathrm{S_i} = \frac{V_i}{V}$ is named "Main Sensitivity Index".

Generally $\sum \mathrm{S_i} \leq 1$; specifically, the sum of the Main Sensitivity indices is equal to 1 for "additive[9]" models.

Global sensitivity index

With reference to the Variance decomposition, the Global Sensitivity Index for the input variable i represents the fraction of variance on the model output that is explained by the input variable i, alone and together with all the remaining variables, in any number and combination. It is defined as:

$$S_i^T = S_i + \sum_i \left[\sum_j S_{ij} + \dots + S_{1\dots n} \right]$$

2.10.2 Areas of application

Generally speaking, a sensitivity analysis could be performed for a number of reasons, including the needs to determine which input variables mainly contribute to the output variability and which ones have significant interactions to be accounted for. Other reasons – less relevant to PSA applications - could refer to the needs to determine which parameters can be eliminated from the final model because insignificant and if all observed effects can be physically explained.

The Variance-based sensitivity analysis has a very general scope and is versatile and effective to support the formulation of appropriate answers for all the above questions.

Its application is specifically suggested when non-linearity in the model is significant and shall be "captured" by the adopted secondary measures.

It could be not the case of PSA applications, specifically for Level 1 PSA and particularly when the analysis is referred to the basic events probability, being the System failure function a Boolean equation. Differently, if Level 2 PSA includes physical models for the phenomena into the containment, the study of the significant variables and the uncertainty analysis could be effectively supported by the Variance-based sensitivity indices.

2.10.3 Definition of Risk Measure

Validity:

As previously explained, variance-based sensitivity analysis is a form of global sensitivity analysis. Within a probabilistic framework, the variance of the output of the model is decomposed into fractions which can be attributed to input variables or to sets of input variables, accounting for the contributions of their single and concurrent variations.

[9] A model $\varphi(\mathrm{x_1}, \dots, \mathrm{x_n})$ is additive if it can be decomposed in sum of n functions, each dependent on a single variable $\mathrm{x_i}$. $\varphi = \sum \mathrm{x_i^2}$ is a non-linear but additive model; $\varphi = \prod \mathrm{x_i^2}$ is a non-linear and non-additive model.

A number of applications have been developed, in a number of application fields. The validity of the approach, despite its limitations mainly regarding the use of normal distributions to represent uncertainty and the effort required for the computation of all order terms, is unquestionable.

Reliability & Consistency:

The variance-based sensitivity indices are introduced consistently with the HDMR representation of the model output (direct risk measure) and then intrinsically recognize the presence of terms depending on a number of variables and then of interactions among them.

Different numerical approaches have been proposed for the computation of the Variance-based sensitivity indices. Some discrepancies could exist in their numerical results. A calculation method which is not computationally suitable, although correct, can give incorrect results.

Methods for the uncertainty propagation and for the computation of the sensitivity indices include the solution of multi-dimensional integrals by sampling-based methods (Monte Carlo or quasi-Monte Carlo, Latin hypercube sampling) and the application of the Fourier transform on a space filling curve in the input space [100], [101]. Sampling-based methods require the computation of the model output for different sets of values of the input variables. An efficient and parsimonious procedure can be adopted for the computation of the main and global sensitivity indices [102]. The Fourier amplitude sensitivity test (FAST) is more efficient than methods based on sampling techniques, although it is usually limited to the computation of the main and total effects[10].

Risk aggregation properties:

The variance-base sensitivity indices, being based on a representation of the model output which is a sum of terms depending on an increasing number of variables, are intrinsically additive.

Rigorously, however, only the main sensitivity index is an additive measure: the index for a group of variables (basic events probability or relevant parameters) is equal to the sum of indices computed for each one of them and can be estimated without additional evaluations of the model.

Conversely, all the higher order indices for a group of variables cannot be estimated as sum of the measures computed for single or subgroups of variables, and requires further computations because new interactions terms are introduced.

Understandability to the PSA community:

The main sensitivity index S_i has a clear interpretation, being the fraction of the variance on the primary risk measure that is “explained” by the variance of each individual input variable (i.e. the reduction of the variance on the model output when the input variable *i* is fixed to its nominal value).

It is useful to remark that the product between the square root of the main sensitivity index and the Birnbaum measure provides the standardized regression coefficients introduced with the linear regression method.

[10] The relationship between FAST and Sobol sensitivity indices was revealed in the general framework of HDMR decomposition [101].

The quantity $1-\sum S_i$ is the fraction of V "explained" by the "interactions" among all the input variables, in any number and combination, whose effects are manifested for the simultaneous changes of input variables and are not taken into account by the super-imposition of the effect due to the OAT changes of variables.

The quantity $S_i^T - S_i$ is the fraction of the variance on the model output that is "explained" by the interactions between the given variable *i and all* the remaining variables, in any number and combination.

The estimation of all Sobol sensitivity indices – if possible in spite of the required effort - provide an abundance of information which may be difficult to interpret. Conversely, they could be computed for a reduced number of variables suspected to have significant interactions with the other ones (having a high difference between the global sensitivity index and the main sensitivity index), as a second level of investigation.

2.10.4 Limitation

A general limitation of the variance-based approach for sensitivity analysis concerns the use of normal distribution to represent the uncertainty on the input variables, which could be not the optimal one. Indeed, it is generally recognized that log-normal distribution better represents the (epistemic) uncertainty, which should be associated to the basic parameters of the model.

Without looking at computational cost, a brute force approach could be applied in order to compute all indices specified in the variance decomposition. Frequently for PSA applications it is computationally too onerous.

As already remarked, the knowledge of the main and global sensitivity indices provides information on the local and global significance of single input variables and on the whole effects of its interactions with the remaining ones. As previously briefly introduced, different numerical methods have been proposed in order to improve the efficiency in the estimation of the sensitivity indices. Anyway, the computational effort required for the application of this approach to large models remains a main concern, suggesting its use when the non-linearity in the model are significant/dominant and secondary measures able to account for them are required.

2.10.5 ASAMPSA_E recommendation on Variance Based approach for Sensitivity Measures

There are no specific recommendations on a harmonized definition.

2.11 Qualitative Risk Measures

Two types of results are obtained in the PSA evaluation: qualitative and quantitative results. Qualitative results include:

- Minimal cut sets (combinations of components failures causing system failure).
- Qualitative importance (qualitative rankings of contributions to system failure).
- Common cause potentials (Minimal cut sets potentially susceptible to a single failure cause).

The minimal cut sets identify possible combinations of initiators and components or system failures that can result into an undesired state that can be core damage, release of radioactivity or some other predefined consequence analysed in the PSA.

The qualitative importance of the cut sets is identified by ordering the minimal cut sets according to their size (number of basic events in the set). Because the failure probabilities associated with the minimal cut sets often decrease by orders of the magnitude as the size of the cut set increases, the ranking according to size gives a gross indication of the importance of the minimal cut set. The identified minimal cut sets are screened in order to identify the minimal cut sets that are potentially susceptible to common cause failures resulting to larger risk of the analysed plant.

The qualitative importance measures are derived from the qualitative, logic structure of the PSA that includes the fault tree and event tree models [80]. The qualitative importance measures include Barlow–Proschan importance [81], structure importance measures ([82], [83]) and minimal cut set importance ([84], [85]). Logic expression of the top event is required for assessment of these importance measures ([86], [87]), limiting the applicability of these measures on real PSA models.

For the qualitative evaluations, the minimal cut sets are obtained by Boolean reduction of the analysed fault and event trees and application of the predefined truncation limits. Application of adequate truncation limits are necessary in order to obtain representative minimal cut sets considering the foreseen small probabilities of the initiating events in the extended PSA.

2.12 Fuel Damage Frequency (FDF)

With the FDF risk measure, we return to the discussion of direct risk measures. It should be noted that the secondary risk measures as presented in sections 2.3 to 2.10 can be defined in relation to any direct risk measure in principle. We therefore do not discuss their definition specific to the following direct risk measures. Moreover, each direct risk measure presented below can be sensibly defined both time dependent and as an average over time. For the respective discussion, see sections 2.1 and 2.2.

2.12.1 Definition of Risk Measure

Risk metric:

There are several definitions of the fuel damage state measure. Conceptually, the fuel damage state metric is either an extension of the core damage state metric or denotes a subset of core damage states at specific locations or operating conditions. According to the most comprehensive definition of a fuel damage state, this is understood as a loss of integrity of fuel elements on the site, which has the potential for a severe accident, i.e. an accident-level release (cf. e.g. section 3.1).

Other definitions include:

- Heatup of the fuel or severe physical impact on the fuel, which lead to anticipated significant releases from the fuel located in the reactor vessel or in the spent fuel pool, if the initiating event happens during non-full-power-operation [74],
- Loss of structural integrity of fuel elements in the spent fuel pool, understood as a subset of the core damage state. [75].

Apart from the Swiss regulator ENSI [74], no other regulators have specifically defined a fuel damage state. Usually, end states designated as fuel damage states are included as “core damage” state into the CDF for a (low-power and shutdown, LPSD) PSA or a PSA for the spent-fuel pool (SFP, see section 2.16).

Risk measure:

Irrespective of the specific definition of the fuel damage metric, the quantification of the FDF is always done with the direct frequency (or probability) of the sequence in the risk model, i.e. it assigns $\varphi(l_{ij})$ to the sequence $s_i \xrightarrow{\varphi(l_{ij})} c_j$. For more discussion, see section 2.1.

It should be noted that there are two versions of Fuel Damage Frequency, i.e. FDF, time average, and FDF, time dependent. The relationship between these two versions is the same as for CDF, time average, and CDF, time dependent. The respective discussion in sections 2.1 and 2.2 apply. For simplicity, both versions will be treated in this section.

Use of uncertainty distributions:

The FDF measure l_{FDF} is initially computed with point values for likelihoods. Uncertainty analysis as for CDF then produces the respective distribution $\varphi(l_{FDF})$. The discussion in section 2.1 applies.

2.12.2 *Areas of application:*

The FDF risk measure as a generalization of the CDF risk measure can (and should) be applied in the same areas as CDF, i.e. PSA for licensing, submissions to the regulator, oversight, design alternatives, risk gap analysis, etc. (see sections 2.1 and 2.2). This will include:

- Risk management
- Risk-informed decision making
- Risk monitors (FDF, time dependent)
- Risk budgeting (FDF, time dependent)

2.12.3 *Discussion*

Validity:

Fuel damage frequency corresponds - similar to CDF – to a well-defined state of the risk model, which can be assigned to adequately developed states of accident sequences. Like CDF, it is a leading indicator for challenges to the fundamental safety objective and aggregates of states at the interface between PSA Level 1 and Level 2. FDF is a valid risk measure for most purposes, depending on which a time-average or a time-dependent version should be applied. The validity of the FDF measure can be improved by a clear definition of the fuel damage state and by a consistent definition of the relationship between FDF and CDF.

Reliability:

Similar to CDF, there can be no unique definition of fuel damage for all kinds of reactor designs. Fuel damage states for a conventional LWR reactor design, a high temperature pebble bed reactor, and a lead-cooled GEN IV reactor will differ significantly. Conceptually, the definition of FDF needs to be consistent with the CDF definition, because both risk metrics are closely related. But once FDF (and CDF) have been clearly established, they allow for – in principle – reproducible PSA modelling.

Consistency:

FDF like CDF induces an order relation satisfying rationality criteria, if risk aggregation properties are properly considered. Basically, the measure is consistent. Furthermore, the consistency between FDF and CDF should be ensured, especially for risk aggregation.

Risk aggregation properties:

Aggregating FDF values over multiple scenarios (i.e. frequency values) is a well-defined operation, if performed on a minimal cut set basis as described above, resulting in a consistent risk measure. For a proper risk aggregation, there needs to be a clear definition of the relationship between CDF and FDF. And as with CDF, PSA specialists and decision makers have to acknowledge the limitations of the FDF measure. As there is no distinction between fuel damage states that likely lead to large releases and those that likely lead only to limited releases, simply aggregating the risk from this scenarios can (and often will) obfuscate the risk profile of the plant with regard to the fundamental safety objective.

Understandability to the PSA community:

FDF is not widely used in the PSA community. However, due to its direct link to CDF, it is well understandable to PSA practitioners and regulators.

2.12.4 Limitation

There are several limitations to the FDF (time average) risk measure. Generally, the FDF metric does not distinguish between severity of core damage (extent of damage to fuel rods) beyond the defining threshold for fuel damage. The respective discussion in section 2.1 applies.

Another limitation, which has already been mentioned above, is that the FDF metric does not preserve (or provide) information on fuel damage characteristics in light of expected releases (e.g. time of fuel damage onset, extent of fuel damage, status of barriers and safety systems, etc.).

Conceptually, the fuel damage metric stands at the transition from PSA Level 1 to PSA Level 2. Because of the limitations of the FDF measure, the interface between Level 1 and Level 2 is usually based on more detailed characterizations of the plant damage state reached. For more detail, see section 2.13. Thus, the FDF measures aggregates risk over the distinct plant damage states.

2.12.5 ASAMPSA_E recommendation on FDF

For the underlying issues on core damage frequency, see section 2.1. In addition, there is a need for a consistent definition of the FDF measure and its relation to the CDF measure.

FDF is defined as a loss of integrity of fuel elements on the site, which has the potential for a severe accident, i.e. an accident-level release.

Semantically the FDF measure provides a more general notion of a PSA Level 1 end state than CDF. Therefore, the ASAMPSA_E project recommends treating core damage states as subsets of fuel damage states: $\{CDF\} \subset \{FDF\}$. As explained in section 2.1, CDF should be understood as a fuel damage state affecting fuel elements located in the reactor core (e.g. the RPV). Consequently, the fuel damage state should be understood as a loss of integrity of fuel elements on the site, which has the potential for a severe accident, irrespective of operating state of the reactor or location of the fuel.

Moreover, the FDF measure needs to be consistent with the plant damage state measure(s) (PDSF) it shall aggregate. With the definition of FDF, all plant damage states should also qualify as fuel damage states (see section 2.13).

For Gen II and Gen III PWR and BWR reactor types, at least one of the following criteria applies to fuel located on the site:

- cladding temperature exceeds the threshold for onset of exothermic Zr-H_2O reaction in a subsection of the core with the potential for a large release (cf. section 3.1).
- rupture of fuel rod claddings releasing fission gases from the rods which, upon, release would amount to a large release (cf. section 3.1).

For CANDU-type reactors, a similar approach should be used that specifically links the fuel integrity to the FDF metric. The fuel damage metric should be defined as follows:

- Maximum fuel sheath temperature exceeds 600 °C, and the duration of post-dryout operation is more than 60 seconds (Potential fuel deformation and fuel element contact with the pressure tube causing its failure)[11]

With regard to FDF, time averaged, the ASAMPSA_E project recommends raising awareness about the limitations of the respective calculation methods. To the extent practicable, the time-averaged value should be computed based on the time-dependent version.

$$\varphi(l_{FDF,T_{\text{av}}}) = \varphi\left(\frac{1}{T_{\text{av}}}\int_{0}^{T_{\text{av}}} l_{FDF}(t)dt\right)$$

2.13 Plant Damage State Frequency (PDSF)

2.13.1 Definition of Risk Measure

Risk metric

A PDS is a group of accident sequences that have similar characteristics with respect to the accident progression and containment performance. Accident sequences allocated to a PDS must have similar characteristics not only in the degree of fuel damage, but also in other characteristics, which influence the release of fission products to the environment.

According to SSG-3 [4] and SSG-4 [5], plant damage states are a grouping “sequences leading to core damage […] based on similarities in the plant conditions that determine the further accident progression” [5], p. 4. Thus, plant damage states constitute the effective interface between PSA Level 1 and Level 2 (cf. Figure 1).

Then, the plant damage state is defined by differentiating the core damage (section 2.1) or fuel damage (2.12) risk metric by a set of additional attributes. A specific plant damage metric (PDS_j) is then defined by a (consistent) combination of attributes. An example of criteria for differentiating these states for the binning of Level 1 sequences is given in Table 2. It is important to note that the adequate definition of plant damage states depends (at least) on the reactor type as well as the objectives and scope of the PSA Level 1 as well as the PSA Level 2.

[11] Performance requirements for the reactor shutdown system(s) for all design basis accidents other than large LOCA and single channel design basis events, such that the fuel integrity and the primary heat transport system integrity is not jeopardized [92]. For large LOCAs and single channel design basis events, the initiating event is a fuel failure per definition.

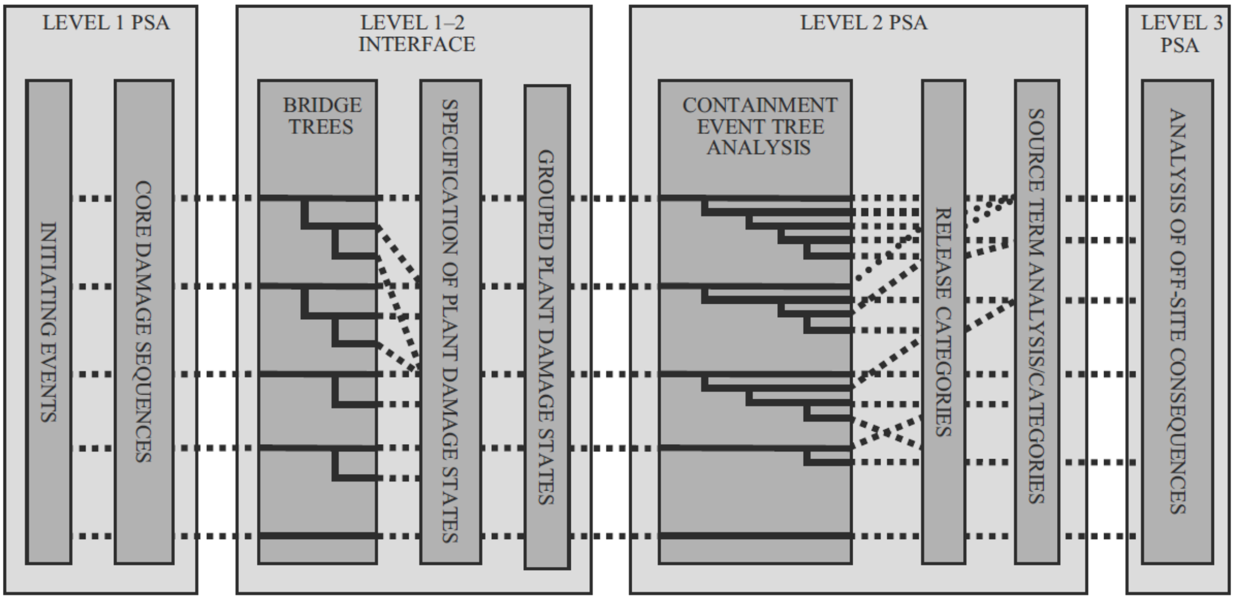

Figure 1 : Connection between PSA Levels [5]

The following tables provide examples of PDS applied in different countries.

Table 2 : PDS Attributes for a German Type PWR Reactor (following [107])

PDS Attribute Class	Recommended Attributes
Initiating event	Transient vs. LOCA Large break LOCA vs. Small break LOCA Stuck-open safety/relief valve Anticipated Transient Without Scram (ATWS) Bypass event (interfacing systems LOCA, or steam generator tube rupture) Status of power supply (SBO, LOOP)
Primary side depressurization	Successful, available but not actuated, unavailable
Injection to the RPV	HP or LP injection systems available or unavailable
RCS pressure at core damage	Below LP systems, below HP system, larger than HP systems
Coolant mass from RCS to containment Water from refueling water storage tanks Water from accumulators	Adequate groups of water masses, differentiated by water source (storage tanks, accumulators, RCS inventory)
Secondary side heat removal	Available/not available
Containment isolation	Isolated / not isolated
Time to core damage	e.g. early / medium / long
Containment leakage control	Available / not available
Air recirculation systems for service and	In operation / not in operation

for equipment compartments	
Operational annulus ventilation system	In operation / not in operation
Annulus air extraction system	Filtered release / isolated

Sometimes, though, PDS categories are extended to include also controlled sequences with limited damage to the reactor fuel. The following table provides an example of PDS (applied in Canada):

Table 3 Exemplary Plant Damage State Definitions (with sequences with limited damages to the reactor fuel)

State	Description
PDS0	Early (rapid) loss of core structural integrity
PDS1	Late Loss of Core Structural Integrity with High PHT Pressure
PDS2	Late loss of core structural integrity with low PHT pressure
PDS3	Loss of core cooling with moderator acting early (<15 min) as last-resort heat sink
PDS4	Loss of core cooling with moderator acting eventually (>15 min) as last-resort heat sink
PDS5	Large LOCA with successful initiation of ECC but partial loss of cooling
PDS6	Single-channel fuel damage with discharge into containment
PDS7	Single-channel fuel damage with discharge into calandria vessel
PDS8	Loss of cooling to fuelling machine
PDS9	LOCA with no significant fuel failures
PDS10	Deuterium Deflagration in Calandria Vessel and/or release of Moderator Inventory into Containment, fuel cooling maintained

Appendix C (Section 10), presents an example of the application of PDS in Canada. Fuel damage category (FDC) frequency is used to represent a collection of event sequences judged to result in a similar degree of potential fuel damage. The FDCs are used as end-states in the Level 1 event trees. Groupings of the fuel damage categories are used to transition from the Level 1 PSA to the Level 2 PSA (Reference [90] and [91]).

The following table provides a simpler example (applied in France).

Table 4: Example of Plant Damage State Definitions (France)

PDS1	Core damage with no containment failure until core degradation.
PDS2	Core damage with early containment failure (containment bypass, containment isolation system failure, …)
PDS3	Core damage with late containment failure (failure of CHRS, …)

Risk measure:

The quantification of a specific PDS metric (PDS_j) is always done with the direct frequency (or probability) of the sequence in the risk model, i.e. it assigns $\varphi(l_{ij})$ to the sequence $s_i \xrightarrow{\varphi(l_{ij})} c_j$, where the consequence c_j contributes to PDS_j, i.e. $c_j \in \mathrm{PDS}_j$. For more discussion, see in section 2.1.

Use of uncertainty distributions:

The discussion in section 2.1 applies.

2.13.2 Areas of application:

PSA level 1 +

Before developing a PSA level 2, it may be useful to extend level 1 PSA to such PDS calculations. This allows providing information on accident scenario that may lead to both fuel damage and a short or long-term containment failure. Such PSA level 1+ which does not include information on severe accident progression can be very useful, for example during a NPP design phase. It may help to reduce the probability of accidents that have a potential to lead to large radioactive release. Such PSA level 1+ can also be considered for risk monitor if based only on L1 PSA. Section 2.17 discusses more in details an example of PSA level 1+ risk metric.

Interface PSA Level 1 to PSA Level 2

Plant damage state risk metrics are traditionally used to construct the interface between PSA Level 1 and PSA Level 2.

If the intent is to use the results of the Level 1 PSA as input to a Level 2 PSA, it is general practice to group the accident sequences that lead to core damage into plant damage states, which will form the interface between the Level 1 PSA and the Level 2 PSA. It is more useful if the plant damage states are specified as a part of the Level 1 PSA (rather than postponing the specification of plant damage states to the first step of the Level 2 PSA) [4].

