Title	Guidance for the definition and application of probabilistic safety criteria
Author(s)	Knochenhauer, Michael; Holmberg, Jan-Erik
Citation	Proceedings of PSAM 10 International Probabilistic Safety Assessment & Management Conference, 7-11 June 2010, Seattle, Washington, USA, pp. Paper 245
Date	2010
Rights	This article may be downloaded for personal use only

Guidance for the Definition and Application of Probabilistic Safety Criteria

Michael Knochenhauer[a*] and Jan Erik Holmberg[b]
[a]Scandpower – Lloyd's Register, Stockholm, Sweden
[b]VTT Technical Research Centre of Finland, Espoo, Finland

Abstract: A guidance document has been developed as part of a four-year Nordic project dealing with the use of probabilistic safety criteria for nuclear power plants. The Guidance sums up, on the basis of the work performed throughout the project, issues to consider when defining and applying probabilistic safety criteria. The Guidance describes the terminology and concepts involved, levels of probabilistic safety criteria and relations between these, how to define a criterion, how to apply a criterion, on what to apply the criterion, and how to interpret the result of the application. It specifically deals with what makes up a probabilistic safety criterion, i.e., the risk metric, the frequency criterion, the PSA used for assessing compliance, and the application procedure for the criterion. It will also discuss the concept of subsidiary criteria, i.e., different levels of safety goals, their relation to defense in depth and to a primary safety goal in terms of health effects or other off-site consequences.

Keywords: PSA, Probabilistic Safety Criteria, Safety Targets, Safety Goals

1. INTRODUCTION

A guidance document [1] has been developed as part of the finalization of a four-year Nordic project dealing with the use of probabilistic safety criteria for nuclear power plants [2 and 3]. The project "The Validity of Safety Goals" was initiated in 2006 by NKS (Nordic Nuclear Safety Research) and NPSAG (Nordic PSA Group), and has interacted with an OECD/NEA task on probabilistic safety criteria in the NEA member countries [4]. The paper gives an overview of the Guidance, using information contained in all of these references.

2. AIM AND SCOPE OF THE GUIDANCE

The Guidance aims at describing, on the basis of the work performed throughout the project, issues to consider when defining, applying, and interpreting probabilistic safety criteria. Thus, the basic aim of the Guidance is to serve as a checklist and toolbox for the definition and application of probabilistic safety criteria. The Guidance describes the terminology and concepts involved, the levels of criteria and relations between these, how to define a probabilistic safety criterion, how to apply it, on what to apply it, and how to interpret the result of the application.

The Guidance specifically deals with what makes up a probabilistic safety criterion, i.e., the risk metric, the frequency criterion, the PSA used for assessing compliance, and the application procedure. It also discusses the concept of subsidiary criteria, i.e., different levels of safety goals, their relation to defense in depth and to a primary safety goal in terms of health effects or other off-site consequences.

3. DEFINING PROBABILISTIC SAFETY CRITERIA

3.1. Introduction

Initially, the definition of a probabilistic safety criterion needs to be discussed, including what makes up a criterion and what levels criteria can be defined on. Figure 1 gives an overview of some (but not

[*] mkn@scandpower.com

all) of the concepts that are involved when defining and applying probabilistic safety criteria, using criteria for core damage and release as an example.

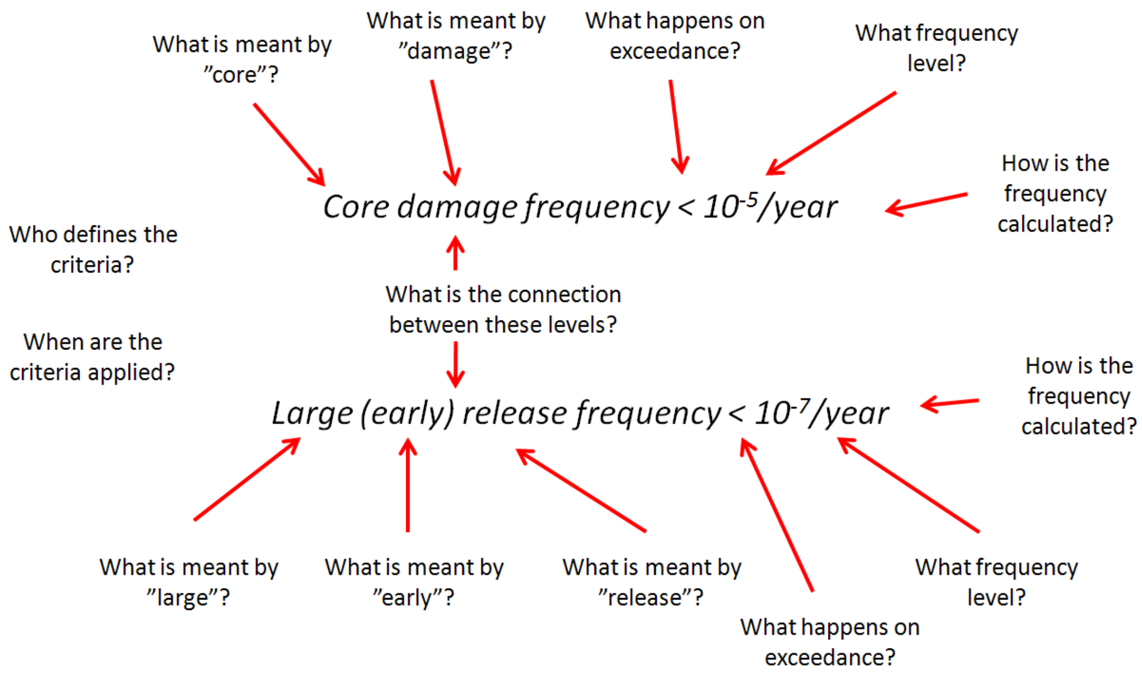

Figure 1. Some concepts involved when defining a probabilistic safety criterion

3.2. Levels of probabilistic safety criteria

Risk criteria related to the operation of nuclear power plants are defined on three levels:

- Society level
- Intermediate level
- Technical level

Society and intermediate level criteria

In many countries, nuclear safety is ultimately governed by qualitative criteria on society level, which are defined in nuclear legislation or issued by regulatory authorities. These criteria differ in wording between countries, but generally presuppose the "prevention of unreasonable risk to the public and the environment". Society level criteria are important as high-level statements, but cannot in themselves be used as a basis for defining numerical criteria.

Intermediate level criteria are more precise and can be both qualitative and quantitative. They typically define "unreasonable" risk by comparison with the levels of risks coming from other involuntary sources of risk, e.g., with fatality risks from other sources of energy production or cancer fatality risks from other unnatural causes to which an individual is generally exposed. Generally they express the requirement that "risks from use of nuclear energy shall or should be low compared to other risks to which the public is normally exposed". Thus, intermediate level criteria are the implicit basis for defining the primary safety goal, which requires an interpretation in numerical terms of what constitutes an unreasonable risk to an individual or to society.

3.3. Technical level criteria

Criteria on technical level are quantitative, and always in some way or other aim at deciding whether a risk is acceptable or not. Acceptability can be judged using criteria which are based on three basically different approaches:

– criteria which define acceptable risks,
– criteria which focus on controlling the risk increase, or
– criteria which define a negligible risk.

Criteria may be of four kinds, i.e., absolute, relative, differential or involving trade-off:

– Absolute Criteria
  Risk is expressed in absolute terms and judged against absolute risk criteria.
– Relative Criteria
  Risk is expressed in relative terms, e.g., in terms of the relative difference between absolute risks on two different levels.
– Differential Criteria
  With this type of criterion, the focus is on the absolute risk increase. Thus, a differential criterion may define the maximal allowed risk increase, e.g., Δf(core melt) < 10^{-7}/year.
– Trade-off Criteria
  This approach assumes a constant risk level, meaning that any changes resulting in additional risk must be compensated by changes reducing the risk back to the original level.

The focus in the guidance is on criteria for over-all assessment of PSA results. Therefore, criteria that are specific for risk informed applications, e.g., differential criteria and trade-off criteria, are not discussed in the guidance.