As an example, from Table 3 presented above, the categories PDS0, PDS1 and PDS2 are considered severe core damage (meltdown) states. All of the fuel in the core is assumed to be compromised in these scenarios. Moreover, level 1 safety goals, expressed in terms of CDF, are assessed based on the sum of PDS0, PDS1 and PDS2. All accident sequences that do not lead to core damage can be described by one of the PDS from 3 to 10. Other PDS are added to include the fuel behavior in the fuel bay, dry storage tanks, etc.

2.13.3 Discussion

Validity:

The PDS approach is commonly used for the interface between L1 and L2 PSA. The definition of PDS leads to add in the L1 PSA modelling dedicated to the containment function. It gives a possibility to enlarge the information got from the level 1 PSA but it increases the complexity of the L1 PSA model. It should be noted that L1 PSA tools often neglect success probabilities in their quantification of sequence (and even consequence) results. In this case, the sum over the (nominally disjoint) PDS results can be larger than the respective CDF or FDF result. However, in most cases the numerical error is significantly smaller than the range of uncertainty (e.g. at one σ level) assigned to the CDF/FDF results and thus tolerable.

Reliability:

The PDS provides some views on the availability of the containment function in case of fuel damage. The scenario included in a PDS with a "containment failure attribute "can be associated to the "large release" accidents. But the scenario included in a PDS with no "containment failure attribute" cannot be associated to accident with "limited consequences": a plant response analysis during severe accident progression is needed to check that the containment can resist to the severe accident conditions. This is the role of L2 PSA.

Consistency:

One difficulty is that, for each NPP design, there is not a single solution to define a set of PDS but multiple possibilities. The ASAMPSA2 project [2] has shown for example that reaching a harmonized definition of PDS would be very difficult.

Risk aggregation properties:

PDS frequencies should not be used for risk aggregation: it can be used to provide a minimal value of LERF or LRF.

Understandability to the PSA community:

The PDS approach is commonly applied and understood by the PSA community. As explained above, the practical implementation of PDS differs between organizations.

2.13.4 Limitations

The most important limitations noticed above are the following

- Different PDSs definitions exists depending on NPP design and L1 – L2 PSA development options (no possible harmonization),
- PDSF can provide only some indication for LERF or LRF and cannot replace a L2 PSA.

2.13.5 ASAMPSA_E recommendation on PDSF

Plant Damage States Frequencies (PDSF) are mainly dedicated to the interface between L1 and L2 PSA. Nevertheless, even if there are limitations, it constitutes a useful complement to the CDF calculated by L1 PSA. It allows estimating a minimal value of LERF and LRF without developing fully the L2 PSA.

It is recommended to implement such metrics in L1 PSA and to use it in applications: this allows introducing some consideration on the containment function in the L1 PSA results.

Some important PDS attributes for PWRs and BWRs are for example:

- Time to core damage,
- RCS pressure at core damage,
- Isolation of the containment,
- Containment bypass scenario.

2.14 Interface Core Damage Frequency (ICDF)

There is not any awareness that this proposed risk measure has been applied somewhere. The risk measure would be defined as the aggregation over all sequences, which contribute to CDF, and which in addition are included in the interface between PSA Level 1 and PSA Level 2. We point out that this is merely a variant of the CDF risk measure and can be derived from the PDS risk measure discussed in section 2.13. Moreover, it is expected that all sequences contributing to CDF (or FDF for that matter) are transferred to the PSA Level 2. In section 2.13 we have already discussed why certain (limited) numerical differences between the sum of PDS risk measures and the CDF/FDF value can arise. Overall, it is not recommended to use this risk measure as a separate direct risk measure.

2.15 Hazard State Frequency (HSF)

It should be noted that this specifically German risk measure has no connection to a natural hazard or internal hazard event, but rather with the meaning of hazard as “being in peril” or “endangerment” (German: “Gefährdung”).

As with the CDF measure, there are in principle two versions of the HSF measure, time averaged and time dependent. Both are treated in this section.

2.15.1 Definition of Risk Measure

Risk metric

The hazard state metrics according to [77] is a condition of the plant, where core cooling is no longer provided by systems (automatically or manually from EOP actions), which have been designed for this safety function. If operators take no further preventive accident managements actions or additional measures that are effective, this state would progress to a core damage state. It should be noted that the hazard state definition in German PSA practice often includes measures formally assigned to preventing accident management, provided they are actuated independently by I&C classified on a level with the RPS.

Practically, the hazard state metric is arrived at by neglecting human preventive accident management measures at the end of the common accident sequence analysis and event tree derivation.

Risk measure

The quantification of the hazard state metric (HSF) is always done with the direct frequency (or probability) of the sequence in the risk model, i.e. it assigns $\varphi(l_{ij})$ to the sequence $s_i \xrightarrow{\varphi(l_{ij})} c_j$, where the consequence c_j contributes to the hazard state.

Use of uncertainty distributions:

There are no differences to the CDF risk measure already explained in sections 2.1 and 2.2.

2.15.2 Areas of application

PSA for licensing, submissions to the regulator, oversight, design alternatives, risk gap analysis,
risk-informed decision making and theoretically: Risk management, EOPs development, risk monitors (time dependent).

2.15.3 Discussion

Validity:

The hazard state metric can be assigned to specific states of a PSA. There is, however, substantial leeway in the definition in terms of what specifically defines a hazard state.

Moreover, the hazard state metric is only a weak leading indicator for the risk of accidental releases, because it aggregates over scenarios with accidental releases and those without any significant releases. In that respect, it can provide less valid information to decision makers than e.g. CDF or FDF. Worse, it may even support a distorted understanding of the risk profile of the plant as captured in FDF and release category measures. Conversely, HSF is not valid as a risk measure for the risk of exceeding DiD Level 3 or the risk of leaving the design basis envelope of the plant, because HSF extends partially into the design extension region.

The HSF measure in connection with the CDF or FDF measure can provide insights in the effectiveness of accident management measures in a general sense. However, these can also – and more specifically – be evaluated by using for example conditional core damage probability or conditional system unreliability measures. The risk aggregation issues for CCDP and similar secondary risk measures are not captured with HSF. Consequently, the validity of HSF for this purpose can be limited.

Reliability:

Similar to CDF, there is no unique, technical definition of the hazard state. Design basis and preventive accident management are necessarily specific to each reactor type and sometimes even plant-specific. This, together with the ambiguities in the definition of the risk metric, results in significant differences in the scenarios included into the hazard state metric. This can lead to substantial differences in PSA results for HSF, even for rather similar plants.

However, if HSF has been clearly defined for a specific plant, it allows for PSA modelling which is in principle reproducible. Differences can then be explained by discretionary choices of PSA analysts.

Consistency:

HSF like CDF induces an order relation satisfying rationality criteria, if risk aggregation properties are properly considered. Basically, the measure is consistent. Furthermore, the consistency between HSF, FDF and CDF should be ensured, especially for risk aggregation. It should be noted that both fuel damage and core damage states are subsets of hazard states.

Risk aggregation properties:

Aggregating HSF values over multiple scenarios (i.e. frequency values) is a well-defined operation, if performed on a minimal cut set basis, resulting in a consistent risk measure. However, since the HSF measures extend to scenarios with widely differing consequences with regard to the fundamental safety objective (scenarios leading to large releases as well as scenarios without any accidental releases), and since HSF provides no distinction between these scenarios, aggregating HSF over different sequences can obfuscate the actual risk contributions and bias decision making processes.

Understandability to the PSA community:

The HSF measure is a commonly used measure within the German PSA community. Understandability of the HSF measure is significantly hindered by the usual connotation of “hazard” in the English language in the field of PSA as natural hazard or internal hazard event. Thus, HSF might be misleading.

2.15.4 Limitation

There are several limitations to the HSF risk measure, which have already been mentioned above. The HSF metric aggregates scenarios with accidental releases and those without any significant releases. Like FDF, it provides no further information regarding the severity of potential releases, the status of the reactor, containment and ventilation systems, etc. Moreover, it is neither a valid risk measure for DiD Level 3, DiD Level 4, design basis accident risk or design extension conditions. Actually, the HSF risk measure is located somewhere between DiD Level 3 risk and the CDF/FDF measure. Aspects of risk captured by the HSF metric can often be captured with CCDP. The HSF measure is not sensitive to risk aggregation issues related to these secondary risk measures.

2.15.5 ASAMPSA_E recommendation on HSF

Due to the issues with the validity of this risk measure and the problems regarding understandability of this HSF, the ASAMPSA_E project does not recommend the use of the HSF measure for extended PSAs.

For assessing the effectiveness of specific emergency operating procedures or preventive accident management actions, the risk measure is well suited. It should be recognized that these risk measures have to be evaluated separately for each scenario. Risk aggregation on e.g. CCDP is only meaningful if Bayes’ law is adhered too.

2.16 Spent Fuel Pool Damage Frequency (SFPDF)

2.16.1 Definition of Risk Measure

Risk metric

The risk metric relates to the challenges to adequately cool the used fuel located in a spent fuel pool (SFP) for events like loss of cooling, loss of inventory and reactivity accidents and the consequent safety system success criteria to cope with the concerned risks, like the systems devoted to decay heat removal and water make-up. This risk metric is a subsidiary of the FDF risk metric discussed in Section 2.12 as it is specific to a location (Spent fuel pool) compared to the more general metric FDF.

Seismic induced structural failures, heavy load drops (e.g. during dry cask movements) as well as reactor induced challenges, like reactor severe accident conditions resulting in adverse SFP conditions or adverse SFP cooling/make-up equipment conditions and related phenomena causing structural failure, like hydrogen explosion, are to be included likewise.

So far the analysis of accident sequences leading to SFP fuel damage based on event tree/fault tree approach (ET/FT) and the probabilistic accident progression analysis based on accident progression event trees (APETs) indicates the FDF and the LRF as the most suitable risk metrics for SFP.

Frequency of Spent Fuel Uncovery could be conceived as a level 1 risk surrogate metric, with reference to accident sequences leading to spent fuel uncovery (and overheating).

<u>Risk measure</u>

Refer to 2.12.1

<u>Use of uncertainty distributions</u>

Refer to 2.12.1

2.16.2 Areas of application:

All the areas concerned with PSA approach adoption and benefits are of interest, that is:

PSA for licensing, submissions to the regulator, oversight, design alternatives, risk gap analysis, etc., see also 2.12.2.

2.16.3 Discussion

With respect to the validity, reliability, consistency and risk aggregation properties of this risk measure, we point out that the SFDF should be defined consistently with the FDF risk measure. Then, the SFDF metric is simply the subset of fuel damage states occurring specifically in the SFP. Therefore, the discussion provided in section 2.12 applies.

We furthermore point out the following. The location of the SFP (for example inside or outside the containment in the reactor building or in a separate storage facility) affects the risk assessment of the plant.

While the risk metric applies only to the SFP, the resulting PSA model needs to consider the interaction with the reactor which cannot be neglected. For instance, the RHR is used to cool both reactor and SFP in common reactor designs, and some initiating events, like loss of offsite power affect reactor and SFP simultaneously and reactor and SFP are interconnected in some operating states like during refueling. Thus the reactor and SFP combined PRA model is needed.

The interaction of severe accident progression in the containment and subsequent adverse impact on the SFP has to be considered as well (hydrogen explosion, availability of safety systems and containment condition): this is particular relevant as far as the SFP is located inside the containment.
Finally, the case of simultaneous severe accidents in the reactor and SFP could contribute significantly to the risk profile.

2.16.4 Limitation

Refer to 2.12.4
Moreover, the SFPDF risk measure applies only to the spent fuel located in a SFP. It should not be extended to include spent fuel in dry storage, e.g. in casks stored at an interim storage facility on the site.

2.16.5 ASAMPSA_E recommendation on SFPDF

We recommend that the SFPDF risk measure is defined as a subset of the FDF risk measure, applicable to spent fuel located in a spent fuel pool on the site. Conversely, we recommend that the CDF risk measure is defined as the subset of the FDF risk measure applicable to fuel located in the reactor core. This implies that $\{CDF\} \cup \{SFPDF\} \subseteq \{FDF\}$.
For the quantification of the integrated PSA model considering both the reactor core and the SFP, the types of results of interest include the following:

- Spent Fuel Damage Frequency (SFDF) in the spent fuel.
- Core Damage Frequency (CDF) in the reactor.
- Damage states both in the SFP and in the core.

We emphasize that such an integrated PSA model needs to systematically consider interactions that involve simultaneous or consequential accident progression in the reactor and the SFP.

2.17 *Radionuclide Mobilization Frequency (RMF)*

During the ASAMPSA_E meetings, there was a discussion on PSA Level 1 risk metrics. It was commented that the main risk measures for PSA Level 1 like e.g. core damage frequency or fuel damage frequency are not well suited for describing several scenarios which might lead to a significant release of radionuclides into the plant as a starting point for a PSA Level 2. The following “radionuclide mobilization” metric addresses these issues. As with CDF, this risk measure can be defined at a specific point in time or as time-averaged. The respective remarks in section 2.1 and 2.2 apply.

2.17.1 Definition of Risk Measure

Risk metric

The risk metric is defined as a loss of the design basis confinement for a source of radionuclides, leading to an unintended mobilization of a significant amount of radionuclides with the potential for internal or external release, e.g. more than 1 TBq I-131 or equivalent[12]. The threshold value and its reference radionuclide (or radionuclides) has to be adjusted to the facility under consideration and the objectives of the study. In setting such a threshold, typical radionuclide inventories of NPP should be taken into account. For a 2.4 GW_{th} BWR core, the radionuclide inventory of I-131 is upwards of 1 EBq (=1,000,000 TBq) and for Cs-137 upwards of 100 PBq (=100,000 TBq), see e.g. [109]. The proposed threshold is therefore already reached if the inventory of one fuel rod is mobilized to a significant degree. For the mobilization of radionuclides it shall be assumed that all radionuclides affected by the loss of the barrier/confinement are mobilized unless they are clearly immobile[13]. Since this risk metric can also be used to examine short-term consequences e.g. to on-site personnel, it should be defined with I-131 as leading isotope. The loss of design basis confinement should be understood in terms of a fault or malfunction that allows radionuclides in significant amounts to get mobilized and be released from their designed location. This applies to significant damage to fuel rod cladding due to excessive cladding temperature and to cladding failures due to mechanical impact (cf. fuel damage frequency) but also to other potentially relevant scenarios like leakages from radioactive waste processing or storage systems, damages to waste storage casks, and other significant sources of radioactivity on a site.

Risk measure

The quantification of the radionuclide mobilization frequency (RMF) is to be done by direct frequency (or probability) of the sequence in the risk model, i.e. it assigns $\varphi(l_{ij})$ to the sequence $s_i \xrightarrow{\varphi(l_{ij})} c_j$ where the consequence c_j contributes to a radionuclide mobilization state.

Use of uncertainty distributions:

There are no differences to the CDF risk measure already explained in sections 2.1 and 2.2.

2.17.2 Areas of application

The RMF is a proposal discussed during the ASAMPSA_E project. Currently, no applications are known. However, the RMF generalizes the CDF and FDF risk measures to a comprehensive PSA Level 1 risk measure for a multi-source PSA.

This risk measure can contribute to the verification of the low probability of events that would induce off-site protective measure without core melt. Such verification has been done for the EPR FA3 but with L2 PSA.

2.17.3 Discussion

Validity:

The RMF risk measure is clearly defined if a threshold value for a representative radionuclide has been set. Then, it can be associated with a well-defined state in the risk model. Moreover, radionuclides that

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[12] The proposed threshold value has been set to 1 % of the lower end 100 TBq I-131 limit for an accidental level release (INES 5) defined in the INES manual [108]. This assumes that short-term consequences are of interest. For long-term consequences, a threshold based on e.g. Cs-137 should be selected. .

[13] For example, radionuclides solved or dispersed in a water circuit with a break (beyond design leakage) should be assumed to be potentially mobilized, whereas the activation products within the piping steel should still be considered immobile.

are becoming potentially mobilized in an uncontrolled and unintended manner are a good leading indicator for the risk of accidental release. There needs to be a clear understanding, though, what is understood under a mobilization of radionuclide and which radionuclides are considered immobile. In order to increase the validity of the risk measure, radionuclides should be considered potentially mobilized unless they are immobile. The latter can be understood as that physical or chemical processes relevant to the respective scenario over the relevant analysis time (i.e. in the order of days or at most weeks for an extended PSA for NPP) will not lead to the transport of the respective radionuclides in relevant amounts from the current location and outside of the boundary of the designed confinement. Similar to CDF, these conditions might change if a sequence is further developed. The RMF metric allows for generalizing the CDF to other relevant radionuclide sources in a NPP in a consistent manner. Conversely, the RMF fundamentally aggregates quite diverse scenarios contributing to risk, from comparatively benign scenarios without a significant risk of on-site and off-site consequences to scenarios with a high probability for severe off-site consequences. This is a significant limitation of this risk measure. Assigning the likelihood (distribution) for the respective sequence(s) ending in a radionuclide mobilization state is a clear and traceable quantification procedure, as is risk averaging over time.

The main difference between the proposed RMF risk measure and the PSA Level 2 risk measures for accidental release is the following. PSA Level 2 risk measures like e.g. LRF are defined on the release of radionuclides to the environment of the plant (off-site release), i.e. at the relevant plant or site perimeter. For such a release to occur, several barriers for the confinement of radionuclides at a NPP (or other high-risk source) have to fail according to the Defense in Depth approach. Consequently, Level 2 risk measures address the risk of multiple barrier failure leading to a release. Conversely, the definition of the RMF risk metric addresses the failure of the first barrier designed to confine a relevant radionuclide source (like e.g. severe cladding failure for the FDF). Then, radionuclides get mobilized and are transported within the plant to locations not specified for the operation of the plant or facility. This leads to a challenge of the next barriers for the confinement (like e.g. the containment in a NPP). The lower radionuclide threshold proposed for the RMF metric ensures that it covers all significant accidental off-site releases as well as less severe releases.

Overall, the RMF is a valid risk measure for a generalized, multi-source PSA Level 1.

Reliability:

The RMF can be clearly defined if recourse to a potential release quantity is made. In this way, it can be consistently applied to a large type of reactor designs and types of radionuclide sources. If the RMF measure has been established, it allows for a reproducible PSA modelling. It is therefore a suitable risk measure for a generalizing multi-source PSA.

Consistency:

The RMF induces an order relation satisfying rationality criteria, if risk aggregation properties are respected. This risk measure is basically consistent.

Risk aggregation properties:

Aggregating RMF values over multiple scenarios (i.e. frequency values) is a well-defined operation, if performed on a minimal cut set basis, resulting in a consistent risk measure. However, it is essential to bear in mind that scenarios assigned to the RMF metric represent widely different scenarios in terms of actual and potential consequences. The risk associated with a leakage in a liquid radwaste treatment system in the auxiliary systems building can be highly relevant to operating staff, but will at worst lead to limited off-site consequences. Conversely, a high pressure core melt during a prolonged SBO scenario might lead to unacceptable off-site contamination. So, while the RMF is suitable for aggregating those widely different risk aspects, it is at the same time not well suited for understanding the full risk profile of the plant with regard to the fundament safety objective.

If a stronger discrimination between scenarios with very severe consequences and more limited consequences based on the RMF measure is intended, then we recommend to define at least two variants of the RMF. In addition to the low threshold metric defined above a “severe radionuclide mobilization” metric (SRMF) could be defined with a radionuclide threshold of e.g. 1 PBq I-131 or even higher.

<u>Understandability to the PSA community:</u>

The RMF measure is currently only a proposed risk measure. It should be understandable to the PSA community, though. Ambiguities can arise from different threshold values or selecting a leading radioisotope other than I-131. Similarly, the issue of mobilized vs. immobile radionuclides can give raise to ambiguities. However, these types of ambiguities can be clearly described and understood. Moreover, due to the comparatively small threshold value proposed for this risk measures (e.g. 1 TBq I-131 (equivalent)), differences in these assumptions should have rather limited consequences for the results and also for the respective conclusions.

2.17.4 Limitation

As already mentioned above, the RMF conceptually aggregates rather diverse sequences in terms of consequences into one common risk measures (figure of merit). While this is one of its advantages, it similarly limits its suitability for understanding the actual risk profile with regard to the fundamental safety objective. With this caveat, the RMF can cover for most conceivable scenarios leading to accidental releases. The most notable exception of cases not covered by the RMF risk measure is direct irradiation from the immobile source. These scenarios, however, are basically irrelevant for off-site consequences.

2.17.5 ASAMPSA_E recommendation on RMF

The source term threshold for defining the RMF metric (e.g. 1 TBq I-131 (equivalent)) needs to be consistent with release metrics selected for the PSA Level 2. Specifically, the source term threshold should not be larger than the threshold for the early release metric (see section 3.2). Additionally, the PSA Level 2 will usually define specific release categories for filtered releases and other scenarios without failures of the containment function. Such releases might be in the range of 10^{-6} of the total core inventory of volatiles, which is consistent with 1 TBq I-131 (equiv.).

The RMF definition given above was developed during the ASAMPSA_E project. The RMF risk measure is recommended to be used for an extension and generalization of the established CDF and FDF risk measures to a multi-source PSA (cf. section 4). It is therefore a suitable and above all complementary risk measure for an extended PSA that addresses potential sources on the site in addition to fuel in the reactor and spent fuel.

It must be pointed out, though, that the RMF risk measure is not well suited for understanding the risk profile of e.g. an NPP in operation. It should be complemented by e.g. CFD/FDF as a PSA Level 1 risk measure.

3 RISK METRICS FOR LEVEL 2 PSA

Most direct measures metrics defined for PSA Level 2 are related to the off-site release of radionuclides. Release measures are constitutive for the definition of PSA Level 2 [5], because the dividing line between Level 2 and Level 3 is put at accidental releases transgressing the plant boundary. Thus, they are explicitly intended to address potential off-site consequences in the environment of the plant. They are therefore typically strong leading indicators for the risk of not meeting the fundamental safety objective (with respect to off-site consequences). The major differences in the release risk measures discussed below lies in the classification with respect to the amount of radionuclides released, the leading (representative) isotope for that class, and in the consideration of (a set of) other attributes (like the timing of the release).

As with CDF, Level 2 release measures can be defined in both a time-averaged and time-depended version (see Section 3.1 and Appendix A for more details). The respective comments in sections 2.1 and 2.2 apply. Moreover, the secondary risk measures presented in sections 2.3 to 2.10 above can be also applied to Level 2 risk measures. Therefore, no additional discussion is provided in this section.

The combined evaluation of accidents for the reactor core and for SFP is appropriate in order to take into account the complete risk in the sense of an extended PSA. This does not affect the following discussion of risk metrics in principle, but practical questions will arise when releases from the core and the SPF occur in different quantity and time scale. Pertinent comments to this issue are provided in section 5.

3.1 Large Release Frequency (LRF)

3.1.1 Definition of Risk Measure

Risk metric:

A large release is commonly understood to be an unacceptable release of radionuclides from the plant into the environment of the plant.

SSG-4 [5] (cf. also [111], [112]) defines a “large release [as] a release of radioactive material from the plant to the environment that would require off-site emergency arrangements to be implemented. The release can be specified in a number of ways including the following:

- as absolute quantities (in Becquerel) of the most significant radionuclides released;
- as a fraction of the inventory of the core;
- as a specified dose to the most exposed person off the site;
- as a release resulting in ‘unacceptable consequences’.”

NEA [69] provides the following general definition: large release frequency (LRF) is expressed in terms of the quantity of radioactive elements such as I-131 and Cs-137 released to the atmosphere.