Criteria on technical level are typically defined on one or more of the following levels, which are all covered by the Guidance:

– Off-site consequence level (corresponding to PSA level 3)
– Radioactive release from plant level (corresponding to PSA level 2)
– Core or fuel damage level (corresponding to PSA level 1)
– Lower technical level (barrier strength, safety function, safety system)

3.4. Main constituents of a probabilistic safety criterion

A properly defined probabilistic safety criterion consists of four parts, which are all further described in separate sections:

– The definition of the criterion
  This defines the criterion, e.g., "the core damage frequency of a nuclear power plant shall be < 10^{-5}/year". In order for the criterion to be relevant, further definitions are required, e.g., of what is meant by "core damage", or by "< 10^{-5}/year".
– The scope of the criterion
  This defines what the criterion is to be applied on, e.g., "a full scope PSA for the power operation mode".
– The target of the criterion
  This defines the plants to which the criterion applies, e.g., "the criterion applies to new plants only".
– The application procedure
  This defines how the criterion is to be applied, including when to apply, how to apply and the consequences of a violation, e.g., "The criterion is to be applied in connection with every major PSA update. In case the criterion is violated, the reason shall be identified and, if needed, corrective actions related to the PSA model, or plant design or procedures, shall be initiated".

3.5 Definition of a probabilistic safety criterion

A probabilistic safety criterion is generally defined by defining a consequence, a metric for the consequence, a risk metric, and a frequency or probability.

- The consequence is the end state considered for a specific probabilistic safety criterion, e.g., the consequence may be “core damage” for a criterion related to PSA level 1.
- The metric is needed in order to define the consequence further, e.g., by defining “core to have occurred if the local fuel temperature in any part of the core has exceeded 1204 ºC.
- The risk metric is defined by assigning a frequency or probability to the metric, e.g., by measuring the risk from “core damage” by calculating the “core damage frequency”.
- The frequency or probability define the acceptance level for the risk metric, e.g., by stating that the “core damage frequency shall be shown to be < 10^{-5}/year”.

Some further definitions relate to the presentation and interpretation of the risk metric, i.e.:

- Consideration of uncertainties
  The criterion should state whether the application relates to the best estimate of the frequency or probability, or if it shall be related to some level of confidence.
- Justification of the definitions made
  Reference documents or supporting analyses are needed to justify the selected definitions, e.g., why the metric "core damage" is interpreted as "fuel cladding temperature > 1204 °C".

3.6 The scope of a probabilistic safety criterion

The scope of a probabilistic safety criterion is defined by the scope of the PSA used to calculate the frequency or probability defining the criterion acceptance level. The international overview performed within the OECD/NEA WGRISK project [4] indicates consensus about using a full scope PSA when applying probabilistic safety criteria (internal events, area events, and external events, full power and shutdown operating modes) PSA. The Nordic project [3] gives some additional views regarding the scope:

- Basically, every source of radioactive release needs to be included, but simplified screening should be acceptable for outside core events.
- All initiating events need to be included, but simplified screening should be acceptable in some cases.
- Every operational state challenging a safety function should be included, but some simplification may be acceptable.
- Regarding status during different life cycle phases, the focus should be on the operating phase, but the criteria need to be known and considered during design.

Thus, if the PSA is not full scope, it is recommended to perform simplified screening analyses for parts missing.

3.7. The target of a probabilistic safety criterion

The target of a probabilistic safety criterion is defined by the plants the criterion is applicable to. In the international overview performed within the OECD/NEA WGRISK project [4], several countries define different criteria for existing plants and new plants, or give the criteria different status. In many cases, probabilistic safety criteria use the same metric for existing and future plants, whereas the numerical values for the frequencies are a factor (typically 10) lower for future plants. In other cases, the criteria involve the same numerical values for the frequencies, but with status as limits for future plants and targets for existing plants. For modernization and life extension, generally the same criteria are applied as for operating plants.

In all countries, criteria are applicable at reactor level. One justification for this is the aim to be able to evaluate the safety of each individual reactor.

4. DEFINING THE CRITERIA

4.1. Off-site consequence criteria

Description

Off-site consequence criteria are most closely related to the primary safety goal, related to off-site health effects or environmental effects. In terms of application, a PSA level 3 is required to address off-site consequence criteria. Risks are divided into fatal acute or fatal late health risks and these can be calculated for an individual or a group. In both cases, risk is defined as the risk to the member of a critical group that receives maximum exposure from an accident. Typically acute health effects have a threshold dose value under which the probability of health effect is zero, but above which the probability of acute health effect is increased with increasing dose. Most late health effects do not have threshold values for dose. Based on these assumptions acute health effects can be expected in the vicinity of the release point, whereas late health effects appear in the public exposed to radiation over larger areas.

Frequency of doses criteria are expressed as rate of exposure in Sv/yr to the individual and/or probability of latent health effects. As off-site consequence criteria are defined for individuals and groups (sometimes differing among on-site personnel and public), and cover both acute and late effects, multiple criteria need to be defined.

Concepts involved

The concepts involved in defining a criterion for off-site consequences are shown and described in Table 1, using as an example one of several criteria defined by the UK HSE [5].

Table 1. Concepts involved in defining an off-site consequence criterion (with example)

Concept	Definition	Example
Consequence	Defines the health effects and the individual/group to which the criterion applies.	Accident resulting in a dose to individuals off-site.
Metric	Qualifies the consequence (in this case "health effect") in terms of a measurable magnitude.	Dose received in the interval 10 to 100 mSv
Risk metric	Defines how the risk is to be expressed.	Frequency of achieving a dose rate in the interval defined.
Frequency/ probability	Defines specific levels related to the frequency/probability.	ALARP (As Low As Reasonably Practicable) approach involving the definition of a basic safety limit (BSL) not to be exceeded, and a basic safety objective (BSO), under which the risk is considered to be broadly acceptable. BSL: $1 \cdot 10^{-4}$ /year BSO: $1 \cdot 10^{-6}$ /year

4.2. Release criteria

Description

Release criteria are related to radioactive release from the plant. In terms of application, a PSA level 2 is required to address release criteria. Internationally, the definition of what constitutes an unacceptable release differs a lot. The underlying reason for the complexity of the definition of an unacceptable release is largely the fact that it constitutes the link between the PSA level 2 results and an indirect attempt to assess health effects from the release. Such consequence issues are basically addressed in PSA level 3, and cannot be fully addressed in a PSA level 2.

The definition of release criteria involves many parameters, the most important ones being the time, the amount and the composition of the release. Additionally, other aspects may be of interest, such as the height above ground of the point of release. This means that there may be a need for multiple criteria, which is however usually very unusual.

Concepts involved

The concepts involved in defining release criteria are shown and described in Table 2, using as an example the release criterion defined by the SSM in Sweden [6] and by STUK in Finland [7]

Table 2. Concepts involved in defining a release criterion (with example)

Concept	Definition	Example
Consequence	Defines the consequence related to the release.	Unacceptable release with respect to long-term ground contamination.
Metric	Qualifies the consequence (in this case "release causing long-term ground contamination") in terms of a measurable magnitude.	Sweden: Release of Cs-137 in excess of an amount corresponding to 0,1% of the core inventory in a 1800 MWt BWR (equivalent to about 103 TBq of Cs-137). Finland: Release of > 100 TBq of Cs-137.
Risk metric	Defines how the risk is to be expressed.	Sweden: No risk metric has been defined by SSM. However, it is stated that a release exceeding the limit shall be" extremely unlikely", indicating consideration of an occurrence frequency. Finland: Frequency of exceeding the release limit.
Frequency/ probability	Defines specific levels related to the frequency/probability.	Sweden:"Extremely unlikely" has been interpreted by the Swedish utilities to indicate a limit in the interval 10^{-6} to 10^{-7} per year. Finland: The criterion is defined as a frequency limit, which is set to $5 \cdot 10^{-7}$ per year.