There is a wide range of specific definitions for the large release metric, e.g.

- AREVA: More than 100 TBq of Cs-137 including dose weighted contribution of other elements,
- LEI: More than 5% of iodine and caesium,

- Dukovany NPP (UJV, Czech Republic): >1% of Cs-137 of the core inventory (responding approximately to 10 000 TBq) released to the environment,
- Temelin NPP (Czech Republic): fission product fraction released through large opening in the containment to the environment,
- Mochovce NPP (VUJE, Slovak Republic): > 3% of volatiles released to the environment,
- Bohunice NPP (Relko, Slovak Republic): > 1% of Cs-137 released from the core inventory to the environment,
- Paks NPP (VEIKI, Hungary): Large release >10000 TBq,
- French 900, 1300, 1450 MWe PWRs before Long Term Operation (LTO) upgrade: release amount exceeding those induced by a late containment filtered venting during a severe accident,
- In Ukraine, large release is defined as requiring public evacuation at the boundary of the protection area.

The specific threshold for a large release depends on two judgments: First, on what constitutes an unacceptable accidental release, and second on what would necessitate (relevant) off-site emergency measures more specifically the following statements can be proposed:

- the specific threshold for large release for one NPP shall be consistent with the general safety objectives defined for this NPP,
- for each NPP, the general safety objectives associated to severe accident management shall include an objective of limitation in space and time of off-site protective measures (this is the main objective of severe accident management strategies),
- the specific threshold for large release is in general lower for the more recent NPPs (typically Gen III NPP) or for the Gen II NPP which have been specifically upgraded for severe accident management.

Harmonization of a specific threshold of large release (numerical values) does not exist.

The risk measure is usually applied to (end-) states in the PSA Level 2 risk model (i.e. a consequence).

Iodine 131 is usually selected as a representative isotope for early consequences due to its 8-day half-life and serious health impact if digested.

Caesium 137 is usually selected as a representative of total long-term consequences due to 30-year half-life and serious environmental impact (soil contamination).

Both I-131 and Cs-137 (as CsI) are significant contributors to the group of volatiles (beyond noble gases) for enriched uranium as well as mixed oxide (U/Pu) based reactor fuels.

For severe accident scenarios, there will typically be a high initial release in the first hours, days, or even weeks of the accident, c.f. e.g. [109]. On a long time scale, there will typically still be releases, but these will usually be irrelevant for the total amount of releases. It is therefore justified to define a reference time T_{ref}, at which further releases from the site is ineffective. Assuming a representative source term is assigned to a sequence and that this is independent of the time of the initiating event, then this source term can be integrated over this reference time (see also Appendix A).

$$r_{A,i} = \int_{0}^{T_{\mathrm{ref}}} r_{A,i}(0,\tau)d\tau$$

The value of T_{ref} needs to be chosen in such a way that the significant part of the release has already happened (e.g. 99%).

3.1.2 Discussion

Validity: Large release is – once defined – a clearly described state in the risk model. Moreover, a large release to the environment is a good leading indicator for failing to meet a fundamental safety objective.

LRF is providing important information on the risk of the plant, aggregated over sequences with relevant off-site consequences. LRF is a particularly good leading indicator for potential long-term loss of land (soil contamination) and other area effects, if defined based on (volatile) radioisotopes with medium to long half-life times like e.g. Cs-137. Nevertheless, depending on the definition of large release, all scenarios that contribute to LRF will not necessarily lead to large land contamination. This is an important limitation of LRF risk measure. It cannot replace a more precise L2 PSA release categorization in function of the amplitude of release for the identification of the more dangerous accidents.

LRF addresses risk objectives stated in SSR-2/1 [112] for the practical elimination of large radioactive releases and WENRA's objective O3 in Ref. [111].

Assigning the likelihood (distribution) to the sequence(s) ending in a large release is a clear and traceable quantification procedure. Averaging risk model results over time is a sensible and consistent way of defining a risk measure. In this regard, LRF is a valid risk measure.

Reliability: There is no unique definition available for the large release metric but the current practice is to define a threshold (either on I-131 or Cs-137) that can be used to identify all scenarios that would need off-site protective measures (with more or less extension depending on the NPP). So, while there is common agreement to base the LRF metric either on I-131 or Cs-137, there is no agreement on the following:

- If the risk metric should be declared based on one isotope only or if contributions from other isotopes from the release vector should be weighted by their radiological importance in relation to the representative isotope.
- The specific quantitative value of the threshold for a large release.
- There is also not necessarily agreement on the time scale for the integration for the large release. While there is agreement that the release needs to be integrated over more than 24 hours, there is no agreement what an appropriate cut-off time would be.

Nevertheless, if the large release metric has been clearly established, it allows for – in principle – reproducible PSA modeling of the accident sequence analysis. In that sense, LRF is a reliable measure even though the large release metrics are not sufficient to identify the scenario that would induce the more serious consequences or to identify situation with short term release for which emergency measures (evacuation) will not be effective. This is an important limitation and LRF, as defined above, cannot be used exclusively for risk ranking.

Consistency: Large release frequency induces an order relation satisfying rationality criteria, if risk aggregation properties are properly considered. Basically, the measure is consistent. However, the issues related to reliability and risk aggregation properties should not be overlooked.

Risk aggregation properties: Aggregating LRF values over multiple scenarios (i.e. frequency values) is a well-defined operation, if performed on a minimal cut set basis (or disjunctive sequences), resulting in a consistent risk measure.

With respect to risk aggregation, PSA specialists and decision makers have to acknowledge the limitations of the LRF measure:

- LRF does not identify release scenarios that develop in a short time and for which off-site emergency measures (evacuation) will not be effective.
- LRF is particularly suited to assess likely effects to the environment of the plant but it does not to discriminate in function of the severity of the accident and cannot be used exclusively for risk ranking. If the LRF source term threshold is rather small (e.g. 100 TBq I-131 equiv.), then LRF aggregates the risk over accidents with comparatively limited consequences as well as manifestly severe releases as for the Fukushima Daiichi and Chernobyl accidents. This might obfuscate the risk profile of the plant to some extent. PSA analysts and decision makers need to be aware of this issue.

Understandability to the PSA community: LRF is a commonly used risk measure. It is well understood in the PSA community as one important release category at the end of a PSA Level 2. However, there are variations in the exact definitions of the LRF metric in function of NPPs and countries.

In principle, the LRF metric can be defined in a consistent manner for relevant types of reactors and other sources on a NPP site.

3.1.3 Limitation

As already mentioned above, the LRF metric does not identify those sequences, for which the off-site emergency measures (especially evacuation) are not effective. It is therefore not a suitable leading indicator for the risk of acute irradiation of the population in the vicinity of the plant.

Depending on the specific threshold set for the LRF, this risk measure might aggregate scenarios corresponding to INES Level 5 (an accident with limited consequences) and INES Level 7 (an accident with major off-site consequences). In these cases, the LRF can obfuscate the risk profile of the plant relevant to decision makers and stakeholders to a certain degree. It might be necessary to complement the LRF risk measure with a dedicated risk measure capturing such very severe scenarios (e.g. a more precise release categorization from L2 PSA).

3.1.4 ASAMPSA_E recommendation on large release measure

The main objective of NPPs severe accident strategies is to limit in time and space the off-site protective measures in case of severe accident. The LRF is a metric that can be used to obtain a measure of the probability of occurrence of severe accidents which would need off-site protective measures not limited in time and space. This is a main result of a L2 PSA and should be part of the NPP safety report.

The use of LRF metrics need to define one or several numerical measures that allows identifying accident corresponding to “large release”. These numerical values have to be defined by the utility or by the regulator.

Such specific threshold for a large release should be consistent with the protection of population and environment. This should be an objective discussed during the NPP design and plant upgrades after the start of operation (especially during PSR).

Moreover, the following best practices are recommended in applying a (more) harmonized definition for the LRF:

- the (representative) source term for determining the amount of release for the scenario should be integrated until no significant further contributions to the (total) release will happen (cf. also ASAMPSA_E [110]). It is thus recommended that the source term should be integrated to cover at least 90% of the expected total release with a high degree of certainty.
- it is recommended to define the LRF metric consistently with respect to an amount of radiologically weighted radionuclides. Weighting factors can be found in the INES manual for some nuclides [108] and in more detail in ICRP publications. It is recommended to use as leading (representative) isotope the following:
    - I-131 if short-term consequences are of particular interest,
    - Cs-137 if long-term (environmental) consequences are of particular interest.
- it is recommended to use LRF specifically as a strong leading indicator for long-term environmental consequences with Cs-137 as representative isotope (e.g. LRF threshold in the range of 100 TBq to 1 PBq Cs-137 (equiv.)).

Alternatively, a release metric related to the INES scale (cf. section3.5) or another limited set of release categories can be used for better describing the risk profile of the plant.

3.2 Early Release Frequency (ERF)

3.2.1 Definition of Risk Measure

Risk metric:

An early release is commonly understood to cover scenarios with releases to the environment, which happen before off-site emergency measures are effective, cf. e.g. [77], [111]. In most cases, “early” release has been defined in the context of “large early release” (LERF), see section 3.3.

There is no agreement on the following issues for the definition of “early”.

- The length of the time period for “early” in hours. Examples vary between 8 hours to 24 hours.
- The point in time, at which counting the time period for “early release” should start. In discussion are particularly: the initiating event (t=0), the declaration of a state of emergency by either the operator or the responsible authority, and the first release.

An early release metric is usually defined based on the leading isotope I-131. If the early release metric is used independently and not as LERF, then there is the question if there needs to be a lower threshold for a release to qualify as early release and at what value such a threshold should be set.

3.2.2 Discussion

Validity: Early release is – once defined – a clearly described state in the risk model.

Assigning the likelihood (distribution) to the sequence(s) ending in an early release is a clear and traceable quantification procedure. Averaging risk model results over time is a sensible and consistent way of defining a risk measure. In this regard, ERF is a valid risk measure.

ERF can be a leading indicator for acute irradiation effects to the population in the vicinity of the plant. Moreover, ERF can capture important aspects of risk to on-site personnel. ERF can address the first aspect of the risk objective as stated in WENRA's objective O3 that accidents "which would lead to early or large releases have to be practically eliminated" [111], p. 26.

As ERF addresses short-time effects, its proper definition should be with I-131 as leading isotope. Noble gas radionuclides like Xe-133 might also be radiologically relevant to short-term irradiation contributors near to the site (and also on the site).

Reliability: There is significant variability in the definitions of the large release metric, see above.

However, if the early release metric has been clearly established, it allows for – in principle – reproducible PSA modeling of the accident sequence analysis. In that sense, ERF is a reliable measure.

Consistency: ERF induces an order relation satisfying rationality criteria, if risk aggregation properties are properly considered. Basically, the measure is consistent.

Risk aggregation properties: Aggregating ERF values over multiple scenarios (i.e. frequency values) is a well-defined operation, if performed on a minimal cut set basis (or disjunctive sequences), resulting in a consistent risk measure.

With respect to risk aggregation, PSA specialists and decision makers have to acknowledge the limitations of the ERF measure. ERF identify release scenarios that develop in a short time but is not very sensitive to the amount of releases. Therefore, ERF aggregates short-terms scenarios with rather limited short-term consequences (depending on an ERF minimum release threshold) and those with high amplitude short-term consequences (e.g. a Chernobyl-type scenario). Moreover, ERF is per definition insensitive to late releases.

This might obfuscate the risk profile of the plant to some extent. PSA analysts and decision makers need to be aware of this issue.

Understandability to the PSA community: ERF is a rarely used risk measure. The variability in the understanding of "early" hampers a more common usage of this risk measure. More importantly, though, there is no agreement between countries that practically excluding "early release" (below the level of large releases) as stated in WENRA's objective O3 [111] applies to current NPP and needs to be evaluated by PSA.

Similar to the LRF metric, ERF can be defined in a consistent manner for relevant types of reactors and other sources on a NPP site.

3.2.3 Limitation

As already mentioned above, the ERF metric aggregates over the short-term release sequences with comparatively minor consequences (e.g. an INES Level 4 scenario with releases in the range of 10 to 100 TBq [108] or a filtered release scenario with releases below 10 TBq I-131) and severe releases (e.g. an INES Level 7 scenario with releases in excess of 10 PBq I-131). The likely health impact of those scenarios will be very different. This is an important limitation of the ERF measure.

Moreover, the ERF is insensitive to releases after the “early” release period. For typical accident scenarios, significant releases are likely to happen after the early period (e.g. after 24 hours), with the Fukushima Dai-ichi accident as a striking example. While these late large releases will likely have only a minor impact with respect to acute irradiation and contamination of the population, they will lead to severe consequences for the environment of the plant (cf. LRF).

3.2.4 ASAMPSA_E recommendation on early release measure definition

With regard to the ERF risk measure, we recommend the following harmonized definitions:

- the start for the “early” period of time should be consistently assigned to the declaration of a state of emergency by the responsible authority. This approach requires that operating staff do recognize that a declaration of emergency is necessary but also that they have the means to communicate this declaration or trigger such a declaration to the authority responsible for off-site emergency measures (usually a regulatory authority).
- the time period for early releases should be determined based on the time needed for performing the appropriate emergency procedures. Precautionary Action Zone (PAZ) and Urgent Protective Action Planning Zone (UPZ) [113] should be defined based on the site characteristics in advance where arrangements are made for the effective implementation of protective actions and other response actions. These zones and distance need to be established such that they provide the most effective response considering local conditions, e.g. With the lessons learned from the Fukushima Daiichi accident, short term evacuation areas would be sectors as far away as 20 km from the site [112], p. 64. Reasonable evacuation times will be depending on the population density and distribution in that area, however 24 hours seem to be a reasonable first approach.
- there should be a minimum release threshold for ERF. A good practice would be to use a maximal release activity for which no off-site protective measures (sheltering, iodine prophylaxis, and evacuation) is needed.

3.3 Large Early Release Frequency (LERF)

3.3.1 Definition of Risk Measure

Risk metric:

A large early release is commonly understood to be an “unacceptable” release of radionuclides into the environment of the plant before off-site countermeasures can reasonably be expected to be in place.

There is a wide range of specific definitions for the large **early** release metric [69], [79], e.g.

- AREVA: More than 100 TBq of Cs137 including dose weighted contribution of other elements **before or around vessel failure time**,
- Dukovany NPP (UJV, Czech Republic): >1% of Cs137 of the core inventory released to the environment **within 10 hours after the beginning of the severe accident** (Tcladding=1200°C),
- EPR Flamanville (France) : effective dose at 500 m exceeds 50 mSv (indicative criteria for evacuation, calculated with a standard meteorological model) **before 24 h**,
- French 900, 1300, 1450 MWe PWRs before LTO upgrade: release amount exceeding (**before 24 h**) those induced by a late containment filtered venting during a severe accident,
- Temelin NPP (Czech Republic): fission product fraction released early (i.e. **within several hours after accident initiator**) through large opening in the containment to the environment,
- Mochovce NPP (VUJE, Slovak Republic): > 3% of volatiles released to the environment **within 10 hours after IE occurs**,
- Bohunice NPP (Relko, Slovak Republic): > 1% of Cs137 released from the core inventory to the environment **within 10 hours after the beginning of the IE**,
- Paks NPP (VEIKI, Hungary): „Early“ means **before or shortly after vessel bottom head failure**; Large: >10000 TBq,
- SARNET recommendation: More than 3% – 10% of the core inventory in the early timeframe (i.e. **before off-site countermeasures can reasonably be expected to be in place**)

The LRF metric should be the combination of the LRF (section 3.1) and the ERF (3.2) metrics, with the following specifics:

- the definition of “early” release should be taken from ERF.
- the definition of “large” should be consistent to LRF.

3.3.2 Discussion

Large Early release is – once defined – a clearly described state in the risk model. Assigning the likelihood (distribution) to the sequence(s) ending in a large early release is a clear and traceable quantification procedure. Averaging risk model results over time is a sensible and consistent way of defining a risk measure. In this regard, LERF is a valid risk measure.

LERF can be a leading indicator for severe acute irradiation effects to the population in the vicinity of the plant. Moreover, LERF can capture important aspects of risk to on-site personnel. LERF can address the first aspect of the risk objective as stated in WENRA’s objective O3 that accidents “which would lead to early or large releases have to be practically eliminated” [111], p. 26.

As LERF addresses short-time effects, its proper definition should be based on I-131 as leading isotope. Noble gas radionuclides like Xe-133 might also be radiologically relevant to short-term irradiation contributors near to the site (and also on the site).

There is significant variability in the definitions of the large release metric, see above. However, if the early release metric has been clearly established, it allows for – in principle – reproducible PSA modelling of the accident sequence analysis. In that sense, LERF is a reliable measure.

With respect to risk aggregation, PSA specialists and decision makers have to acknowledge the limitations of the LERF measure. LERF identify severe release scenarios that develop in a short time.

In most severe accidents, the release fractions of Cs (indicating long-term consequences) and of Iodine (indicating short-term consequences) are rather similar. Therefore, each scenario which contributes to the LERF is also very likely to contribute to the LRF (assuming consistent values for “large” releases).

On the other hand, the LERF does not include late releases. Therefore, safety assessments relying exclusively on LERF may dismiss late releases.

LERF is a frequently used risk measure. The variability in the understanding of “early” and “large” did not hamper a common usage of this risk measure.

3.3.3 Limitation

LERF is frequently used, but because there is a large variety in the definition of “large” and “early”, it is nothing more than an indication that under the local conditions severe health effects must be considered with a certain frequency, and without possibility for efficient plant-external mitigation measures.

LERF is per definition insensitive to late releases. Therefore late releases would not be identified. If the three Fukushima core melt accidents had been subject to a time grouping, they had probably all been binned into “late” releases. This is adequate because precautionary emergency measures could be and had been initiated outside of the plant.

However, a PSA focusing exclusively on LERF ignores the large releases occurring later in these sequences. Applying LERF as the only result of a PSA is obviously misleading and inacceptable.

Therefore, LERF is a valid risk measure, but it must not be used as the only risk measure.

3.3.4 ASAMPSA_E recommendation on LERF

Since LERF is widely used, but not precisely defined, there is urgent need for a harmonized definition. Basically, LERF is based on a qualitative definition (e.g. release of a radioactive quantity which can cause acute health effects before any plant-external mitigation measures are possible). However, this example for a qualitative definition needs significant input from tasks beyond L2 PSA (health effects assessment, availability of external countermeasures), which are hardly available in a L2 project.

Therefore, for practical reasons a definition is recommended in the form of precise metrics (e.g. release of more than 100 TBq of I-131 less than 8 hr after declaration of emergency). A suitable international working group should agree on such a metric. However, given the long lasting wide application of LERF in different local definitions (some of them encoded in rules and regulations) there is little hope for harmonization.

3.4 Release Categories Frequency (RCF)

3.4.1 Definition of Risk Measure

The concept of "Release Categories" is a very well-known and a widely used approach in PSA L2 in order to describe consequences of severe accidents. A practical guide to defining and applying release categories is provided in [5]. Part of the following text is taken from this reference.

Many of the end states of the containment event tree are identical or similar in terms of the phenomena that have occurred and the resulting release of radioactive material to the environment. Similar end states should be grouped or binned together to reduce the number of distinct accident sequences that need analysis. In order to do this a set of attributes has to be specified that relate to the possible transport mechanisms of the radioactive material and failure mechanisms of the containment that can be used to characterize the release categories. Typical attributes that have been used in specifying the release categories for light water reactors are shown in Table 7 of [5]. Typically, there are around five attributes. The most important one is the containment failure mode, and each attribute may have two to ten variations (e.g. containment intact, containment is vented, containment fails late, containment fails early, containment is bypassed, containment is not isolated). In principle, this process can generate a very large number of release categories, but in practice, most PSA L2 manage to limit the number to around ten release categories.

Reference [2] (ASAMPSA2, Volume 1, page 98) provides some examples for the presentation of the results.

3.4.2 Discussion

Release categories are a good indicator for the validity of the DiD concept: It can be seen how many barriers fail in which way and with which frequency, and whether barriers remain intact. Since release categories do not imply analyses of radionuclide behavior (which may be difficult to track), they will consume less resources and entail less uncertainty than source term based results. Therefore, they are useful indicators for the plant resilience, and a necessary basis for the assessment of source terms.

However, release categories as an end state of a L2 PSA cannot be considered satisfactory, since they cannot provide information on accident consequences in themselves, and all quantitative risk targets are based on some type of radioactive release quantification.

3.4.3 ASAMPSA_E recommendation on RCF

Release categories are a well-known and widely used concept which should be used for:

- Assessing the plant response to the challenges of the severe accident,
- checking the DiD concept under severe accident conditions,
- guiding the assessment of radioactive releases through various release paths.

From an "extended" PSA point of view, there is no need for modifying the existing approach for release category definition and use. A particular case, however, would be the analysis of multiple releases from a multi-unit site undergoing more than one severe accident. No good practice for defining release categories exists for these cases.

3.5 Frequency of Loss of containment functions

This section is an extract from Reference [2] (ASAMPSA2, Volume 1, page 96).

3.5.1 Definition of Risk Measure

In the following paragraphs, the term "containment failure mode" concerns all release paths in the case of an accident with a loss of the containment function. For example, a steam generator tube rupture is considered as a "containment failure mode" although in reality it is the bypass of an intact containment.

<u>Example of risk metric: First containment function failure</u>

An approach for presenting the results of a L2 PSA consists of defining the APET outputs (release categories) with the <u>*first*</u> failures of a containment function during the accident progression. This approach is simple to perform with APET tools that take into account the chronology of the accident but may be more difficult if the chronology is not explicitly addressed (L1PSA APET tools).

For example, the frequency of an accidental sequence that leads to the containment failure modes Mode 1 <u>and</u> Mode 2 will exclusively contribute to the frequency of the containment failure mode Mode 1 if it occurs before failure Mode 2.

This presentation may not be correlated to the severity of the accident (if the worst containment failure is the second one, it will not appear) and must be used carefully.

<u>Example of risk metric: Dominant containment failure mode</u>

If the L2 PSA results exhibit sequences including several containment failure modes (for example a leak into the reactor building followed by a basemat penetration), it may be useful to define a scaling of the different containment failure modes related to their severity. The definition of severity may consider both the amplitude of release and the accident kinetics. For example an induced steam generator tube rupture is often considered as one of the worst situations for a PWR as it may combine a short delay before atmospheric radioactive release and high amplitude of release.

This presentation can be considered as the standard way for a result presentation of a L2 PSA. However a clear definition on the scale of "dominant" may not be easy. For example, it is not obvious how to compare an early containment failure with limited leak size to a late containment failure with large leak size. The main limitation is that the dominant containment failure modes mask other containment failures in a sequence. This can bias the L2 PSA applications, especially if some conservatism has been introduced in the APET assumptions related to some "dominant" containment failure modes.

Example of risk metric: Individual containment failure mode

For the L2 PSA applications, it may be useful to separately calculate the frequency obtained for each containment failure mode in order to discuss the interest of specific plant improvements regarding the specific contribution of the considered containment failure modes to the risk.

This should be also used to demonstrate that some specific risks can be excluded: for example, if the frequency of late containment failure by hydrogen combustion during MCCI phase was found to be very low, it should be checked that this result is not obtained because previous failure modes have masked it.