Figure 2 summarises the numerical criteria defined for large (early) releases, as presented in [4]. Typically, the definitions for "large release" is not the same for all organisations. However, it can be seen that objectives vary between 10^{-7} and 10^{-5} per year, which is a rather large spread. Magnitudes are sometimes based on IAEA safety goals suggested for existing plants, i.e., on the level of $1 \cdot 10^{-5}$ per year [8]. However, many countries define stricter limits, between $1 \cdot 10^{-6}$ per year and $1 \cdot 10^{-7}$ per year. Requirements for new plants are typically stricter (in terms of frequency) than for existing ones, and are mandatory as opposed to indicative.

Recommendation

The Guidance gives the following recommendations for probabilistic safety criteria related to releases:

- Probabilistic safety criteria should always be defined for unacceptable release.
- It may be considered to define more than one release criterion, related to at least acute health effects and long-time effects.
- If the over-all scope of the probabilistic safety criteria also includes outside core events, sources of radioactivity outside the core will also need to be addressed, at least in a simplified conservative manner.

– An ALARP approach is used in many countries, and has some advantages from the risk management point of view. It is therefore recommended to consider introducing ALARP type criteria.
– Regarding the frequency criterion, a limit on the level of $1 \cdot 10^{-7}$ per year (as defined in Sweden) seems to be unusually strict. Internationally, typical values for a frequency limit is about 1·10-6 per year, with the objective one order of magnitude lower, i.e., at 1·10-5 per year.
– The definition of the consequence and risk metric needs to be done and documented with care, including proper justification and references.

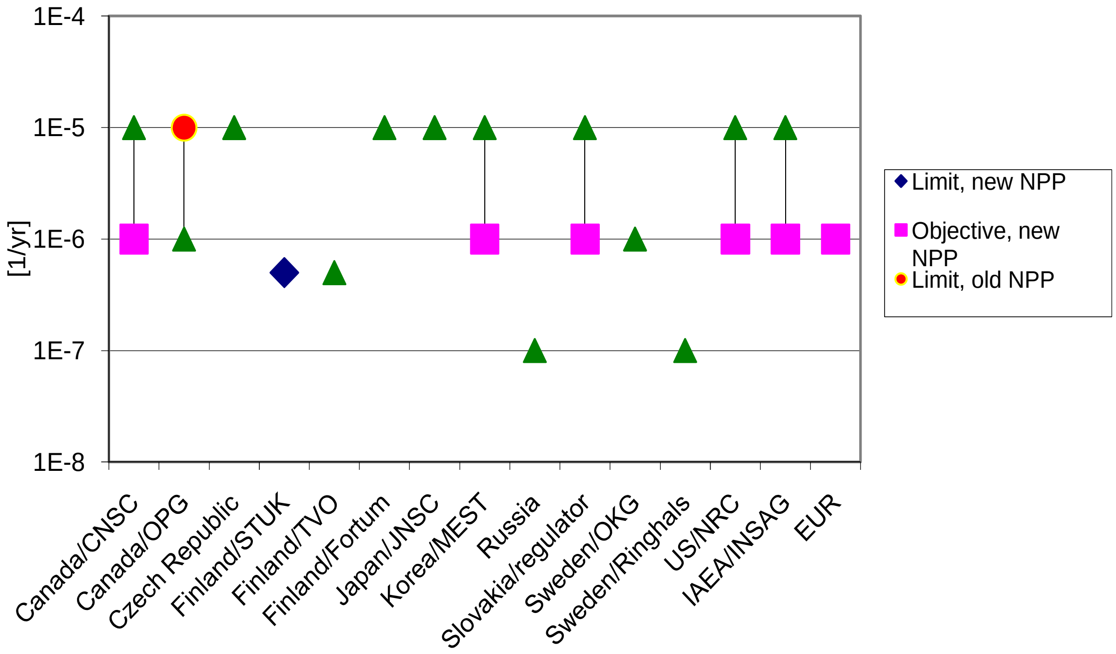

Figure 2. Numerical criteria defined for large release[1]

4.3. Core damage criteria

<u>Description</u>
Core damage criteria are related to damage to the fuel in the core. In terms of application, a PSA level 1 is required to address core damage criteria. It is worth noting, that there is some vagueness in the use of the concept "core damage", as fuel may be damaged or overheat in other locations than the core. The definition of what constitutes a core damage is rather homogenous among countries using the criterion, usually defined as local fuel temperature above 1204 ºC, i.e., the limit defined in 10 CFR 50.46 [9].