For example, the frequency of an accidental sequence that leads to the containment failure modes Mode 1 and Mode 2 will contribute to both of the frequencies of the containment failure modes Mode 1 and Mode 2. In addition it may be of interest to document the combinations of failures that occur. For example, if a containment bypass is combined with a basemat melt through, the frequency of simultaneous occurrence for both failure modes should be given to complete the information.

For each quantification (or each Monte Carlo run), the sum of each individual containment failure frequency plus the frequency of situations without containment failure, may largely exceed the L1PSA total frequency if the APET allows the quantification of multiple containment failures in each accident sequence. This result has to be clearly explained to the final L2 PSA user.

3.5.2 Limitations

In case of a core melt accident, loss of the containment function indicates that practically no engineered safety barrier exists between the melting core and the environment. Therefore, this is synonymous to a very severe release to the environment. But within this category, the release quantity will vary depending of the properties of the accident and its progression, e.g. timing of the release (influencing the degree of deposition and thus retention inside the building volumes), availability of mitigating actions (e.g. sprays, filtered ventilation in buildings), and status of buildings outside of the containment (e.g. intact or damaged by external hazard or by hydrogen burst). The variation of the released quantities can easily attain an order of magnitude. If such uncertainty is tolerable, the frequency of loss of containment function is a valuable measure.

3.5.3 ASAMPSA_E recommendation on measure for loss of containment function

There is already a widespread good practice in L2 PSA to identify the frequency of the loss of containment functions. The application of this measure is further encouraged, with the following comment: It is recommended to distinguish for core melt sequences:

- Intact containment with design basis leakage,
- Intact containment with filtered venting,
- Loss of containment function due to a leak or rupture of the containment structure,

- Loss of containment function due to failure of the containment isolation,
- Loss of containment function due to bypass through interfacing systems (for BWR including non-isolated break of feedwater or steam lines outside of the containment),
- Loss of containment function due bypass through steam generator tube leak (PWR only).

It may be interesting to introduce an approach, which has similarity to the well-known core damage frequency (CDF) concept of L1 PSA (See Section 2): Define a “Containment Failure Frequency” (CFF). The CFF would comprise all CDF sequences where the containment function is lost. The CFF could attain the same weight in safety assessment as the traditional CDF. One could imagine assessing plant improvements or comparison with safety targets in terms of CFF. Of course such a general property cannot capture all relevant attributes, but the same applies for the very popular CDF measure. This shortcoming did not prevent the CDF measure from becoming the best known and worldwide accepted measure for severe accidents.

3.6 Frequency of “Kinetics Based” Release Categories

Examples of this risk metric are provided in Reference [2] (ASAMPSA2, Volume 1, page 100). It’s either based on containment failure time or delay before obtaining an activity release limit depending of the containment failure mode.

3.7 Functional and Phenomena Based Risk Metric

3.7.1 Definition of Risk Metric

For French PWR safety reassessment, EDF has chosen a risk metric that focuses on safety insights instead of precise source term quantification. The aim is to get a functional analysis of the risk in order to target area for safety improvement, without focusing on the quantification of the source term depending on specific release hypothesis (leakage rates, iodine behavior, scrubbing factor…).

To meet this objective, EDF has defined 7 “functional” release categories:

- 5 release categories for atmospheric releases,
- 2 release categories for underground releases.

The atmospheric and underground releases are assessed for each sequence of each Level 2 PSA event tree. This means that for each sequence of a Level 2 PSA event tree two consequences are assessed: the first one is a release category for atmospheric releases and second one is a release category for underground releases.

- **Atmospheric Release Categories**

5 functional atmospheric release categories are defined related to the emergency countermeasures characteristics:

- R1: Large Early Release (containment break or bypass before 24h) => emergency countermeasures are not sufficient to protect the public due to the short delay and the large amount of release.
- R2: Large Late Release (containment break after 24h) => emergency countermeasure are not sufficient to protect the public due to the large amount of release.
- R3: Late filtered releases (Filtered Containment Venting after 24h) => this Release Category is the reference one for the application of Stringent Countermeasures (evacuation of the public up to 5 km and sheltering up to 10 km).
- R4: core melt releases without containment loss (or bypass) and without Filtered Containment Venting opening => Release Category for Limited Countermeasures (sheltering or limited evacuation).
- RD: Design Basis Accidents (LOCA, SGTR... *without core melt*) => very limited or no countermeasures.

According to these definitions, correspondence with international L2 risk metric can be provided:
R1 is associated to “Large Early Release Frequency (**LERF**)”. R2 could be seen as a “Large Late Release Frequency” (LLRF, but neither defined nor used in international L2 risk metrics), and (R1 + R2) is associated to Large Release Frequency (**LRF**).

Additionally, as these release categories cover a large scale of release (even possibly different orders of magnitude), it is necessary to include additional functional information for safety analysis. For example:

- For R1 release category: release contribution from ß mode (with distinction between equipment hatch releases -if equipment hatch has a direct opening to the outside- and other penetration releases), release contribution from Severe Accident Phenomena involved (for example H2 or steam explosion risk), release contribution from a LOCA, from SGTR…
- For R2 release category: release contribution from Filtered Containment Venting failure, from H2 risk in inter containment space....

- **Underground Release Categories**

There are only two functional underground release categories (intact or failed basemat), as it is stated that the long term consequences of radioactive releases through basemat are difficult to manage.

- RP: basemat failure
- RI: intact basemat

<u>Illustration</u>

According to the above risk metric definitions, an illustration of the results (Risk Measure) that can be provided from a level 2 PSA is given on the figure thereafter:

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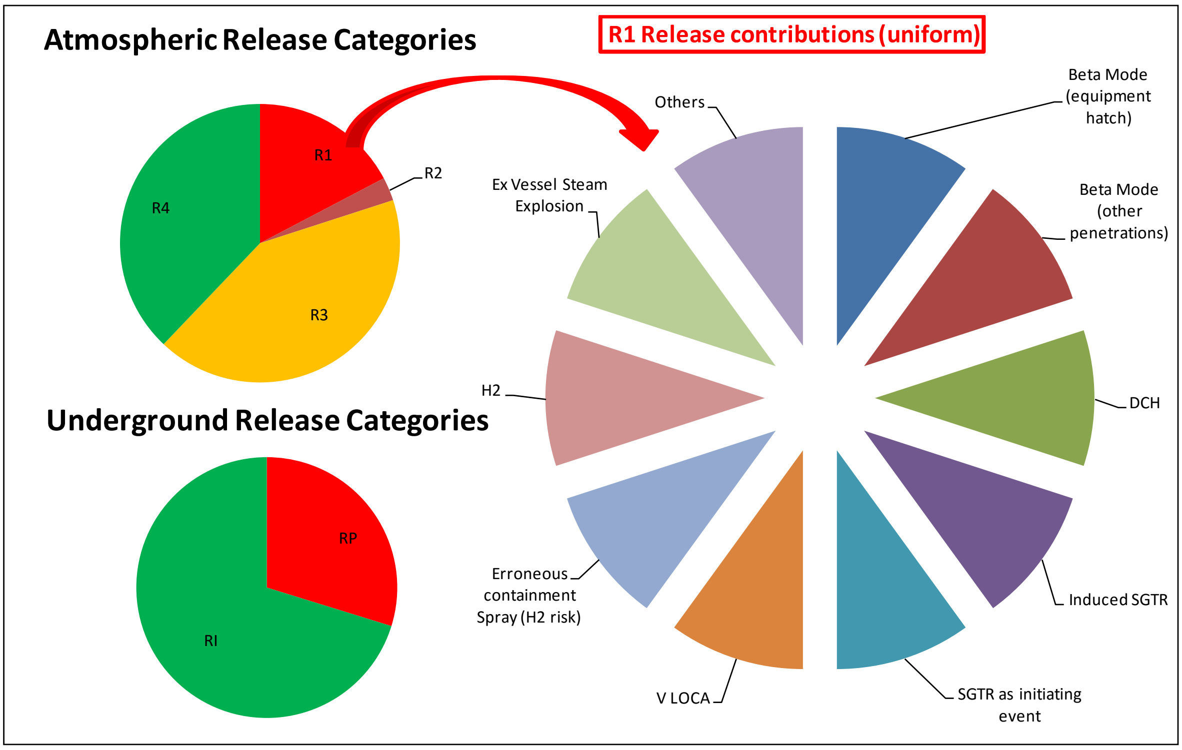


Figure 2 : Example of Results Provided by L2 PSA

While analyzing results as presented as above and additionally analyzing the related initiating event family for each release category, it is quite easy to efficiently define priorities for safety enhancement, depending on the objectives fixed in the safety reassessment context: plant modifications, operating procedures, human training….

3.7.2 Limitation

This risk metric is not adequate if Level 3 assessments are required.

3.7.3 ASAMPSA_E recommendation on phenomena-based measure

The functional risk metric developed by EDF is as a pragmatic and industrial way to focus on safety insights and improvements instead of being polluted burdened by specific release hypothesis and source term calculations.

This risk metric is easy to understand, even for non-specialists. It is suitable for hazards, but it should be associated with a hazard extension to avoid inappropriate summation / aggregation between inhomogeneous risk evaluations.

3.8 Frequency of Release Based Categories

The previous section presents risk metrics that provide information related to the failure of the different containment functions during a severe accident. This is a “system-oriented” presentation of results. Another approach is to present the results through the level of consequences, for example the total atmospheric release of activity (Bq), with a containment failure. Examples of this risk metric are provided in Reference [2] (ASAMPSA2, Volume 1, page 98).

3.9 Absolute Severity Metric

This section is an extract from Reference [2] (ASAMPSA2, Volume 1, page 108).

L2 PSA aims to calculate the possible sequences of release and their frequencies. The releases are supposed to be defined by their amplitude (expressed in Becquerel for each important isotope) and their kinetics. Any assessment of consequences beyond the plant perimeter is considered to be part of L3 PSA and is not state-of-the-art for L2 PSA.

In the practical application, the L2 PSA analysts need to make the link between the amplitude and kinetics of release and the consequences of the accident before deriving relevant conclusions. This may lead to the need for L3 PSA but for many organizations the development of a full-scope L3 PSA (including assessment of health and environmental impact, taking into account all the local conditions) would be a huge task regarding internal resources.

To overcome this difficulty, some organizations have developed extended “L2 PSA” (“L2+ PSA”) and have added some simplified assessments of the release consequences to help in the presentation of the conclusions. For example, the L2 PSA developed by IRSN for the French 900 MWe and 1300 MWe PWRs is a “L2+ PSA” and include, for each Release Category, a calculation of the atmospheric dispersion and dosimetric impact (with standard meteorological conditions and without any assumptions regarding counter-measures).

GRS has performed a L2 PSA for a German 900 MWe BWR. Parts of the final result consisted of a frequency distribution of “radiological relevance”. For this purpose, the APET was linked to a simple and fast running source term assessment module. This module produced a source term for each individual sequence of the APET. The source term considered four different radioisotopes (I-131, Cs-137, Te-132, Kr-88). For each of these isotopes a relative radiological impact per Bq of release has been defined based on short term health effects. Finally, the total radiological relevance of the combined release of all four isotopes has been calculated for all source terms. Combined with the frequency of source terms, a frequency distribution of the radiological relevance could be produced.

The objective of this chapter is to describe some complementary risk measures / safety indicators that may be calculated by a L2+ PSA. This part should not be considered as state-of-the-art but it proposes some ideas for a multi-criteria analysis and some flexible views regarding the link between risk measures and quantitative safety goals.

3.9.1 Definition of Risk Measures

The main difficulty in assessing the severity of an accident is to take into account the different nature of the potential accident consequences:

- early fatalities,
- early injuries,

- late cancer fatalities and related severe diseases or injuries,
- permanent or temporary loss of land,
- number of persons relocated temporarily or permanently,
- the ground contamination (soil surface, groundwater, river ….),
- the loss of economic resources (industry, agriculture …),
- the negative image impact (locally, regionally, nationally depending on the amplitude of the consequence),
- the negative impact for nuclear industry (for the specific plant type but also the whole industry …),
- etc.

A precise assessment of all potential accident consequences for every release category would need the development of L3 PSA, and would highly depend on the plant location.

For the simplicity and the clarity of the presentation of L2 PSA results, there is an interest in building an “accident absolute severity metrics” that would provide an indication of the severity of an accident without any considerations related to:

- the location of the plant (the local meteorological conditions, the population density, the economic activities, and the environment are taken into account to assess the “absolute” severity of the accident),
- the possibility and the efficiency of the emergency actions for the protection of the population.

Such “absolute severity metrics” would address only the NPP safety features without any consideration of offsite environment and the emergency response prepared by the local and national authorities reflecting thus safety of a plant only as a complex technical facility according to IAEA definition of nuclear safety corresponding to a source of radioactivity. The facility in this respect is limited and confined by containment as the last physical barrier of high integrity as part of currently accepted DiD approach. It could be named an “intrinsic reactor severity scale”. It is particularly appropriate for the utility (or vendor) analysis when trying to improve the NPP safety features.

The following approaches provide some examples that could be used.

<u>Application of the INES scale</u>

A solution may be to use an existing scale (example: INES scale [108]):

<u>Categorization based on INES scale</u> may be used for the assessment of the “absolute severity metric”, within L2 PSA without need of performing L3 PSA (see example in Section 3.10.2). In this case the existing grades of INES scale originally developed for real accidents is used as the risk parameter for grouping sequences by their severity based on releases aiming to assess their contribution to total risk (i.e. consequences with their frequencies) allowing thus also to judge besides other things if a plant is balanced. Conversion of releases to a range of absolute consequences (example see in Table 6) makes possible to compare Severe Accident risks with other

industrial risks. This can be useful to justify that no significant addition to other industrial risks exists (see IAEA basic safety principles for nuclear power plants, INSAG-3 Rev.1, INSAG-12[14].)

Categorization based on projected doses calculations

Each release category obtained from a L2 PSA is associated, for each considered isotope, to one set of kinetics and amplitude of atmospheric release. It may be useful in the final presentation of the results to calculate the radiation impact of the release for different distances and delays with some standard meteorological conditions.
Such a presentation of results may help in the communication of L2+ PSA results. For example the following can be calculated:

- The projected effective dose (i.e. the dose likely to be received by an individual through all pathways when no protective actions are implemented) at different distances (e.g. 2, 10, 20, 50 km) and time scales (e.g. 15 days, one year, 50 years),
- The thyroid dose at the same distances and time scales.

When using one criterion (for example projected dose at 2 km, 15 days), it becomes possible to classify the different accident scenarios in terms of risks (frequency x consequence) and to have a relatively clear indication of the severity of the accident regarding health effects.

Categorization based on ground deposit of fission products

Long-term ground contamination by aerosols like Cs137 constitutes a significant impact of a NPP severe accident. It may be useful for the final presentation of the results to calculate the deposition of Cs137 (or other radionuclides) on the ground, at different distances of the NPP (e.g. 2, 5, 10, 20, 50 km). The results can be compared to the zoning criteria that may be used for the post-accidental management. Such information can provide a relatively clear indication regarding the long term impact of the considered accidents.

3.9.2 Discussion

The following are some considerations that should be taken into account in the evaluation of accident absolute severity metrics:

Specific information linked to emergency planning:

L2 PSA results can be used to discriminate between the sequences that can be managed by the emergency offsite measures and those which can be not. This compatibility depends mainly on both the kinetics of the accident and the spatial extension of the counter-measures.

[14] From IAEA “The protection system is effective as stated in the objective if it prevents significant addition either to the risk to health or to the risk of other damage to which individuals, society and the environment are exposed as a consequence of industrial activity already accepted. In this application, the risk associated with an accident or an event is defined as the arithmetic product of the probability of that accident or event and the adverse effect it would produce.”

If the L2 PSA is extended to some atmospheric dispersion calculations and projected doses, then it is recommended that the following should be provided for each release category:

- the time scale available before reaching some counter-measure criteria (projected dose for sheltering or evacuation, thyroid dose for iodine prophylaxis),
- the distance to which each short term countermeasure (sheltering, evacuation, iodine prophylaxis) should be applied.

Both distances and time scales can be compared to the provision of the emergency plans by the L2 PSA analysts. Each release category can be qualified as “compatible or not” to the emergency plans.

Diagrams Frequencies-Consequences

All PSA assessments of accident consequences (absolute severity scale, projected doses (calculated at a defined distance), ground contamination (Activity of Cs^{137} deposit, annual dose induced by deposit) versus frequency can be presented as “cumulative probability for exceeding a certain consequence vs extent of Consequences” or “RC frequency x extent of Consequences diagram”.

3.9.3 ASAMPSA_E recommendation on Level 2+ PSA

Accident absolute severity metrics would provide an indication of the severity of an accident and are valid metrics for risk assessment. Some are suggested in this section however their limitations should be acknowledged.

3.10 Integral Risk or Total Risk Measures

The idea of integral risk is based on the definition of risk as multiplication of frequency and consequences.

3.10.1 Definition of Risk Measure

A measure of the “total source term risk” can be obtained by a formula like:

Total risk = $F_1 \times A(RC_1) + F_2 \times A(RC_2) + \ldots.. + F_n \times A(RC_n)$,

where n is the release mode, F_n is the frequency of the release category RC_n for the n mode and $A(RC_n)$ is the amplitude of the consequence calculated for the release category RC_n (in Bq).

This type of evaluation may be applied whatever the nature of consequence calculated but this has significance only if release categories are defined such as:

$F_1(RC_1) + F_2(RC_2) + \ldots.. F_n(RC_n)$= Total L1 PSA fuel damage frequency.

This can be applied for each “point” of an APET quantification, or each run in the case of Monte-Carlo simulation.

In L2 PSA consequences are typically calculated in terms of activity releases (in Bq) to the environment. It needs to be defined which isotopes should be considered (e.g. just I-131 and Cs-137, or a more complete set of radionuclides). In addition, the individual isotopes have different consequences in terms of health effects per unit of activity released. If for each relevant isotope a suitable factor can be defined which characterizes its relative health effect, the resulting total risk would be a measure which partly incorporates L3 PSA issues. It must be noted that best practices recommend to include all radionuclide groups including noble gases in the assessment.

An example is proposed in the next section. It is a proposal by CCA (organization partner). The methodology was started under ASAMPSA2 (Chapter 6.4 of ASAMPSA2; Reference [2], Volume 1, page 122) and developed much further, see Reference [43] and [114].

3.10.2 Common Risk Target methodology, CRT (Proposal by CCA)

3.10.2.1 Definition of CRT Risk Measure

The CRT methodology and risk target were developed to try to find a parameter complying with existing safety requirements and criteria published by IAEA. The following boundary conditions were taken into account by developing the methodology and deriving the parameter:

a. It is the parameter representing risk (defined as the product of frequency and consequences expressed in Bq).
b. It should follow constant risk principle, graded approach and it should represent balanced plant with no extreme contributions of particular sequences to the total risk.
c. It should comply with multi-unit site requirement.
d. It should be a quantitative parameter.
e. Its quantitative value should be comparable with risks stemming from other industrial activities.
f. It should take into account all radiological sources in the site.
g. It should be objective as a contrary to subjective.
h. It should comply with harmonization requirements.
i. It should be the maximum admissible risk value for a safe plant.
j. It should guarantee adequate scientific level and credibility by being technically derived not only designated. (And as much as possible, also accepted by wider scientific public.)

The INES scale (see Table 5 and Figure 3) consists of grades indicating severity of an accident, from which the higher ones are based on “I-131 equivalent” releases in Bq. The expression “I131 equivalent” means that all released nuclides are to be converted to I-131 using radiological equivalence (multiplication factors) corresponding to their health impact as the fraction/multiple of I-131 which is in this sense set to 1 [108], Appendix I]. The corresponding INES grade should be only subsequently assigned to the total value of I-131 equivalent releases in Bq. This should be done while evaluating the severity of real accidents, for which except the extent of releases also other aspects are taken into account for final INES evaluation. (Details see the reference [108]). The same principle for release grouping in Bq by INES grades is used in CRT methodology for L2 PSA. Some examples of radiological equivalences are available in INES Manual [108]. The International Commission on Radiological Protection (ICRP) keeps updating the tables for most possible conversion factors [116]. They are also inbuilt in some calculation tools e.g. MACCS2.

Table 5: General Criteria for Rating Events in INES

Description and INES Level	People and the environment	Radiological barriers and controls at facilities	Defence in depth
Major accident Level 7	- Major release of radioactive material with widespread health and environmental effects requiring implementation of planned and extended countermeasures.		
Serious accident Level 6	- Significant release of radioactive material likely to require implementation of planned countermeasures.		
Accident with wider consequences Level 5	- Limited release of radioactive material likely to require implementation of some planned countermeasures. - Several deaths from radiation.	- Severe damage to reactor core. - Release of large quantities of radioactive material within an installation with a high probability of significant public exposure. This could arise from a major criticality accident or fire.	
Accident with local consequences Level 4	- Minor release of radioactive material unlikely to result in implementation of planned countermeasures other than local food controls. - At least one death from radiation.	- Fuel melt or damage to fuel resulting in more than 0.1% release of core inventory. - Release of significant quantities of radioactive material within an installation with a high probability of significant public exposure.	
Serious incident Level 3	- Exposure in excess of ten times the statutory annual limit for workers. - Non-lethal deterministic health effect (e.g. burns) from radiation.	- Exposure rates of more than 1 Sv/hr in an operating area. - Severe contamination in an area not expected by design, with a low probability of significant public exposure.	- Near accident at a nuclear power plant with no safety provisions remaining. - Lost or stolen highly radioactive sealed source. - Misdelivered highly radioactive sealed source without adequate radiation procedures in place to handle it.
Incident Level 2	- Exposure of a member of the public in excess of 10mSv. - Exposure of a worker in excess of the statutory annual limits.	- Radiation levels in an operating area of more than 50 mSv/h. - Significant contamination within the facility into an area not expected by design.	- Significant failures in safety provisions but with no actual consequences. - Found highly radioactive sealed orphan source, device or transport package with safety provisions intact. - Inadequate packaging of a highly radioactive sealed source.
Anomaly Level 1			- Overexposure of a member of the public in excess of statutory limits. - Minor problems with safety components with significant defence in depth remaining. - Low activity lost or stolen radioactive source, device or transport package.
	No safety significance (Below scale/Level 0)		

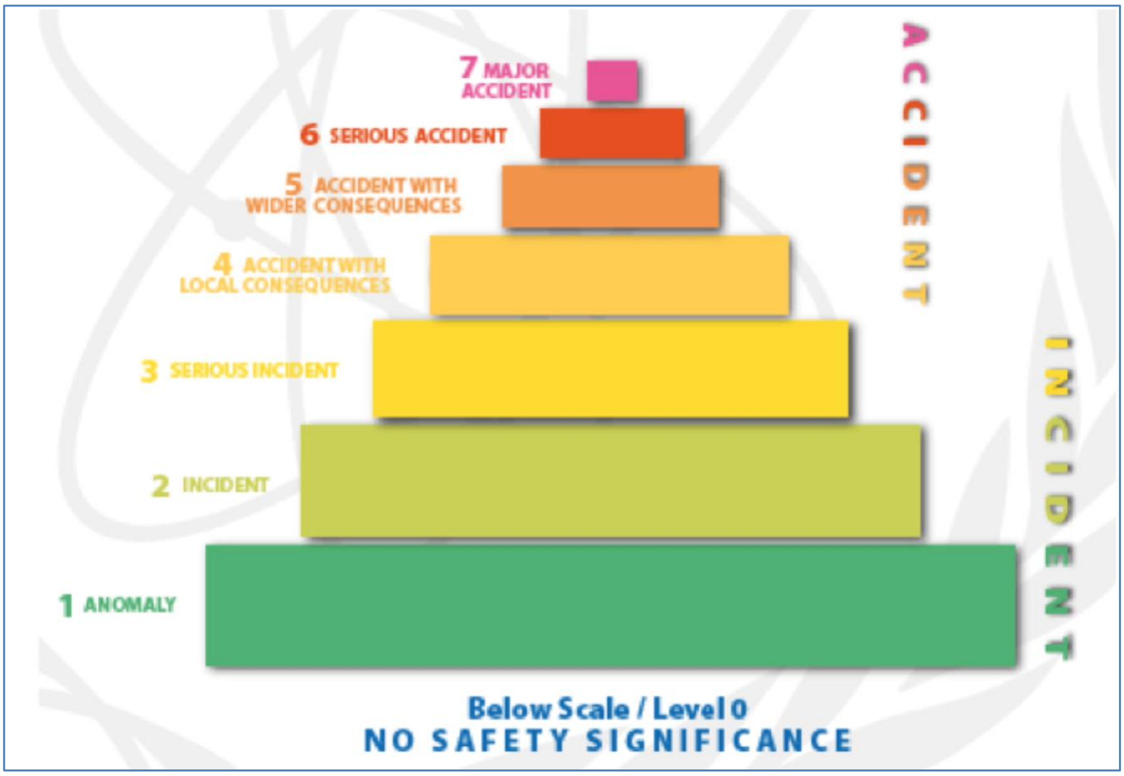


Figure 3 : INES Scale

3.10.2.2 Assessment of Absolute Consequences and Risks per Accident

Table 6 shows average expected results or rather range of expected results for severe accident consequences based on PSA L2 analyses of different plants in different sites (e.g. [117], [118]) taking into account also Chernobyl and Fukushima analyses [119]. The results show estimate of absolute consequences and risks following a severe accident [120]. Only deaths, both acute and late are shown, and extent of lost land. Late deaths are shown as risk in terms of one in number of world population, i.e. in term of global risk referred to quantity of population. It is obvious from Table 6, that acute fatalities as one type of consequences of severe nuclear accidents are negligible in comparison with late fatalities and other long term risks (injuries are not included) and therefore, the parameter of acute fatalities as a measure of severity of accident is not relevant. From the results obtained the following can be summarized:

a) Range of risks is population independent (i.e. not dependent on population densities) because they are normalized to population in an area, e.g. they are independent of the density and can be exported to any site.

b) Calculated risks are not site dependent - because of probabilistic weather modeling - any number of persons at a given location will be exposed to exactly the same amount of radiation, which only depends on dispersion coefficients.

c) Range of risks is plant independent - any plant that has an accident ending in core damage not treated and stopped in-vessel will release (no matter what type of plant it is) I-131 equivalent in quantities between 1E16 and higher Bq (INES7).

d) Range of calculated risk is weather independent - they depend only on dispersion, which is calculated probabilistically rotating the weathers around 360 degrees, and thus risk does not depend on the direction of the wind since all directions are taken into account.

e) One lifetime excess death corresponds approximately to 1 TBq of I-131 equivalent released. (Table 6)

For the results shown in Table 6 total effective doses were calculated to provide the Total Effective Dose equivalents using a simplified method developed within the study [118]. A more sophisticated model was used in the study [117] to provide quick estimates of consequences for different sites and types of power plants. The work [118] was performed to answer the question if risk measures based on activity of releases are an effective surrogate for estimation of offsite consequences - the MACCS calculations were compared with the results of calculations based on activity of releases. The MACCS code [120] was used to calculate weights for activity of releases for each of the radioactive groups, and absolute weights were calculated for two sites and weathers. Then, relative weights were compared for the two sites, and found to be insensitive to population, weather patterns, and topography. The absolute weights were then used to estimate offsite consequences on the basis of activity of releases for the various groups, and the results were compared to MACCS calculations for 14 plant specific source terms [117]. A separate assessment was also performed using only the total activity resulting from aerosol. Results from the three sets of calculations are very close for all the offsite consequences which have been considered. Therefore, it can be concluded that risk measures based on activities of releases are a good surrogate for the performance of calculations of offsite consequences.

Specifically, the method from [117] used a comparison of results of MACCS calculations for a reference source term for BWR reactor with 3600 MW(Th) operating power for the Surry site (MACCS default site) and Central European site, respectively. The reference source terms were releases of a fixed fraction of each radionuclide group under the assumption of early energetic releases and late slow releases. By admission of the code developers, MACCS using 60 radionuclides for releases can estimate between 90 and 95% of prompt and chronic consequences calculated for the full 400 to 500 radionuclides present in a typical LWR core [120]. The "equivalency" data (effects/Bq) was determined separately for all types of consequences.

Table 6 : Assessment of Absolute Consequences and Risks per Accident

Releases		Absolute Consequences			Individual Risks per Accident			
						Excess Deaths		
								GLOBAL
INES	I-131 equiv. [Bq]	Acute Fatalities	Lifetime Excess Deaths	Lost Land [km^2]	Acute within 10 km	within 10 km	within 40km	Excess death for $> E^8$ population (within 1600 km)
4	< 200	0	0-100	0 - 5	< 1E-4	< 1E-4	< 1E-4	< 1E-5
5	200-2000	0	100 - 1000	5 - 100	1E-4 - 1E-3	1E-4 - 1E-3	1E-4 - 1E-3	1E-5 - 1E-4
6	2000-20000	0-1	1000 - 10000	100 - 500	1E-3 - 5E-3	1E-3 - 5E-3	1E-3 - 5E-3	1E-4 - 5E-4
7 ("Large")	> 20000	0-100	10000 - > 1000000	500 - > 10000	5E-3 - 5E-2	5E-3 - 5E-2	1E-3 - 5E-2	5E-4 - 2E-2

As discussed previously and in in Section 1, there is no common understanding of "acceptable" risk and each domain of economy/industry has its own explanation. Aiming to find a Common Risk Target to compare the PSA results for evaluation of safety of nuclear facilities the three following major postulates based on the IAEA definitions are used within the CRT methodology derivation:

A. Risk is the product of frequency and consequences [121].
B. Risk from nuclear facilities should not significantly enhance the already accepted risks from other industrial activities [121].
C. The total risk should be a constant (i.e. the frequency of accidents resulting in larger consequences should proportionally decrease with consequences in accordance with the Farmer curve, or IAEA INES scale) [121][122].

For the postulate C the Farmer curve is used, that represents constant risk. For better demonstration of the concept of risk the following table is used:

Table 7 : Concept of Risk

	Consequences		
Frequency	**Low**	**High**	**Very high**
Low			
High			
Very High			

The grey fields in the table represent the acceptable results from the point of view of total risk and they are obtained by multiplication of the corresponding cells in the table. The border may be defined as the highest product coming from combinations very high-low, or high-high. This way, the Farmer curve is defined, i.e. a constant acceptable risk is defined. Now, the task is to find the corresponding levels of Low, High and Very high frequency and consequences.

For the purpose of finding grades of consequences the IAEA INES scale may be used, since it is graded according to the amount of releases during an incident/accident similar way as shown in the table above, and therefore for the levels Low, High and Very High the INES grades INES5, INES6 and INES7 can be taken which differ in principle by one order of magnitude. According to [121] Principle 3, 3.15 "Safety has to be assessed for all facilities and activities, consistent with a graded approach". Here it should be reminded that risks coming from Severe Accidents of nuclear power plants are part of total nuclear risks that are related also to mining, manufacturing, reprocessing, transport, normal operation, design basis accidents, decommissioning and disposals.

The following measures of risk are generally considered while evaluating total risks: individual risk, societal risk, environmental risk and economic risk, therefore also for ASAMPSA_E and CRT methodology derivation the same consideration is applied (see Chapter 1.3 and Equation 3.10.1 below for Risk of Severe Accident RSA).

IMMEDIATE *LONGTERM*

$$R_{SA} = R_{indiv} + R_{soc} + R_{enviro} + R_{costs} \tag{3.10.1}$$

Acute deaths
Lifetime deaths
Environmental damage.
Wide-spread econ.damage

Acute injuries
Loss of property
Loss of employment

Societal disruption
Lifetime hospit.
Moral/Psycho risk
Lifetime health problems

Loss of non-agricult. land
Loss of waters .
Loss/injury water ecosyst.
Loss/injury aerial ecosyst.
Loss/injury terrest. ecosyst.
Condemned area
Interdicted area
Crop disposal
Milk disposal

Lost production
Repair
Clean-up
Actuarial injuries

represents LERF,

represents generally evaluated Industrial risk,

represents part of Common Risk Target (to be comparable to industrial risk expressing mostly deaths),

represents total scope of Common Risk Target (minimizing releases, it means all risks).

As it comes from statistics on real accident in Chernobyl with existing long-term 30-year observations and measurements (e.g. [123] and many others), acute fatalities are negligible in comparison with other long-term consequences following a nuclear accident, which is apparent also from Table 6 based on MACCS2 calculations. The formula above with its graphic distinction of particular components of total risk shows LERF - being the widely used risk measure in L2 PSA focusing on early releases. LERF is tied more or less to releases of I131, which are considered to be the most devastating in terms of immediate (or rather delayed) human health impact ONLY. This is caused by its nature as far as chemical and radiological characteristics influencing thyroid. Thus, from the point of view of absolute consequences usually expressed in fatalities, LERF is not an indicator to relay upon in terms of safety and “intrinsic” effects of a severe accident. It should be noted, that LERF was not developed for safety purposes, but for radiation protection purposes.

3.10.2.3 Level 2 Risk Assessment (Containment matrix)

The containment fragility analysis identifies various containment failure modes due to various loads. Typical containment loads analyzed are for instance: in-vessel steam explosion, ex-vessel steam explosion, RCS blow-down and generation of non-condensable gases, loads due to hydrogen and carbon monoxide combustion, due to high-pressure gas releases, direct containment heating, basemat melt-through etc.

Containment accident progression is analyzed, where the methodology of Containment Event Trees/Accident Progression Event Trees (CET/APET) is normally used. The APET produces a large number of end-states and since some of them are identical or similar in terms of key release attributes, they are grouped together with respect to similar radiological characteristics and potential off-site consequences. This grouping is referred to as release categories, release bins, release modes, release classes, or source term bins.

Typical form of the results referring to PDS and release categories is a containment matrix, one example follows in Table 8.

Table 8 : Example of Containment Matrix

PDS	Release bin categories/Release classes RC										PDS Freq.
	SGTR	Early Cont. Failure			Late Cont. Failure	Bypass		Cont. Unisol.	Basemat melt	No CF	
		Type1	Type2	Type3	Type1	Type2	Type2				
1	F_{SGTR1}										f1
2	F_{SGTR2}										f2
n	f_{SGTRn}										fn
BinFreq./year (sum (R1:Rn)	f_{SGTR}	f_{ECF-T1}	f_{ECF-T2}	f_{ECF-T3}	…					f_{NoCF}	Sum(f1:fn) =sum(C_2:C_{m+1}) **=FDF**
Release Class	**RC1**	**RC2**	**RC3**							**RCm**	

The risk is assessed as follows:

R = f × c, where

R is Risk,

f is frequency,

c are consequences.

After a containment matrix is obtained (see Table 8), within the L2 PSA analysis, the environmental release quantities (source terms) associated with each release categories (containment failure modes) are estimated. In this part, all radionuclides are grouped on the basis of similarities in the thermodynamic and chemical properties of the various radionuclides. Source terms are expressed in fractions of core inventories assessed for each radionuclide group based on the supposed status of the core at the time of accident. As an example of radionuclide groups used in L2 PSA the following set of groups can be given:

- noble gases represented by Xe,
- volatile halogens represented by CsI (I),
- volatile alkaline metals represented by CsOH (Cs),
- volatile chalcogens represented by Te,
- semi-volatile alkaline metals represented by Sr-Ba,
- refractory early transition elements represented by Mo and
- refractory platinoids, tetravalents and trivalents represented by Ru-La-Ce.

Release categories characterize major classes of accident sequences in terms of the nature, timing, and magnitude of the release of radioactive material from the plant during a severe core damage accident (See also Section 3.4). The factors addressed in the definition of the release categories include the response of the containment structure, timing, and mode of containment failure; timing, magnitude, and mix of any releases of radioactive material; thermal energy of release; and key factors affecting deposition and filtration of radionuclides. Release categories can be considered the end states of the Level 2 portion of a PSA.

The calculations of activity of release in Bq are performed for each radionuclide group, and each of the source terms together with frequency of release corresponding to frequency of release categories/plant damage states. Typical results referring to Release Classes, radionuclide groups and frequencies are shown below in Table 9 (See also Section 3.4).

Table 9: Example of table of source terms for particular release classes

Release Classes	Radionuclide groups										RC Frequency
	Xe	CsI	CsOH	Te	Sr-Ba	...					
	Source terms										
RC1	$ST_{RC1\text{-}Xe}$	$ST_{RC1\text{-}I}$	$ST_{RC1\text{-}Cs}$	$ST_{RC1\text{-}Te}$	$ST_{RC1\text{-}Sr}$						f_{RC1} (f_{SGTR})
RC2	$ST_{RC2\text{-}Xe}$	$ST_{RC2\text{-}I}$	$ST_{RC2\text{-}Cs}$								f_{RC2} ($f_{ECF\text{-}T1}$)
RCm	$ST_{RCm\text{-}Xe}$	$ST_{RCm\text{-}I}$	$ST_{RCm\text{-}Cs}$	…							f_{RCm} (f_{NoCF})
Core inventory [Bq]	CI_{Xe}	CI_{I}	CI_{Cs}	CI_{Te}	….						=FDF =sum fRC1:fRCm)

Release classes with their frequencies are extracted from the last row in Table 9 given above. Multiplying total core inventory in Bq of particular radionuclide groups and source terms (i.e. fraction of total core inventory) the final activities of releases for particular release classes with their frequencies and radionuclide groups are obtained. Activities of releases with frequencies corresponding to particular Release Classes are used further for derivation of total risk. According to the definition of risk, using the activities of releases shown in Table 9, total risk can be expressed by the following matrix:

Sequence/ Release Class	**frequency**	**conseq.**	**Risk**
s_1	f_1	c_1	r_1
s_2	f_2	c_2	r_2
s_3	f_3	c_3	r_3
…			
s_n	f_n	c_n	r_n

Total risk $\quad$ ***sum*** $(r_1 : r_n)$ $\quad$ (3.10.2)

3.10.2.4 CRT Assessment

As it follows from the above given matrix, the risk can be defined as:

$$R \leq \Sigma_i f_i \times c_i \tag{3.10.3}$$

where

i is the i^{th} release mode (class, sequence, source term),

f_i is the frequency per year of the i^{th} release mode, and

c_i is the consequence in Bq of **I-131 equivalent** for the i^{th} release mode.

Particular sequences may be grouped according to assessed releases c_i and graded following the INES scale. Thus, we obtain 3 groups of releases/sequences:

- group INES5, where c_{i5} Є {200 through 2000} TBq of I-131 equivalent,
- group INES6, where c_{i6} Є {2001 through 20000} TBq of I-131 equivalent,
- group INES7, where c_{i7} > 20000 TBq of I-131 equivalent.

The parameter should be based also on IAEA requirements of balanced NPP with no significant risk coming from different release classes and no impact to the population and environment, using a graded approach. Hence it can be concluded:

$$c_{i7} \approx 10 \times c_{i6} \approx 100 \times c_{i5} \qquad (3.10.4)$$

Further let us say that we have *j* sequences INES5, *k* sequences INES 6 and *l* sequences INES7, where

$$j + k + l = i \qquad (3.10.5)$$

In order to comply with the IAEA suggestion that risk should be balanced ("there should not be excessive contribution to risk by any release mode"), it is further assumed that the combined release modes included in Level 5 (i,5), Level 6 (i,6), and Level 7 (i,7) of the INES scale should give approximately equal contributions to the total risk, i.e.,

$$\Sigma_j (f_{j,5} \times c_{j,5}) \approx \Sigma_k (f_{k,6} \times c_{k,6}) \approx \Sigma_l (f_{l,7} \times c_{l,7}) \qquad (3.10.6)$$

Moreover, as it comes from expression (3.10.2) the total risk is the sum of all risks. Therefore, it can be concluded for total risk:

$$R = sum (\Sigma_j (f_{j,5} \times c_{j,5}); \Sigma_k (f_{k,6} \times c_{k,6}); \Sigma_l (f_{l,7} \times c_{l,7})) \qquad (3.10.7)$$

For the INES group frequencies - based on the PSA L1 event tree principles - it follows that:

$$\Sigma_j f_{j,5} = f_5 \qquad (3.10.8)$$

$$\Sigma_k f_{k,6} = f_6 \qquad (3.10.9)$$

$$\Sigma_l f_{l,7} = f_7 \qquad (3.10.10)$$

$$f_5 + f_6 + f_7 = FDF \qquad (3.10.11)$$

Based on the above given considerations the total risk can be thus expressed as follows:

$$R = f_5 \times c_5 + f_6 \times c_6 + f_7 \times c_7 \qquad (3.10.12)$$

From where - using the expression (3.10.4):

$$R = f_5 \times c_5 + f_6 \times 10 \times c_5 + f_7 \times 100 \times c_5 \qquad (3.10.13)$$

$$R = c_5 \times (f_5 + 10 \times f_6 + 100 \times f_7) \qquad (3.10.14)$$

And using the expression (3.10.11) with simple rounding for risk in approximation it follows:

$$R = c_5 \times FDF \tag{3.10.15}$$

To be conservative and comply with the 3 IAEA-INSAG objectives [121] with requirement of minimum releases and high confidence limit of FDF it can be then concluded that the final expression for Common Risk Target should be:

$$CRT = C_{INES5low} \times FDF_{max} \tag{3.10.16}$$

where

CRT is the Common Risk Target with no significant contribution to already accepted other industrial risks

$C_{INES5low}$ corresponds to **INES5 lower level of releases** corresponding to 200 TBq I-131 equivalent

FDF_{max} is individual Fuel Damage Frequency maximum of a single unit per reactor year corresponding to a high level confidence safety limit of risk with no significant contribution to already accepted other industrial risks

Substituting INES5 lower level, the expression (3.10.15) for Individual Common Risk Target (ICRT) of single unit on the site is derived:

$$\boxed{ICRT = 200 \times FDF_{max}\ TBq\ of\ I\text{-}131_{\ equivalent}\ ^{*}} \tag{3.10.17}$$

*We should keep in mind that when referring to I-131 equivalent we are addressing ALL radionuclides and instead of I-131 equivalent any radionuclide might be used. Nevertheless, for transparency reasons in terms of preserving the link to INES grades, I-131 equivalent seems to be the most appropriate.

One of the deficiencies in the currently used approach trying to address the problem of the nuclear safety or risk from the operation of NPPs is the fact that currently used limits do not consider higher number of units on a site, and thus the assessment of potential risk of the whole nuclear installation may be underestimated. To comply with the requirement of no significant contribution to the risk from other industrial activities the total risk of the site should be evaluated, because the whole installation contributes to the risk influencing the vicinity or farther areas around the site, and not only a single unit. A good example is again Fukushima disaster, where common cause failure of four units out of six occurred.

Therefore, the total universal common risk target - still keeping in mind the requirement of the constant total risk – is expressed as follows:

$$\boxed{UCRT = \Sigma m\ IRm + \Sigma n\ IRccf\text{-}n} \tag{3.10.18}$$

where

m is number of units in the site,

n is number of possible combinations of common cause failures of the units,

IR is the individual risk of a single unit calculated using CRT method as described above,

IR_{ccf-n} is the risk of one common cause combination calculated for common cause initiators for more units using CRT method as described above.

Thus the UCRT addresses the problems that arose after the Fukushima accident by involving integrated site risk analysis. It involves also the problem of single source initiators that may cause multi-unit accidents due to cross-unit dependencies such as shared support systems, spatial interactions (flood propagation pathways) etc.…, common cause failures, or operator actions. It also offers the solution of the problem of common cause initiators challenging simultaneously units at multi-unit site (earthquakes, external floods, severe weather) [125].

3.10.2.5 Verification of Common Risk Target Adequacy

In Section on definition of the CRT all requirements to be fulfilled by the new CRT parameter are summarized and below the verification of the parameter against the requirements of §3.10.2.1 are performed:

- The parameter is representing risk, which is product of frequency and consequences expressed in Bq, - thus requirements a) are fulfilled.
- It should follow constant risk principle, graded approach and it should represent balanced plant with no extreme contributions of particular sequences to the total risk.

  As it follows from the derivation process, the requirements b) were fulfilled.
- It should comply with multi-unit site requirement.

  The equation (3.10.11) contains expression of risk assessment of a multi-unit plant and it takes into account possible risks stemming also from common causes. Thus, the parameter fulfils the attribute set in item c.
- It should a quantitative parameter.

  The derived parameter CRT has a value expressed in Bq, which is possible to convert to absolute consequences, thus it has its quantitative value and the attribute from item d) is fulfilled.
- Its quantitative value should be comparable with risks stemming from other industrial activities.

  As it follows from equations (3.10.4), (3.10.10) and (3.10.11) for CRT, the parameter is expressed in Bq, and using Table 4 converting Bq to absolute consequences, it is possible to compare it with risks stemming from other industrial activities. Thus, attribute form e) is fulfilled.
- It should take into account all radiological sources in the site.

  The frequency part of the CRT parameter represents FDF, not only CDF, as it is currently expressed. Thus the parameter fulfils the f) attribute.
- It should be objective as a contrary to subjective.

  The CRT risk parameter is derived pursuant to generally used L2 PSA practices using containment matrix and common mathematical procedures, using also IAEA INES scale as well as the results of validated codes MELCOR and MACCS2 and commonly accepted definitions for safety, risk, Farmer curve etc. From what was written above it follows, that the CRT was derived based on objectively accepted principles, definitions and procedures, and so it is not based on subjective judgment. Thus it fulfils the g) attribute.
- It should comply with harmonization requirements.

  The presented CRT parameter is based on generally and commonly valid and accepted principles of IAEA, that is the subject authorized to establish internationally respected principles, requirements and recommendations, which should be implemented into legislation of particular countries. Thus it fulfils also the requirement h).
- It should be the maximum admissible risk value for a safe plant.

CRT reflects safety, since it is based on minimization of releases during a severe accident corresponding to lower level of INES5. Releases are related to the source of radioactivity, which is the core, the fuel in the fuel pools as well as fuel casks. Thus, the CRT parameter corresponds to IAEA safety definition related to the source of radioactivity minimizing releases, from which it follows, that it fulfils the i) attribute.

- It should guarantee adequate scientific level and credibility by being technically derived not only designated. (And for as much possible, also accepted by wider scientific public.)

Credibility of the method is guaranteed by publishing it in prestigious scientific journal Nuclear Engineering and Design [43] as well as it was a subject of thesis [114]. Thus, the CRT parameter fulfils the last requirement j).

More details about the CRT methodology is provided in Reference [43] and [114].

The CRT methodology helps to better analyze the L2 PSA results, to judge if a plant is balanced at level of particular INES grades contributing to total risk, or at level of particular sequences belonging to particular INES grades. This may be used also for decision making as presented in [127]. The advantage of the method is, that makes link between real and hypothetical accidents through the INES scale and makes possible to compare nuclear risks between each other as well as with other industrial risks lowering the need of L3 PSA performance.