<u>Concepts involved</u>
The concepts involved in defining a criterion for core damage are shown and described in Table 3, using as an example criteria defined for the OKG by E.ON Nordic [10]

Figure 3 summarises numerical criteria defined for core damage, as presented in [4]. The frequency limits regarding core damage vary between $1 \cdot 10^{-4}$ and $1 \cdot 10^{-6}$ per year. The criterion is usually justified by reference to USNRC and/or IAEA documents, or by comparison with international practice. The IAEA core damage criteria suggested for existing plants are on the level of $1 \cdot 10^{-4}$ per year [8]. Requirements for new plants are typically stricter (in terms of frequency) than for existing ones, and are mandatory as opposed to indicative.

[1] The definition and timing of "large release" varies among the countries

Table 3. Concepts involved in defining core damage criteria (with example)

Concept	Definition	Example
Consequence	Defines the consequence related to the fuel overheating.	Severe core damage
Metric	Qualifies the consequence (in this case "severe core damage") in terms of a measurable magnitude.	"Severe" is not qualified, but previous versions of the safety policy have referred to 10 CFR 50.46 (local fuel temperature above 1204 ºC).
Risk metric	Defines how the risk is to be expressed.	Frequency of exceeding the limit. Note: As long as "severe" is not defined, there is some vagueness in the definition of the risk metric.
Frequency/ probability	Defines specific levels related to the frequency/probability.	The criterion is defined as a frequency limit, which is set to 1·10-5 per year.