3.10.2.6 Limitations

There are limitations which are intrinsic to the INES scale, for example, it only takes into consideration the atmospheric release (the liquid release and ground contamination are not taken into account),

3.10.3 Discussion

The “total risk” integrates the risk due to all event sequences into a single metric, and it even can be a measure for the integral of all off-site consequences due to all possible sequences. This is a very appealing concept which provides risk of releases as an indicator of nuclear safety linked to radiation source only, enables a comprehensive presentation and analysis of results, and it supports decision making.
Of course one numerical value – as actually in case of other risk metrics being already in use - cannot capture the majority of information which is available within a L2 PSA. Therefore, the integral risk must never be the only L2 PSA result. Once the total risk is established, it enables, for example, easy identification of individual contribution of L2 PSA elements like release categories, accident phenomena or core damage state (CDS) to the total risk.

3.10.4 ASAMPSA_E recommendation on total risk measure

According to most guidelines (e.g. [5]), L2 PSA should provide a set of source terms together with their frequencies. If a L2 PSA complies with this requirement, it can provide a total or integral risk measure as a complement to the many other risk measures under consideration. This can be done by integrating the risk due to all event sequences into a single metric by summing up all activity releases multiplied by their respective frequencies. Technically, this is an easy task for L2 PSA which has all accident sequences and release categories with their respective source terms available.

An attractive feature which comes with a single value for the integral risk is the possibility to compare it to a risk target. Without such a single value, having just a set of several L2 PSA result characteristics, it is difficult to define a consistent set of various targets for the different result characteristics.

However, the total risk measure CRT, as actually also other currently used risk metrics, combines sequences with very different degrees of uncertainty, and therefore safety insights and reliability of each of them have to be analyzed or approved with safety regulators in each country.

This approach is an interesting contribution for further harmonization of L2 PSA practices.

The CRT method was already applied as an example for SAM applications (see in ASAMPSA_E) as well as for processing of Full Scope L2 PSA results for Mühleberg NPP in Switzerland in 2013 in case of considered backfitting reflecting the Fukushima accident [124]. For decision making, an example using CRT method is given within ASAMPSA_E [127].

4 MULTI-SOURCE PSA AND SITE LEVEL RISK METRICS

In this section, we discuss the extension of PSA Level 1 and Level 2 risk metrics and risk measures to multi-source and site level risk metrics and measures. The starting point of the discussion is the observation that multi-unit (multi-source) accident sequences may be caused by two classes of initiating events:

- Site Common-Cause Initiators (SCCIs): Initiators that simultaneously challenge all of the units at the site. SCCIs include initiators that are caused by external hazards (e.g. earthquakes, severe weather).
- Single-Unit Initiators (SUIs): Initiators that occur at one unit. SUIs generally include initiators caused by internal hazards such as internal events (e.g. loss of main feedwater, loss of coolant accidents), internal floods, and internal fires. SUIs may cause multi-unit accidents due to cross-unit dependencies such as shared support systems, spatial interactions (e.g., internal flood and internal fire propagation pathways), common cause failures or operator actions.

As shown in the figure below [86], this concept, which has been defined in the context of a single unit PRA, needs to be refined to resolve the extent of impact on a multi-unit site. A comparison of the initiating event treatments in multi-reactor vs. single reactor PRAs is provided in

Table 10. It should be noted that site level risk measures are abbreviated by adding a leading 'S' to commonly used risk measures like CDF or LERF.

The crucial observation is that such extensions of direct[15] unit-level risk measures to site level risk measures can be defined in a straight forward manner for the commonly used risk metrics of PSA. This is justified by the following arguments:

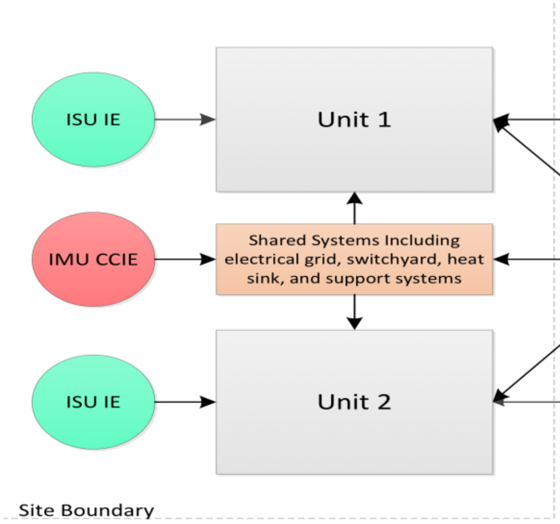
- a direct risk metric references a specific state or condition of the plant and is applicable to a sequence $s_i \xrightarrow{\varphi(l_{ij})} c_j$ assigned to the respective consequence.
- the site is – formally speaking – an integer set of radionuclide sources r_K, for which an addition over two elements is well defined.
- the risk metric (consequence) is either applicable to the radionuclide source or not. If the risk metric is not applicable, its contribution to the risk measure is identically zero, otherwise the respective distribution for the sequence is its likelihood distribution φ. Distribution aggregation constraints apply.
- consequently, the risk metric can be "summed up" over the set of radionuclide sources $\{r_K\}$ in a well-defined manner[16].

There is one important distinction between direct risk measures with respect to extending them to a site-level risk measure:

1. The risk metric is based on a (binary) condition of the respective sources, which is either fulfilled or not fulfilled. Then, the extension to a site-level metric is trivial. The risk aggregation needs to respect Boolean logic to prevent double-counting of the simultaneous occurrence of the consequence at more than one source (see also below). The salient example for this kind of risk measure is CDF/FDF.
2. The risk metric is derived by partitioning a "continuous" property into a limited number of classes by threshold values. The formal extension of the risk metric is straight forward. However, there are likely contributions to a certain class by the simultaneous occurrence of sequences which individually do not meet the criteria for the class. The salient example for this kind of risk metric is a release metric like LRF. This will require some care in building the site-level risk model.

In appendix A, a more formal discussion is provided. We continue our discussion with a brief example.

[15] For the definition of direct and secondary risk metrics, see appendix A.
[16] There are some objections that the based metrics are not just a linear combination of reactor based metrics. It is therefore important to define a consistent aggregation approach over the sources.



Nomenclature :

ISU IE – Internal Single Unit Initiating Event

ESU IE - External Single Unit Initiating Event

IMU SCCIE – Internal Multi Unit Site Common Cause Initiating Event

EMU SCCIE – External Multi Unit Site Common Cause Initiating Event

Figure 4 : Initiating Event Categories for Multi-Unit PRA [86]

Table 10: Comparison of Initiating Event Treatment in Single and Multi-Unit PRAs

Characteristic	Single Unit PRA	Multi-Unit PRA
Completeness Objective	• Capture all events that make significant contributions to single unit risk metrics: CDF, LERF, RCF, CCDF	• Capture all events that make significant contributions to site risk metrics: SCDF, SLERF, CPMA, SRCF, SCCDF
Internal Event Categories	• Events with unique impact on reactor safety functions ○ Transients ○ LOCAs ○ SG/HX faults • Loss of offsite power • Support system transients	• Single unit internal events impacting each reactor unit separately • Loss of offsite power events impacting each combination of two or more reactor units • Faults in shared systems impacting each combination of two or more reactor units
Internal Plant Hazards	• Flood induced initiating events • Fire induced initiating events • Other hazard induced initiating events	• Single unit internal hazard events impacting each reactor unit separately • Internal hazard events in common areas impacting each combination of two or more reactor units
External Plant Hazards	• Seismically Induced initiating events • Tsumani/external flooding induced initiating events • High wind hazard induced initiating events • Other external hazard induced initiating events	• Single unit external hazard events impacting each reactor unit separately • External hazard events impacting each combination of two or more reactor units
Initiating Event Frequency Basis	• Events per reactor calendar year	• Events per site calendar year

In order to understand the development of the total site risk estimate, let's consider a three-unit site with units labelled Unit 1, Unit 2, and Unit 3. There are seven possible outcomes that involve release from one or more units, as listed below:

- single-unit outcomes: Unit 1, Unit 2, Unit 3
- dual-unit outcomes: Unit 1 and Unit 2, Unit 1 and Unit 3, Unit 2 and Unit 3
- triple-unit outcomes: Units 1 and Unit 2 and Unit 3

Specifically, there are three single-unit outcomes, three dual-unit outcomes, and one triple-unit outcome. The various outcomes can be depicted on a diagram, as shown in Figure 5, where all of the outcomes that affect a specific unit are included within a circle [87].

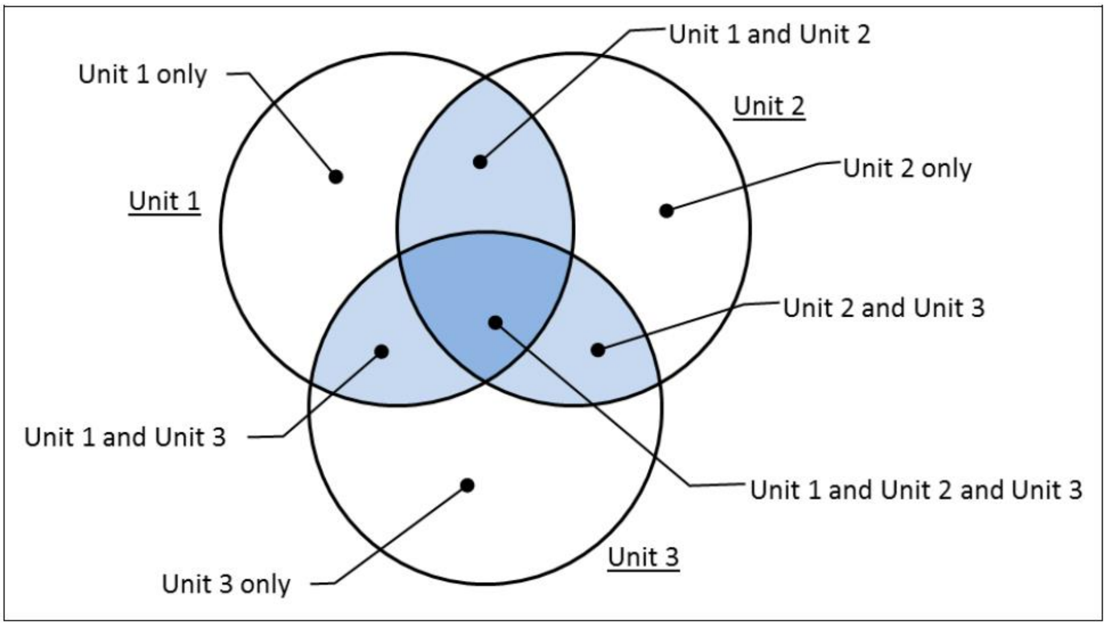


Figure 5 : Diagram Depicting Multi-unit Accidents [87]

As an illustrative example let's consider the occurrence of a SCCI at a three-unit site with units labelled Unit 1, Unit 2, and Unit 3, and define the following events, as in Figure 5:

1. Event U1 = release from Unit1
2. Event U2 = release from Unit2
3. Event U3 = release from Unit3

From these fundamental definitions, we can have the following seven compound events:

1. P($U1/SCCI$) = probability of release from only Unit1
2. P($U2/SCCI$) = probability of release from only Unit2
3. P($U3/SCCI$) = probability of release from only Unit3
4. P(($U1$ and $U2$)/$SCCI$) = probability of release from Unit1 and Unit2
5. P(($U2$ and $U3$)/$SCCI$) = probability of release from Unit2 and Unit3
6. P(($U1$ and $U3$)/$SCCI$) = probability of release from Unit1 and Unit3
7. P(($U1$ and $U2$ and $U3$)/$SCCI$) = probability of release from Unit1 and Unit2 and Unit3

The events being defined as disjunctive, the total probability of having a release from the site as a consequence of a SCCI is the sum of all these terms. Note that no limits on the quantity of accidental releases have been set for this example.

In general, for a site that has n units the number of outcomes that involve exactly k out of n units is $\binom{n}{k} = \frac{n!}{k!(n-k)!}$. For a site with n units, $2^n - 1$ disjunctive events have to be considered if all possible combinations need to be explicitly determined for calculating a site risk measure. For sites with more than 3 units this will be an incentive for more conservative and bounding approaches instead of a rigorous investigation of all potential interdependencies. Conceptually, this issue is particularly relevant for Small Modular Reactors (SMR).

Regarding the relationship of unit-level and site-level risk measures, the following observations can be made.

- Risk measures for the risk of a unit

  These are the "traditional" risk measures, indexed here for differentiation, like FDF_{unit}, CDF_{unit}, or LRF_{unit}.

- Risk measures for the risk of a site

  This comprises risk measures specific to the site. Based on the discussion above, several important unit-specific risk measures can be extended in a straight forward approach as site risk measures, i.e. FDF_{site} or LRF_{site}.

- In Appendix A, we provide a theoretic discussion of risk aggregation over direct risk measures for a multi-unit model. Based on this discussion, we remark that the following relationship is true for binary type direct risk measures like CDF/FDF.

$$FDF_{\text{unit}\_i} \leq FDF_{\text{site}} \leq \sum_i FDF_{\text{unit}\_i}$$

  This rests on the assumption that unit-specific PSA for deriving unit-specific risk consider the interdependencies with other units (as conditional probabilities) in their models. As explained above, the site-specific risk for a certain event (and three units) can be decomposed as follows.

$$FDF_{\text{site, event}} = f_{\text{event}} \cdot (CFDP_{1\&2\&3} + CFDP_{1\&2\&\neg 3} + \cdots + CFDP_{1\,\&\,\neg 2\,\&\,\neg 3} + \cdots)$$

  and

$$FDF_{\text{unit}\_1,\text{ event}} = f_{\text{event}} \cdot (CFDP_{1\&2\&3} + CFDP_{1\&2\&\neg 3} + CFDP_{1\,\&\,\neg 2\,\&\,\neg 3})$$

  The assertion follows by comparing the different sums.

- However, unit-specific PSA are usually determining unit-specific risk for an event as e.g.

$$FDF_{\text{unit, event}} = f_{\text{event}} \cdot (CFDP_{\text{unit}}).$$

  Therefore, site risk measures are often hard to determine accurately based on unit-specific PSAs. The PSA models, particularly the event tree/fault tree modelling of unit-specific PSAs at multi-unit sites needs to be adapted to allow for evaluations for the multi-unit part of the risk, unless the conservative bound mentioned above is used. If the site risk profile needs to be determined in an accurate and consistent manner this may lead to a restructuring of the PSA models in a unit-specific part and into (several) multi-unit parts.

- For non-binary type risk measures, e.g. for LRF as a salient example of a risk metric defined over a continuous variable like the amount of release of Cs-137 in Bq, the issues are more convoluted. The following observations can be made.

  For determining the LRF_{site} it is not sufficient to consider only scenarios for which the release from one unit reaches the large release threshold. In addition, potential scenarios affecting multiple units, for which the total release only jointly reaches the large release threshold, have to be considered. This will impose constraints on the eventual development of the PSA model. In fact, this poses one of the major problems in integrating several single-unit PSA models into a comprehensive and consistent multi-unit PSA model. Importantly, LRF_{site} is not necessarily bounded by the sum of $LRF_{unit,i}$.

  This issue is particularly relevant for all release-type risk measures.

We can conclude that the extension of unit-specific risk measures to site-specific risk measures can be done based on established risk metrics like CDF/FDF or LRF.

We continue to investigate the extension of the PSA model to a multi-source PSA. We can note the following:

- Conceptually, there are few differences between a multi-unit and a multi-source model. In fact, every multi-unit model is a multi-source model. Thus, the remarks on extending unit-specific to site-specific risk measures are also applicable to the multi-source case.
- In addition to fuel in the reactor, the spent fuel in the fuel pool or during transport on the site is adequately treated by the established Level 1 and Level 2 risk measures like FDF or LRF.
- Established Level 2 risk measures are in principle fully applicable for all types of relevant sources at a NPP or research reactor. There might be the need to adapt the leading isotopes for a Thorium-cycle reactor or a transmutation type reactor, but this does not change the overall approach. For practical purposes and for current NPP and research reactor sites, the established risk measures are sufficient.

With respect to scenarios potentially leading to multiple releases, it must be emphasized that this fact has to be considered by PSA analysts when making and justifying claims in bounding assessment. If there is a relevant possibility that several potential sources (core, SFP, etc.) can reach an accidental state in the same scenario, the claims on FDF, LRF, and ERF have to reflect the following:

- the analysed event challenges the safety functions for the core and the SFP (and further sources). If a simultaneous challenge does not reduce the reliability of the safety functions, then an accident state is reached if the event is not controlled for one source. For example with respect to the FDF measures, the (point value) claim on FDF should be estimated as

  $$\mathrm{FDF}_{\mathrm{event}} = \mathrm{f}_{\mathrm{event}} \cdot (\mathrm{CFDP}_{\mathrm{core}} \vee \mathrm{CFDP}_{\mathrm{SFP}} \vee \dots) \approx \mathrm{f}_{\mathrm{event}} \cdot (\mathrm{CFDP}_{\mathrm{core}} + \mathrm{CFDP}_{\mathrm{SFP}} + \cdots)$$

  CFDP is the Conditional Fuel Damage Probability.

  Especially for LRF and other risk metrics defined over continuous variables, the situation can be more complicated, if there are important scenarios that do not reach the large release threshold by themselves, but exceed that threshold if releases from several sources occur. This possibility needs to be checked. A theoretical discussion can be found in [84]. Moreover, the observations on the aggregation of risk from different sources discussed previously are also applicable.
- PSA analysts should check if there are any indications that the reliability of safety functions could be reduced. Possible indications include:
  - safety functions rely on the same (trains of) safety systems,
  - the same (trains of) supply or support systems have to be operable,
  - controlling the event depends on using the same, limited resources,
  - challenges to barriers or the integrity of the containment are elevated by (simultaneous) accidental states in multiple locations,
  - operator or crisis centre actions or decisions are less reliable (or no longer feasible) due to accidental scenarios in multiple locations.

In these cases, it is recommended making a direct claim on the respective conditional probabilities. For the example of FDF, the claim should be made as

$$\mathrm{FDF}_{\mathrm{event}} = \mathrm{f}_{\mathrm{event}} \cdot \mathrm{CFDP}_{\mathrm{event}}.$$

For screening purposes, it should be checked that bounding assessment claims on conditional probabilities for the event are larger than the sum of claims, which would have been made on single sources, i.e. e.g. $\mathrm{CFDP}_{\mathrm{event}} \geq \mathrm{CFDP}_{\mathrm{core}} + \mathrm{CFDP}_{\mathrm{SFP}} + \cdots$

- If decision makers, important stakeholders or PSA analysts want to cover the issue of multiple sources in screening more explicitly, there are several possibilities.

  Scenarios, which potentially affect multiple locations could be retained at lower screening thresholds for the respective risk measures (a factor of 10 would seem sensible). Screening could be performed against risk measures specifically tailored for multiple locations, i.e. by using risk measures conditional on accident scenarios for multiple sources. However, this is may not be necessary for screening. It might, however, be necessary to structure the detailed PSA analysis for the unit in such a way.

Based on this discussion, we arrive at the following conclusion with regard to site-level risk measures. Direct risk measures for PSA Level 1 and Level 2 can be extended to a site-level risk measure in a straight forward and well-defined way. Given the definition of the unit-specific risk metric, the respective site-level metric is also defined (and vice-versa).

Furthermore, we point out the following:

- The construction of a site-level risk model (i.e. multi-unit and multi-source) requires care. Commonly used SSC, other provisions as well as resources, including human resources, have to be identified and included adequately in the risk model. This, however, is not directly related to the site-level risk measures. Importantly, binary type risk measures make for an easier building of the site-level risk model.
- The selection of the appropriate risk measures for a site-level model depends on a lot of factors, which specifically include the objectives of the PSA and the resulting scope of the model.
- With regard to secondary risk measures, these depend only on the underlying direct site-level risk measure. Therefore, no fundamentally different approaches need to be defined. One obvious addition the derived risk measures discussed above is the contribution of a certain source to the site-level risk. This is conceptually the same as computing the results for a group of e.g. components for a single-source model. Thus, e.g. an importance measure is also naturally extended to the site-level.

5 RECOMMENDATIONS ON RISK METRICS FOR AN EXTENDED PSA

Many risk measures have been discussed in the previous sections with the aim of being complete and well founded. However, in practice there is no lack in availability of risk metrics, but in the harmonized selection of such metrics. Therefore, to be practical and in order to contribute to harmonization of PSA application, just four risk metrics are recommended in the present section: Two for PSA level 1 and level 2 each.

In general, L1 PSA risk metrics assess the risk within a plant, whereas L2 PSA risk metrics are related to risks of releases, which reflect the requirement of fundamental safety objective - to protect people and the environment from harmful effects of ionizing radiation applying to all circumstances that give rise to radiation risks.

5.1 Risk Metrics for an extended Level 1 PSA

The Level 1 risk metric has to be defined as those end states of the PSA Level 1 model that are classified as accidental. In that sense, the risk metric aggregates over the plant damage state metric(s), which are assigned to the accidental end-states of the PSA Level 1.

From the review of widely used risk measures, FDF measure, defined as a loss of integrity of fuel elements on the site, which has the potential for an accident-level release, provides a more general notion of a PSA Level 1 end state than other direct risk measures as CDF. CDF that should be understood as a fuel damage state affecting fuel elements located in the reactor core, is considered as a subset of FDF. Similarly, risk measures related to other locations than the core as SFPDF are also subset of the FDF risk measure. FDF is a direct risk measure that encompasses all these secondary risk measures. Moreover, the FDF measure needs to be consistent with the plant damage state measure(s) (PDSF) it shall aggregate.

FDF risk measure has the following limitations. It does not distinguish between severity of core damage (extent of damage to fuel rods) beyond the defining threshold for fuel damage and it does not preserve (or provide) information on fuel damage characteristics in light of expected releases (e.g. time of fuel damage onset, extent of fuel damage, status of barriers and safety systems, etc.).

Because the main risk measures for PSA Level 1 like e.g. core damage frequency or fuel damage frequency are not well suited for describing several scenarios which might lead to a significant release of radionuclides into the plant as a starting point for a PSA Level 2, a new metric, “Radionuclide Mobilization Frequency, RMF” (Section 2.17), addresses these issues. This risk metric is defined as a loss of the design basis confinement for a source of radionuclides, leading to an unintended mobilization of a significant amount of radionuclides with the potential for internal or external release, e.g. more than 1 TBq I-131 Equivalent[17]. The threshold value and its reference radionuclide (or radionuclides) have to be adjusted to the facility under consideration and the objectives of the

[17] The proposed threshold value has been set to 1 % of the lower end 100 TBq I-131 Equivalent limit for an accidental level release (INES 5) defined in the INES manual [108]. This assumes that short-term consequences are of interest. For long-term consequences, a threshold reflecting e.g. Cs-137 should be selected. .

study. The RMF conceptually aggregates rather diverse sequences in terms of consequences into one common risk measures (figure of merit). While this is one of its advantages, it similarly limits its suitability for understanding the actual risk profile with regard to the fundamental safety objective.

The RMF was proposed during the ASAMPSA_E project. The RMF risk measure is recommended to be used for an extension and generalization of the established CDF and FDF risk measures to a multi-source PSA (cf. section 4). It is a complementary risk measure for an extended PSA that addresses potential sources on the site in addition to fuel in the reactor and spent fuel. Currently, no applications of RMF are known, and there is no consensus on the threshold value and its reference isotopes. However, the RMF generalizes the CDF and FDF risk measures to a comprehensive PSA Level 1 risk measure for a multi-source PSA. This risk measure can also contribute to the verification of the low probability of events that would induce off-site protective measure without core melt.

It must be pointed out, though, that the RMF risk measure is not well suited for understanding the risk profile of e.g. an NPP in operation. It should be complemented by e.g. CFD/FDF as a PSA Level 1 risk measure. FDF would be the recommended metric in this case.

5.2 Risk Metrics for an extended Level 2 PSA

The above sections on possible risk metrics for level 2 PSA provide a comprehensive summary on this topic. Although existing PSA at maximum only partly apply the many options for different risk metrics, there is a large choice of metrics available. This wide selection of risk metrics is also applicable for extended PSA.

Nevertheless, the following remarks are due:

It is of interest to have not only a single value presenting the total risk (whatever that may be) from the set of units on the site, but to be able to determine the contribution of initiating events (e.g. external hazards) and different plant operation states and particular SSCs. This requirement is not at all specific for extended PSA; it is comparable to providing the risk contributions from different issues in traditional PSA.

The risk metrics applied in an extended PSA for a multi-unit site should be identical with the risk metrics provided for individual units. The risk of each individual unit at a particular site should be given, and also the cumulative risk for all units on a site. Of course one could imagine complicated risk patterns from multi-unit sites. The accidents in Fukushima Dai-ichi are a striking example for different accident evolutions initiated by the same external hazard in different reactor blocks on the same site. But again, this does not necessarily call for additional or modified risk metrics. In principle, the different release histories from different reactor blocks are comparable to a sequence of release episodes from a single reactor. It has to be conceded that calculating these risks from multi-unit sites is really challenging, but there is no reason for introducing additional risk metrics or dismissing other metrics which have been proposed for single unit PSA.

From the various metrics discussed above, the following are recommended as particularly suited for characterizing PSA level 2 results.

5.2.1 Measure for loss of containment function

There is already a widespread good practice in L2 PSA to identify the frequency of the loss of containment functions. The application of this measure is further encouraged, with the following comment:
It is recommended to at least distinguish for core melt sequences:

- Intact containment with design basis leakage,
- Intact containment with filtered venting,
- Loss of containment function due to a leak or rupture of the containment structure,
- Loss of containment function due to failure of containment systems (e.g. open ventilation systems, open hatches),
- Loss of containment function due to bypass through interfacing systems (for BWR including non-isolated break of feedwater or steam lines outside of the containment),
- Loss of containment function due to bypass through steam generator tube leak (PWR only).

It may be interesting to introduce an additional metric, which has similarity to the well-known core damage frequency (CDF) concept of L1 PSA, “Containment Failure Frequency” (CFF). The CFF would comprise all sequences where the containment function is lost – whatever the reason (See Section 3.5.3).

5.2.2 PSA Level 2 total risk measure

Depending on judgments involving also non-scientific considerations, the “total risk” of any installation can be defined in very different ways, e.g. in loss of value (of the plant and for the environment), or in health effects - which are far from being a precise category (e.g. distinguish long-term health effects from short-term health effects). The present document is about PSA level 2, and therefore the “total risk” which is proposed here is related to PSA level 2 issues.
The total risk measure should be seen as an optional complement to the many other risk measures under consideration. This can be done by integrating the risk due to all event sequences into a single metric by summing up all activity releases multiplied by their respective frequencies. Technically, this is could be an easy task if L2 PSAs have all accident sequences and release categories with their respective source terms available. When documenting the PSA, the contributions of interest to the total risk measure (e.g. specific initiating events, failure of particular SSCs, and potential of SAMs for reducing the total risk) should be indicated. Based on this information, it is possible to assess whether the design is well balanced, or whether particular improvements should be considered.
The parameter Common Risk Target (presented in Section 3.10.2) is an example application of “total risk”. It is the risk of releases expressed in Bq which is possible to convert to absolute consequences to be comparable to other industrial risks.

Note that, as the sum will add a lot of assumptions with different level of conservatism, it may be hard to get conclusive insights from this metric. Considering that, it is presented here as an optional metric to be discussed with the country's safety regulator.

Nevertheless, the attractive feature which comes with a single value for the integral risk is the possibility to compare it to a risk target. It allows for “rational” decision making and for the identification of an “optimal decision” in principle. Without such a single value, having just a set of several different L2 PSA result characteristics, it is difficult to define a consistent set of various targets for the different result characteristics. Unfortunately, the PSA community is far from having consensus on what might be the proper harmonized risk measure. It is recommended that pertinent working groups precisely define the appropriate metrics (e.g. the isotopes to be considered, or the introduction of a parameter representing health effects for the individual isotopes). Once such a metric is defined and accepted by decision makers it can be completed by pertinent risk targets.

6 CONCLUSIONS

This report provides a review of the main used risk measures for Level 1 and Level 2 PSA. It depicts their advantages, limitations and disadvantages and develops some more precise risk measures relevant for extended PSAs and helpful for decision-making. This report does not recommend or suggest any quantitative value for the risk measures; this is part of other ASAMPSA_E deliverables. It does not also discuss decision-making in detail.

The risk measures investigated in this report are related to the Level 1 and Level 2 PSA for NPP and the properties and characteristics of risk actually included into these models. Level 3 PSA risk measures and risk metrics are not discussed in this report but Level 2+ risk measures are partly covered. Level 2+ PSA is understood as a Level 2 PSA with a simple model extension for releases to the environment of the plant (Level 3 PSA).

The choice of one appropriate risk measure or a set of risk measures depends on the decision making approach as well as on the issue to be decided. The general approach for decision making, as discussed in Section 1.5, aims at a multi-attribute decision making approach. This can include the use of several risk measures as appropriate.

There is a general trend for the safety goals to be developed in a hierarchical structure, from qualitative high levels, rooted in the legislation, to more quantitative, surrogate safety goals (e.g., CDF, LRF) at the bottom levels. The demonstration of relationship between the surrogate metrics (e.g., CDF, LRF) to upper level objectives remains a subject that needs to be clearly defined.

The report shows that many of currently used risk metrics provide useful information if several of them are used in combination appropriately.

A throughout review of each currently used risk metric is provided in Section 2 for Level 1 PSA and in Section 3 for Level 2. Not all of them are widely used or agreed on.

Section 5 provides recommendation on risk metrics to be used for an extended PSA. For Level 1 PSA, Fuel Damage Frequency and Radionuclide Mobilization Frequency are recommended. For Level 2 PSA, the characterization of loss of containment function and a total risk measure based on the aggregated activity releases of all sequences rated by their frequencies are proposed.

It must be noted that there is still a lot of work is ahead for the definition of site integrated risk metrics for extended PSA. There are many ongoing international efforts to address some of the technical challenges as (e.g. IAEA, USNRC…):

- identification and definition of integrated site risk metrics that measure multi-source effects.
- identification and definition of comparable surrogate risk metrics for spent nuclear fuel sources.
- quantification of multi-source accident sequences involving success paths for one or more sources.

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8 APPENDIX A: RISK DEFINITION

8.1 Glossary

Scenario: s_i

Consequences: c_j

distributions: φ

likelihood: l_{ij}

8.2 Introduction

There is considerable discussion in the scientific community (or rather communities) on the appropriate definition of risk. In this regard, see e.g. a series of papers by T. Aven [18], [19], [20], [21], [22], or the discussion in [52]. The ASAMPSA_E project does not aim at resolving this discussion.

Conceptually, risk is understood as an uncertain event with adverse consequences [19], [32], [35]. This is usually not applicable to past events, for which exists certainty on their occurrence and (most of) their consequences. Therefore, the term risk will be applied with respect to future events within this report. The risk of these future events is assessed with a risk model.

In light of the definitions and requirements on current PSA (cf. e.g. IAEA SSG-3 [4] and SSG-4 [5]) and the classification scheme by Aven [20], the approach to risk is the following.

> Risk is understood in terms of a set of (representative or bounding categories of) scenarios (s_i), a set of (representative or bounding categories of) consequences dependent on failure or success of the safety measures (c_j), and the combined likelihood together with the associated uncertainty (distributions φ) for the scenarios and the occurrence of the consequences ($\varphi(l_{ij})$). $\{s_i, c_j, \varphi(l_{ij})\}$

It should be noted that this definition basically rephrases the classic “Level 2” definition given by Kaplan and Garrick [35], but does not follow their definition of the terms probability and frequency. The use of the term “likelihood” entails the concepts of “probability” (a σ-additive measure on a set of events into the dimensionless interval [0, 1]) and “frequency” (the number of event repetitions within a class of events, here understood to be over a period of time with SI dimension s^{-1}). For small non-negative frequency values these can be interpreted as the probability for observing the event within the given time period. With this definition, usually frequency distributions are assigned to scenarios whereas the transition to consequences is described by (conditional) probability distributions. The inclusion of uncertainty into the likelihood approach is a foundational aspect of quantitative risk analysis[18]. Moreover, the fact that both scenarios as well as consequences are usually categories representing an interval of similar events/outcomes is explicitly mentioned.

[18] The classification by Aven distinguishes between probability and uncertainty [20]. The current state-of-the-art for PSAs uses probabilities or their uncertainty distributions depending on the issue, without significant problems.

Distinctions as to whether the (measured) risk is a true or real world property or an attributive property of a model based on an analyst's understanding [18], [32] are not investigated in this report as well. For this report, PSA models and their consequences are assumed as given; the issue of their interpretation is relegated to the decision maker. Still, the uncertainty and sensiti of PSA models and their results to analysts' modelling choices and the impact on risk metrics is an issue for this report as well.

It should be noted that risk models are often used to assess the impact that past events have on their results. This is not – conceptually – a risk assessment. However, it is a legitimate use of risk models which often provides valuable insights. Therefore, the use of risk models for assessing past events and the appropriate metrics for that purpose have been explicitly included in the scope of this report.

8.3 Risk Metrics and Risk measures

For the purpose of this report, the risk metric should not be separated from its quantification procedure, although the latter belongs to the definition of the risk measure[19]. The quantification procedure defines a mapping μ from the aspects of risk, i.e. characteristics related to $\{s_i, c_j, \varphi(l_{ij})\}$, into a measurable quantity (usually $\mathbb{R}$ or $\mathbb{Z}$)[20]. It is important to mention that for the purpose of this project, no specific restriction can be established that valid risk measures for PSA are coherent [60] or convex [61]. In fact, risk measures do not even need to be (sub-) additive or exhibit positive homogeneity.

Conceptually, risk metrics/risk measures in risk models need to be suited for the assessment of future events. However, risk models and thus risk metrics are often used for the assessment of past events. In these cases, the risk metrics turn into so called "safety performance indicators" [32]. In this report, this use of risk metrics will be explicitly denoted.

It is important to note that risk measures (i.e. the quantification results for the risk metrics) can be time dependent.

The risk metrics/risk measures are further distinguished into direct risk metrics and secondary (derivative) risk metrics.

8.3.1 Direct Risk Measures and risk Metrics

Direct risk metrics are understood as those risk metrics (as basis of a risk measures) which can be expressed as or are defining attributes of consequences c_j in a suitable risk triplet $(s_i, c_j, \varphi(l_{ij}))$. These risk metrics can be quantified with a suitable likelihood function $\varphi(l_{ij})$ or a rescaled version $\mu(\varphi(l_{ij}))$ thereof, where μ is at least bijective and monotonous.

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[19] Theoretically, a risk metric can be quantified with different measures. In order to prevent potential ambiguities, this report will usually reference risk measures and only refer to risk metrics, if specifically the metric aspect is addressed. Potential, alternative quantification schemes for risk metrics will be not investigated.

[20] Note that this does include even qualitative (categorial) measures, which can be mapped to a subset of $\mathbb{Z}$.

Obviously, one trivial risk measure μ for the doublet (s_i, c_j) is $\mu(s_i,c_j) = \varphi(l_{ij})$. We identify this risk measure as the natural or standard risk measure for the risk metric of (s_i, c_j). Thus, risk measures are defined as distributions (cf. also [35]). Important properties of these distributions are percentiles (especially the 5%, the median, and the 95% percentile), the mean, and the higher moments of the distribution including the standard deviation. For a lot of practical purposes, instead of distributions either point values or mean values are used and the fundamental aspect of a risk measure as a distribution is (consciously) neglected.

For the consequence c_j, the risk measure is defined as

$$\mu(c_i) = \mu\left(\bigcup_i (s_i, c_j)\right).$$

Already at this point it should be noted that for a set of scenarios s_i, the risk measure for the consequence c_j risk metric should at least be monotonous, i.e. $\mu(s_1,c_j) \prec \mu\left((s_1,c_j) \cup (s_2,c_j)\right) \prec \cdots \prec \mu\left((s_1,c_j) \cup \ldots \cup \mu(s_i,c_j)\right)$ (given a suitable definition of the "≺" relation). Moreover, in a number of important cases, the risk measure is additive[21],, i.e. $\mu(c_j) = \mu(s_1,c_j) + \cdots + \mu(s_i,c_j)$, or at least convex. Specifically, the risk measure should be additive if the scenarios are mutually exclusive, i.e.:

$$\bigcup_i (s_i,c_j) = \{(s_1,c_j),\ldots,(s_i,c_j)\}.$$

If the standard risk measure is used for (s_i, c_j), i.e. $\mu(s_i,c_j) = \varphi(l_{ij})$, and if the scenarios are mutually exclusive, then the convolution $\mu(c_j) = \varphi(l_{1j}) * \ldots * \varphi(l_{ij})$ is a natural risk measure for the consequence c_j i.e. the distribution for the aggregated sequences can be derived as the convolution of the individual distributions. Importantly, the mean values are then additive, i.e. $E\left(\mu(c_j)\right) = E\left(\varphi(l_{1j})\right) + \cdots + E\left(\varphi(l_{ij})\right)$. Distributions and means are usually calculated in uncertainty analysis by applying Monte Carlo methods.

Salient examples for direct risk metrics include "core damage" with the quantification (risk measure) as "core damage frequency" and "large early release" with "large early release frequency". Similarly, the unavailability of a safety system can be expressed as a (preferably unconditional) "system unavailability probability" or a "system unavailability frequency" depending on the specific analysis performed.

Some direct risk metrics (i.e. "time to system failure") are constructed by effectively rescaling the "likelihood" function. For the given example, instead of looking at the frequency of system failure, its inverse value is used. This illustrates that there is significant freedom in the choice of a measure function μ.

The question of how risk measures from different consequences are aggregated is less clear. Even if the consequences are disjoint ($(s_i,c_1) \cap (s_i,c_j) = \emptyset$), and therefore

$$\bigcup_j (s_i,c_j) = \{(s_i,c_1),\ldots,(s_i,c_j)\},$$

simply adding up the risk measures of the consequences in the sense of

$$E\left(\mu\left(\bigcup_j (s_i,c_j)\right)\right) = E(\mu(s_i,c_1)) + \cdots + E\left(\mu(s_i,c_j)\right)$$

might not lead to an acceptable risk measure, although it is certainly well-defined. One such example are Level 3 PSA risk metrics for accidental exposure for a person near the site in mSv. If the exposure risk is broken up into

[21] See the next paragraph for the treatment of "adding up" distributions. If the risk measures are treated as (probability) distributions, more complicated operations are needed.

categories for different radiation dose intervals, e.g. in categories of < 1 mSv, < 10 mSv, < 100 mSv, < 1000 mSv, < 10000 mSv, and > 10000 mSv, then simply adding up the exposure frequencies is certainly a possibility. However, most would see this as an inadequate representation of the actual accidental radiation dose risk. The issue of risk aggregation over risk metrics/risk measures (i.e. consequences) is particularly relevant for aggregating risk from multiple sources or units, for which mutual exclusivity is no longer a valid assumption. For direct risk measures commonly used in PSA, it is possible to distinguish between two important types of consequences.

The first type of consequence is defined over a binary event like e.g. fuel damage. In that case the consequence c_j is part of a Boolean domain and can be identified as c $\equiv 1 \in \{0,1\}$ and importantly $\neg c \equiv 0$. For aggregating the risk from two independent sequences for two sources A and B using an additive measure like $\varphi(l_{ij})$, it then follows.

$$s_{A,i} \xrightarrow{\varphi(l_{ij})} c_j \cup s_{B,k} \xrightarrow{\varphi(l_{kj})} c_j \;\; \Rightarrow \varphi_{A+B}(c_j) = \varphi(l_{ij}) * \varphi(l_{kj})$$
$$\Rightarrow E\left(\varphi_{A+B}(c_j)\right) = E\left(\varphi(l_{ij})\right) + E\left(\varphi(l_{kj})\right) - E\left(\varphi(l_{ij})\right) E\left(\varphi(l_{kj})\right)$$
$$s_{A,m} \xrightarrow{1-\varphi(l_{ij})} \neg c_j \cap s_{B,n} \xrightarrow{1-\varphi(l_{kj})} \neg c_j \;\; \Rightarrow \; \varphi_{A+B}(\neg c_j) = \left(1-\varphi(l_{ij})\right) * \left(1-\varphi(l_{kj})\right)$$
$$\Rightarrow E\left(\varphi_{A+B}(\neg c_j)\right) = 1 - E\left(\varphi_{A+B}(c_j)\right)$$

The extension to more than two sequences is straight forward. The important property is that aggregating risk over the sources for this binary type can be done by simply adding up (i.e. convoluting) the respective distributions for all sequences assigned to the respective consequence. For mean values of probability distributions, Boolean logic can be applied.

For the second type of consequence the issues are more complicated. This type of consequence is defined over a continuous or at least integer variable like e.g. the amount of release in Bq or the exposure in mSv. The example of release metrics is particularly relevant to ASAMPSA_E. In that case, a bounding or at least representative source term and therefore release value r (e.g. Bq of Cs-137) is assigned to each sequence. These are then put into a release category (assigned to a consequence c_j) based on release characteristics, e.g. if certain threshold values θ_c for the release are exceeded or not. This can be noted for two sources and two sequences as follows

$$s_{A,i} \xrightarrow{\varphi(l_{ij})} \{c_j | \, r_{A,i} \geq \theta_c\}, \quad s_{B,k} \xrightarrow{\varphi(l_{kj})} \{c_j | \, r_{B,j} \geq \theta_c\}$$

These sequences obviously contribute to $\varphi_{A+B}(c_j)$. But for sequences not assigned to consequence c_j, we only know that the release value is somewhere below the threshold value θ_c. The two sequence(s) for source A and B that do not reach the threshold value might be split up in two additional sequences, e.g.

$$s_{A,m} \xrightarrow{1-\varphi(l_{ij})} \{\neg c_j | \, r_{A,i} < \theta_c\} = s_{A,m1} \xrightarrow{\varphi(l_{m1\neg j})} \left\{\neg c_{j,1} | \, \frac{1}{2}\theta_c \leq r_{A,i} < \theta_c\right\} \cup s_{A,m2} \xrightarrow{\varphi(l_{m2\neg j})} \left\{\neg c_{j,2} | r_{A,i} < \frac{1}{2}\theta_c\right\}$$

and analogously for source B. Based on these simplifying assumptions the total contributions to $\varphi_{A+B}(c_j)$ can be derived by convoluting the distributions from three cases, i.e.

$$s_{A,i} \xrightarrow{\varphi(l_{ij})} \{c_j | \, r_{A,i} \geq \theta_c\} \;\cup\; s_{B,k} \xrightarrow{\varphi(l_{kj})} \{c_j | \, r_{B,j} \geq \theta_c\}$$
$$\cup \;\; s_{A,m1} \xrightarrow{\varphi(l_{m1\neg j})} \left\{\neg c_{j,1} | \, \frac{1}{2}\theta_c \leq r_{A,i} < \theta_c\right\} \cap s_{B,k1} \xrightarrow{\varphi(l_{k1\neg j})} \left\{\neg c_{j,1} | \, \frac{1}{2}\theta_c \leq r_{A,i} < \theta_c\right\}$$
$$\Rightarrow E\left(\varphi_{A+B}(c_j)\right) = E\left(\varphi(l_{ij})\right) \vee E\left(\varphi(l_{kj})\right) \vee \left(E\left(\varphi(l_{m1\neg j})\right) E\left(\varphi(l_{k1\neg j})\right)\right)$$

The important property is that for aggregating risk over the sources not only those sequences, which did reach the threshold for one source, need to be considered, but that there are in addition further contributions from sequences, which only jointly reach or exceed the threshold value. Importantly, extending the example described above to more than two sequences is in no way straight forward. This complicates matters significantly, because the definition of sequences for a multi-source model needs to take these additional contributions compared to the binary event case into account.

In fact, multi-source models for consequences based on continuous variables can only be decomposed into single-source models in an approximate way. When developing the accident sequences (i.e. event trees) for a multi-source model for these kinds of consequences, it is necessary to actively look for the combination of sequences which only jointly reach a consequence threshold. This will necessitate a more detailed development of sequences for scenarios near these consequence thresholds, especially considering potential correlations between the likelihood distributions of the respective single-source sequences.

We further point out that the source term $r_{A,i}$ for a certain (accident) sequence in general also depends on time. In order to determine $r_{A,i}(t)$ for a sequence, a representative reference time τ has to be defined. Then, the source term has to be integrated over that time period τ, like e.g. 72 hours, for each starting point t.

$$r_{A,i}(t) = \int_0^\tau r_{A,i}(t, t+\tau')d\tau'$$

If $r_{A,i}(t) \geq \theta_c$ then this sequence at the time t qualifies as a contributor to the c_j. The time average for the source term has then to be performed over those times for which the sequence qualifies for c_j.

$$r_{A,i} = \frac{1}{\int (t \in [0, T_{av}] | r_{A,i}(t) \geq \theta_c) dt} \int_{(t \in [0,T_{av}] | r_{A,i}(t) \geq \theta_c)} r_{A,i}(t)\, dt$$

Conversely, the average frequency over the time interval T_{av} needs to be computed over the whole interval with zero contributions if the sequence does not qualify.

$$\varphi(l_{ij})_{av} = \frac{1}{T_{av}} \int_{(t \in [0,T_{av}] | r_{A,i}(t) \geq \theta_c)} \varphi(l_{ij})(t)dt$$

This convention allows for a consistent computation of time-average release type risk measures. In most cases, the PSA models will not provide explicitly time-dependent source terms, nor will this be necessary. Then, common time-averaging over the likelihood will be sufficient. In some instances, e.g. for a shutdown PSA for an extended period shutdown (like 1 year), the time dependence of the source term might need to be considered.

8.3.2 Secondary Risk Measures and Risk Metrics

Secondary risk measures are understood to be aspects of risk that are derived from analysing the risk measure of a direct risk metric. The derived risk metric is again the aspect of risk which is quantified. The definition is extended to include also derived risk measures based on other secondary risk measures.

Obviously, secondary risk metrics/risk measures cannot be treated independently of the underlying risk metric/risk measure.

In order to define a secondary risk measure, the definition of the risk measure μ has to be extended. The uncertainty distribution $\mu(c_j)$ is a function of a (countable) number of inputs (parameters, generically with distributions) for the scenarios and the transition to the consequences, $\mu(c_j) = f(p_1, \dots p_N) = f(\{p_n\})$.

Use of secondary risk measures:

Regarding the use of secondary risk measures, there are no fundamental differences to CDF, time averaged.

However, the following specific differences should be considered. All secondary risk measures need to be evaluated at a specific point in time, T. For some, like ΔCDF, time-averaging actually makes sense.

$$\Delta CDF, T_{av} = \frac{1}{T_{\mathrm{av}}} \int_{0}^{T_{\mathrm{av}}} \Delta CDF(t) dt$$

For others, including most importance and sensitivity measures, the results are in general distinct depending on the time integration, for example (See below for more details):

$$FV_i\left(l_{CDF,T_{av}}\right) \neq \frac{1}{T_{\mathrm{av}}} \int_{0}^{T_{\mathrm{av}}} FV_i\left(l_{CDF}(t)\right) dt \neq E\left(\frac{1}{T_{\mathrm{av}}} \int_{0}^{T_{\mathrm{av}}} FV_i\left(\varphi\left(l_{CDF}(t)\right)\right) dt\right).$$

There is no simply rule on which of these measures provide valid and useful information and which do not. These kinds of in-depth investigations are outside of the scope of this report.

Relative risk measures

One obvious risk measure is to define the change in a (direct) risk measure (i.e. distribution) at certain events (i.e. parameter change from p_n to $\tilde{p}_n$ relative to a "baseline" risk value). Then, the change in the risk measure associated with the measure for metric c_j is

$$\Delta\, \mu(c_j) = \mu(c_j)|\tilde{p}_n - \mu(c_j)|p_n.$$

Important examples of this type of risk measure are changes in the core damage frequency ($\Delta\mathrm{CDF}$), which is used for PSA of NPP. The extension to groups of parameters (a subset $\{p_m\} \subset \{p_n\}$) exhibiting a change in parameter (distribution) values $\{\tilde{p}_m\}$ is evident.

Importance and sensitivity measures

A secondary risk metric I_n (importance measure) can be defined by investigating the change of $\mu(c_j)$ at suitable changes in the (distribution of) values of single parameters ($\delta(p_n) = \tilde{p}_n - p_n$) as

$$I_n = \frac{\delta\left(\mu(c_j)\right)}{K\,\delta(p_n)},$$

or

$$I_n = \frac{\mu(c_j)|\tilde{p}_n - \mu(c_j)|p_n}{K}$$

The factor K is used to normalize these measures. Often K is chosen as $K = \mu(c_j)|p_n$

Familiar examples are importance measures [41], [24], [28] like Fussel-Vesely importance but also Birnbaum importance or risk achievement worth for one basic event or one reliability parameter, as there are very few restrictions on $\delta(p_n)$.

The extension to groups of parameters (a subset $\{p_m\} \subset \{p_n\}$) exhibiting a simultaneous change in parameter (distribution) values $\delta(\{p_m\})$ is evident. Familiar examples are group importance measures, e.g. the Fussel-Vesely importance for a whole plant system.

Importance measures are highly relevant examples of secondary risk metrics/measures currently used in the PSA for NPP.

It should be noted that importance measures I_n are described by distribution functions in the generic case since $\mu(c_j)$ as well as the parameters p_n are defined by distribution functions as well. The current practice in PSA for

NPP rarely considers this property. Mostly, importance measures are calculated based on point estimates (e.g. mean values for the underlying parameters).

The aforementioned definition of an importance measure also covers a wide range of sensitivity measures. If sensitivity measures are understood as simply shifting parameter values (e.g. by a factor of 10), this simply corresponds to a specific choice of the variation of the parameter. If sensitivity calculations are instead understood as investigations on the influence of uncertainty distributions for selected parameters, this is enclosed via an appropriate choice of parameters (e.g. the distribution parameters for a failure rate distribution of the lognormal type).

Conditional consequence metrics

The risk model $\{s_i, c_j, \varphi(l_{ij})\}$ can be extended to include intermediate states e_k, i.e.

$$s_i \xrightarrow{\varphi(l_{ij})} c_j \Leftrightarrow s_i \xrightarrow{\varphi(l_{ik})} e_k \xrightarrow{\varphi(t(c_j)|e_k)} c_j$$

with conditional transition probabilities $t(c_j)|e_k$. The sequences $s_i \to e_k \to c_j$ are the constitutive elements of an event tree (sequence graph). The extension to multiple intermediate states is obvious.

With this definition, conditional consequence metrics/measures can be defined as

$$\mu(c_j|e_k) = \varphi(t(c_j)|e_k)$$

in a natural way. As examples, conditional core damage probability or conditional containment failure probability or system unavailability upon the condition of IE occurrence (often called system unavailability on demand) have been used for PSA of NPP.

Obviously, risk aggregation should not be done by summing up, even for the standard definition of the conditional risk measure. The rules for conditional probabilities have to be applied. Finally, whether risk aggregation (over scenarios or over consequences) results in a sensible risk measure is an issue in itself.

Statistical measures

Since risk measures (direct or derivative) are defined as distributions over $\mathbb{R}$ or $\mathbb{Z}$, the usual statistical tools for random variables can be applied. Important examples would be the computation of a covariance for a risk measure and a parameter, e.g.

$$cov(\mu(c_j), p_n) = E\left[\left(\mu(c_j) - \hat{\mu}(c_j)\right)(p_n - \hat{p}_n)\right],$$

where $\hat{\mu}(c_j)$, $\hat{p}_n$ denote the mean values and E the expectation of the joint distribution. With the standard deviations σ_μ, σ_p, this can be used to compute Pearson's correlation coefficient

$$\rho_{\mu,p} = \frac{cov(\mu(c_j), p_n)}{\sigma_\mu \cdot \sigma_p}.$$

Analogously, other statistical measures can be used to analyse the risk measure distributions, their properties and correlations. Currently, this is rarely done in PSA for NPP.

Risk indices

If certain properties or parameters are ranked based on another risk measure, this (formally) constitutes a derivative risk measure (a mapping on a subset of $\mathbb{Z}$). Such a ranking is properly called an index [32]. Obviously, risk indices can be defined for a lot of quantities, e.g. scenarios, consequences, parameters, etc. and these can be based on direct risk metrics (e.g. core damage frequency) or derivative ones (e.g. Fussel-Vesely importance). As these risk indices contain very little information about the actual underlying risk metric (and measure), pure ranking schemes will not be investigated in this report.

8.4 Risk Measures and Minimal Cut Sets

The relation between risk measures, event tree/fault tree models, and mininal cut sets merits some remarks. When constructing a risk model (i.e. PSA model) for a specific sequence $s_i \longrightarrow c_j$ in order to quantify l_{ij}, the event tree/fault tree model for the sequence $s_i \longrightarrow c_j$ describes the system failure function

$$f_{s_i,c_j}: \{X_1, \dots, X_n\} \to \{0,1\}.$$

Here, $\{X_1, \dots, X_n\}$ denotes the set of basic events (and initiating events and logical switches/house events) in the event tree/fault tree model. The system failure function can be described (in case of the rare event approximation) by the set of minimal cuts, i.e. minimal combinations of basic event failures $MC_i = (X_{i1} = 1) \wedge \dots \wedge (X_{ik} = 1)$. The list of m mutually exclusive minimal cuts $\{MC_i\} = \cup_{i=1}^{m} MC_i$ is often a good approximation to the system failure function in terms of the likelihood l, i.e.

$$l(f_{s_i,c_j} = 1) = l(\{MC_i\})|_{s_i,c_j} = l\left(\bigcup_{i=1}^{m} (X_{i1} = 1) \wedge \dots \wedge (X_{ik} = 1)\right).$$

The arguments in the previous section on risk model sequences being disjoint can be translated directly into (and should be interpreted as) arguments on the lists of cut sets being disjoint, i.e. all cuts being mutually exclusive. Moreover, risk measures in PSA for NPP (regarding their frequency or probability aspect) are usually quantified by computing the likelihood of each cut $l(MC_i) = l(X_{i1} = 1) \cdot \dots \cdot l(X_{ik} = 1)$ in the (mostly truncated) list and summing up the contributions with e.g. the Min Cut Upper Bound formula. In uncertainty analysis, the respective distributions can be determined, usually with a Monte Carlo approach.

Furthermore, it is well known that the mean value for a minimum cut set likelihood can be estimated from the mean likelihoods of its constituent basic event, if (and only if) the respective basic events are mutually not correlated. If the basic events are correlated, e.g. by the same failure rate, then the minimum cut mean value will depend on the respective distributions and may deviate significantly.

8.5 Model Representations

Some importance and sensitivity measures are based on two different representations of the model output (for a direct risk measure): Taylor series representation and High Dimensional Model Representation (HDMR).

8.5.1 Taylor series representation

The Taylor series representation of the function $f(\mathrm{x}) = \mathbb{R}^n \to \mathbb{R}$ with respect to the change $\Delta \mathrm{x}_i = \mathrm{x}_i - \mathrm{x}_i^0 \ \mathrm{i} = 1 \dots \mathrm{n}$ of the input variables is:

$$f(X) = f^0 + \sum_{i=1}^{n}\left(\frac{\partial f}{\partial x_i}\Delta x_i\right) + \sum_{i=1}^{n}\sum_{j=1}^{1}\left(\frac{\partial^2 f}{\partial x_i \partial x_i}\Delta x_i \Delta x_j\right) + \sum_{i=1}^{n}\sum_{j=1}^{n}\sum_{k=1}^{n}\left(\frac{\partial^2 f}{\partial x_i \partial x_i}\Delta x_i \Delta x_j \Delta x_k\right) + \dots$$

where $f^0 = f(x_1^0, \dots, x_i^0, \dots, x_n^0)$.

8.5.2 High Dimensional Model

Consider a function $f(\mathrm{X}): \mathbb{R}^n \rightarrow \mathbb{R}$ of a random variable $X \epsilon \mathbb{R}^m$ with a probability density function $p(X) = \prod_i p_i(x_i) \geq 0$.

According to the high dimensional model representation (HDMR), $f(X)$ can be written as the sum of terms which depend on an increasing number of input variables:

$$f(X) = f_0 + \sum_{i=1}^{n} f_i(x_i) + \sum_{i=1}^{n} \sum_{j=i+1}^{n} f_{ij}(x_i, x_j) + \ldots + f_{1 \ldots n}(x_1, \dots, x_n)$$

The constant term f_0 corresponds to the average value of f(x) with respect to all variables:

$$f_0 = \int \varphi(X) \cdot \prod_{k=1}^{n} p_k(x_k) dx_k$$

The *n* terms f_i are the "main effects"; each term depends on a single variable and is the difference between the average value of $f(x)$ with respect to all the variables but the one at issue (which is fixed) and the constant term:

$$f_i = \int f(X) \cdot \prod_{\substack{k=1 \\ k \neq i}}^{n} p_k(x_k) dx_k - f_0$$

The $\frac{n(n-1)}{2}$ terms f_{ij} are the “second order interaction”; each term depends on two variables and is the difference between the average value of $f(x)$ with respect to all the variables but the two variables at issue (which are fixed) and the terms previously estimated (Main effects and constant term):

$$f_{ij} = \int f(X) \cdot \prod_{\substack{k=1 \\ k \neq i,j}}^{n} p_k(x_k) dx_k - f_i(x_i) - f_j(x_j) - f_0$$

In general, the interaction term $f_{i \ldots m}$ is the difference between the average value of $\varphi(X)$ with respect to all the remaining variables $n - m$ and the contribution due to any lower order terms.

9 APPENDIX B: APPENDIX B: NUCLEAR POWER PLANT RISKS (FROM CCA)

This appendix provides further background on potential effects of accidental releases, which should be considered for the definition and interpretation of PSA Level 2 and Level 2+ risk measures.

According to [B.2] "More than five millions people live in areas of Belarus, Russia and Ukraine that are classified as 'contaminated' with radionuclides due to the Chernobyl accident (above 37 kBq/m^2 of Cs-137). Amongst them about 400 000 people lived in more contaminated areas – classified by Soviet authorities as areas of strict radiation control (above 555 kBq/ m^2 of Cs-137). … However, about 100 000 residents of the more contaminated areas receive more than 1mSv annually from the Chernobyl fallout. Although future reduction of exposure levels is expected rather slow, i.e. of about 3 to 5% per year, the great majority of dose from the accident has already been accumulated."

The global aspect of the area affected by a nuclear accident can be demonstrated also by measurements and assessments performed by various independent organizations. For instance, after the Chernobyl accident a world map on Cs-137 contamination was constructed and issued by European Commission Joint Research Centre, Institute for transuranium elements – Radioactivity Environmental Monitoring [B.3]. Also numerous measurements of radiation contamination after the Fukushima accident were performed. For example the German Federal Institute for Geoscience and Natural Resources simulated the dispersion of radioactivity released from the Fukushima accident [B.4]. Also the German Bundesamt für Strahlenschutz (BfS), die Physikalisch-Technische Bundesanstalt (PTB) und der Deutsche Wetterdienst (DWD) Luftstaubsammler performed measurements in 49 locations [B.5].

Also information is available, issued in Slovakia about two weeks after the Fukushima accident based on measurements performed at the Faculty of Mathematics and Physics of the University of Comenius in Bratislava [B.6]. Whereas 20th March (9 days after Fukushima accident) physicists measured activity 0.7 mBq/liter of I-131 in rain waters, in a sample from 28th March they recorded 500-fold increase (0.43 Bq/liter). After the Chernobyl accident the detailed report for needs of the Scientific Committee of United Nations Organization was processed, based on measurements of 29 Czechoslovak institutions including NPPs. The highest contamination of the ground due to Cs-137 was in the areas Dunajská Streda 12,200 Bq/m^2, Komárno 10,510 Bq/m^2, Žiar nad Hronom 8,470 Bq/m^2, Galanta 7,270 Bq/m^2, Nitra 6,980 Bq/m^2, Levice 6,410 Bq/m^2, Stará Ľubovňa 5,270 Bq/m^2, Nové Zámky 4,670 Bq/m^2, Lučenec 4,670 Bq/m^2, Dolný Kubín 4,430 Bq/m^2. (Tables 1.4.9 and 1.4.10, measured 17.6.1986, [B.7]). The measurements of agricultural products in Slovakia showed the following mean contamination (range in brackets): root crops and corn - under 10 Bq/kg, fodder crops – Ru-103 150 Bq/kg (0 to 600), Cs-134 240 Bq/kg, (20 to 1000), Cs-137 440 Bq/kg (40 to 2,000), mushrooms Cs-137 290 Bq/kg, Cs-134 130 Bq/kg.

Measured background in the above given Slovak areas (photons dose equivalent above the surface) was in the range 0.09 to 0.12 μSv/h, it means ~ 0.8 to 1mSv/yr ([B.7], Table 1.1.2). Average natural background worldwide is reported currently to be in total 2.4 mSv/yr, and the man made fallout as 0.007 mSv per year (averaged over the planet, quantification before Fukushima) [B.8]. The level of man-made fallout corresponds more or less to releases due to all atomic tests and mainly the Chernobyl accident that emerged within last 50 years. Professor Koprda from the Slovak University of Technology alleges the total amount of Cs-137 releases due to atomic tests as of order of magnitude 1.2 E^{22} Bq [B.9], which decayed in 2011 to level of 9.8 E^{21}Bq, corresponding to an average world value about E^{-5} Sv/yr (about 0.01mSv/yr). This number corresponds well with the reported background in Japan [B.10].

It is obvious that the natural background is 3 orders of magnitude higher than the averaged fallout from man-made nuclear activities and thus, the risks coming from man-made nuclear activities seem to be small.

The economic global impact was obvious immediately after the Chernobyl accident, since Ukraine has always been held to be a significant producer of various agricultural products, mainly sunflower seed oil. A total 7,843 km^2 of agricultural land was removed from service in the three countries affected by the accident (Belarus, Ukraine and Russia) and timber production was halted for a total 6,942 km^2 of forest [B.11]. According to the Chernobyl Forum as an initiative of the IAEA, in cooperation with the WHO, UNDP, FAO, UNEP, UN-OCHA, UNSCEAR, the World Bank and the governments of Belarus, the Russian Federation and Ukraine [B.2]"*The Chernobyl accident, and government policies adopted to cope with its consequences, imposed huge costs on the Soviet Union and three successor countries Belarus, the Russian Federation and Ukraine. Although these three countries bore the brunt of the impact, given the spread of radiation outside the borders of the Soviet Union, other countries (in Scandinavia, for instance) sustained economic losses as well. "… However, the magnitude of impact is clear from a variety of government estimates from 1990s, which put the cost of the accident, over two decades, at hundreds of billions of dollars." … In Ukraine, 5-7 percent of government spending each year is still devoted to Chernobyl-related benefits and programs. In Belarus, government spending in Chernobyl amounted to 22.3 percent of national budget in 1991, declining gradually to 6.1 percent in 2002."* According to [B.12] total spending by Ukraine on Chernobyl in 25 years after the accident makes about US$198 billion.

The immediate economic impact after the tsunami disaster in Japan was a reduction of export of many industrial products as well as agricultural products. In the first three months Japan lost about 3.5 % of total GDP which is about 1.6 E^{11} USD [B.13]. Farmers in Fukushima prefecture, Japan's fourth-biggest rice producer, may not plant the grain this year, which represents ~15% of Japan's total output. The drain on growth of Japanese economy will be 2 to 4% of GDP [B.14], which represents about 8.4 E^{10} to 1.7E^{11} USD per year.

The risk of such events as Chernobyl and Fukushima is of potentially very high level, and therefore should be carefully considered in analyses. These accidents show that severe accidents with large consequences having large impact happen, therefore it is not sufficient to treat nuclear safety as safety of a design of nuclear power plants at an individual (as far as plant/site) or local (as far as consequences) level. Nuclear safety is a global issue and therefore, common risk targets-common safety limits as repeatedly required by the IAEA and the European community have a real reason to be defined.

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10 APPENDIX C: PICKERING NPP FUEL DAMAGE STATES FREQUENCIES

The set of PDSF measures was discussed in section 2.13 mainly with a view to the CDF risk measure. As a further example, we provide a variant of a set of PDS risk measures explicitly referencing fuel damage criteria as used for Pickering NPP in Canada (cf. [90] and [91]) (Pickering, Canada):

Fuel damage category (FDC) frequency represents a collection of event sequences judged to result in a similar degree of potential fuel damage. The FDCs are used as end-states in the Level 1 event trees. Groupings of the fuel damage categories are used to transition from the Level 1 PRA to the Level 2 PRA (Reference [90] and [91]). The range of events or event sequences covered by the FDCs is defined by the scope of the PRA. From the event tree analysis, general types of accident sequences can be identified. They are presented below in general order decreasing severity of fuel damage:

- Severe Core Damage: Sequences with the potential for loss of core structural integrity.
- Limited Fuel Damage: Loss of fuel cooling requiring the moderator as a heat sink.
- Prolonged loss of heat sink.
  - Inadequate cooling to fuel in one or more core passes following a large loss of coolant accident (LOCA) with unsuccessful emergency coolant injection system (ECIS) initiation.
  - Sequences leading to fuel damage in one channel with and without accompanying automatic containment isolation (button-up).
- Negligible Fuel Damage: Inadequate cooling to fuel in one or more core passes following a large loss of coolant accident (LOCA) with successful ECIS initiation.

The FDCs used in Pickering PRA (Reference [90] and [91]) are presented in the following table.

Table 11: Pickering Fuel damage Categories [90]

FDC	Definition	Typical Events in FDC
FDC1	Rapid loss of core structural integrity.	Positive reactivity transient and failure to shutdown the reactor.
FDC2	Slow loss of core structural integrity.	LOCA with failure of HT inventory makeup and failure of moderator heat sink.
FDC3	Moderator required as heat sink in the short term (< 1 hr after reactor trip).	Small LOCAs and failures of HTS makeup before one hour, and successful moderator heat removal.
FDC4	Moderator required as heat sink in the intermediate term (1 to 24 hr after reactor trip).	Small LOCAs and failure of HTS makeup on demand or during mission before 24 hours, and successful moderator heat sink. A loss of all heat sinks leading to breaks in the HTS, with successful HTS inventory makeup.
FDC5	Moderator required as heat sink in the long term (> 24 hr after reactor trip).	LOCAs with and failure of HTS makeup after 24 hours, with successful moderator.
FDC6	Temporary loss of cooling to fuel in many channels.	Large LOCA with successful ECI.
FDC7	Single channel fuel failure with sufficient release of steam or radioactivity to initiate automatic containment button-up.	End-fitting LOCA and fuel ejection with successful ECI. LOCA stagnation feeder break and successful ECI.
FDC8	Single channel fuel failure with insufficient release of steam or radiation activity to initiate automatic containment button-up.	In-core LOCA and fuel ejection, with successful ECI. Large fuel blockage. LOCA stagnation feeder break.

FDC9	LOCAs with no fuel failure (ECIS successful); potential for significant economic impact.	Small LOCAs or large LOCA with successful ECI.
S	Success plant state. No fuel failure, ECIS not required.	LOCA with successful D2O makeup and long term heat sink. No LOCA events with a successful heat sink.
FDC1-SD	Rapid loss of core structural integrity.	Positive reactivity transient during shutdown and failure to terminate the event.
FDC2-SD	Slow loss of core structural integrity.	LOCA with failure of HT inventory makeup and failure of moderator heat sink.
FDC5-SD	Moderator required as heat sink in the long term (> 24 hr after reactor shutdown).	LOCAs with and failure of HTS makeup with successful moderator.
FDC7-SD	Single channel fuel failure with sufficient release of steam or radioactivity to initiate automatic containment button-up.	End-fitting failure with fuel ejection and successful ECI. Large flow blockage or stagnation feeder break and successful ECI.
FDC9-SD	LOCAs with no fuel failure (ECIS successful); potential for significant economic impact.	LOCAs with failure of D2O make-up, but successful ECI and a heat sink.
(1) SD: Shutdown		

Interface between PSA Level 1 and Level 2 [90]:

A subset of the FDCs (1-7), those that involve release of significant quantities of fission products from the core, is used to develop the interface between Level 1 and Level 2, the Plant Damage States (PDSs). The plant damage states serve to reduce number of the sequences assessed in the Level 2 analysis to a manageable number while still reflecting the full range of possible accident sequences and their impacts on the plant.

Only two FDCs are used to represent the range of sequences that result in severe core damage, FDC1 for rapid accident progression resulting from failures to shut down the reactor when required and FDC2 for all other sequences. FDC1 is conservatively assumed to cause early consequential containment failure and is assigned to a unique PDS, PDS1.

FDC2 is not assumed to result in immediate containment failure and was subdivided into three PDSs (2-4) to examine the potential for random and consequential failures of containment systems that could eventually lead to enhanced release to the environment:

- PDS2 represents sequences affecting a single unit with release into containment;
- PDS3 represents sequences affecting more than one unit;
- PDS4 represents single unit sequences with a release pathway that bypasses containment.

Areas of application:

Some of the areas of application of the fuel damage frequency is compliance with the country regulation. Ontario Power Generation (Reference [90] and [91]) uses the fuel damage categories (FDCs) to calculate the frequency of severe core damage, for comparison to the relevant Ontario Power Generation safety goal and the Canadian regulation. Severe core damage is defined to be the sum of the FDC1 and FDC2 frequencies (See previous section for the definition of FDC1 and FDC2).

Discussion:

The example from Pickering NPP is specific to CANDU reactor.

It is interesting to see that the Fuel Damage Categories (FDC) provide intermediary information before severe core damage and PDS. This a not a common practice for other NPPs but it is a good example to show that defining Plant Damage State is a plant specific activity.