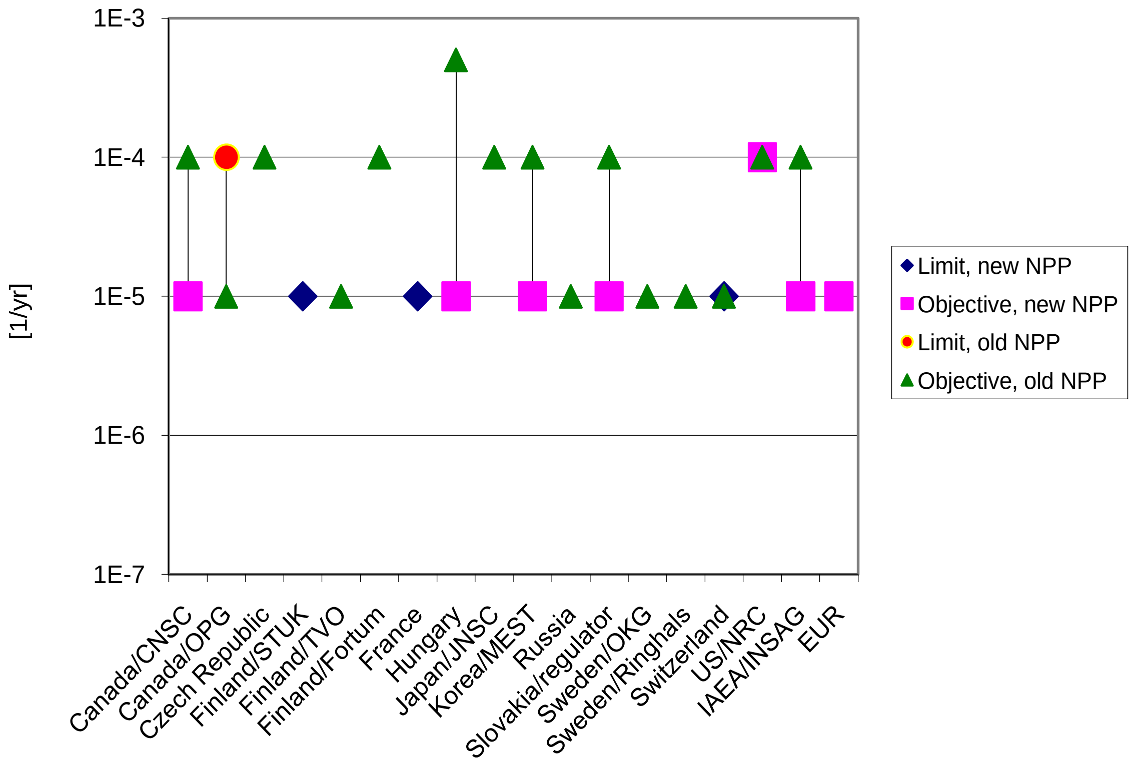

Figure 3. Numerical criteria defined for core damage.

<u>Recommendation</u>

The Guidance gives the following recommendations for probabilistic safety criteria related to releases:

- Probabilistic safety criteria should always be defined for core damage.
- It might be considered to use a more general wording in order to include fuel damage in other locations than the core, e.g. the fuel pool. On possibility would be to use the term "Fuel damage" or "Fuel over-heating".
- If the over-all scope of the probabilistic safety criteria also includes outside core events, sources of radioactivity outside the core will also need to be addressed, at least in a simplified conservative manner.

- An ALARP approach is used in many countries, and has some advantages from the risk management point of view. It is therefore recommended to consider introducing ALARP type criteria.
- Regarding the frequency criterion, typical values internationally for a frequency limit is about $1 \cdot 10^{-5}$ per year, with the objective one order of magnitude lower, i.e., at $1 \cdot 10^{-4}$ per year.
- In most cases, the conditional probability to criteria for unacceptable release is one order of magnitude.
- The definition of the consequence and risk metric needs to be done and documented with care, including proper justification and references.

4.4. Lower level criteria

Description
In the Guidance, the concept of lower level criteria applies both to criteria that are defined on a lower technical level than core damage, and on criteria on any level related to barrier strength. In all of these cases, the criteria will help in assessing the strength of the defence in depth.

Concepts involved
The concepts involved in defining a lower level criterion are the same as on higher levels, but the definitions may obviously differ considerably from case to case. In Table 4, an example is given for a containment integrity criterion.

Table 4. Concepts involved in defining lower level criteria (example for containment integrity criterion)

Concept	Definition	Example
Consequence	Defines the consequence related to the fuel overheating.	Loss of containment integrity (resulting in an unacceptable release) after core damage has occurred.
Metric	Qualifies the consequence (in this case “loss of containment integrity”) in terms of a measurable magnitude.	Must be based on the metric already defined for the criteria on the levels of core damage and release.
Risk metric	Defines how the risk is to be expressed.	Probability of exceeding the metric given by the release criterion, given that the metric given by the core damage criterion has been exceeded.
Frequency/ probability	Defines specific levels related to the frequency/probability.	The criterion is defined as a conditional probability, with a limit set to 0,1.

Recommendation
The Guidance gives the following recommendations for lower level criteria:
- Lower level criteria can be useful for assessing barrier strength, especially in a defence in depth context. In order to create a connection with defence in depth, it is recommended to consider defining barrier strength criteria for higher technical levels.
- Lower level criteria can be useful as design guidance on lower technical levels, which would considerably broaden the usefulness of probabilistic safety criteria. However, few such applications have been made to date, and in order to assure relevance in the definition of lower level criteria, it is recommended to investigate this issue further.
- In case barrier strength criteria are defined for higher technical levels, the definition of consequence and risk metric must be based on the consequences and metrics already defined for the criteria on the higher technical levels.
- The definition of the consequence and risk metric needs to be done and documented with care, including proper justification and references.

5. CONCLUSION

Based on the wide scope of the Nordic project [2 and 3] in terms of exploring basic concepts involved in the definition and application of probabilistic safety criteria, and of experiences and needs expressed by Nordic utilities and authorities, as well as on the broad international overview provided by the OECD/NEA WGRISK task [4], a Guidance for the definition and application of probabilistic safety criteria has been developed [1]. The Guidance includes a detailed context description as well as practical recommendations for defining criteria on all technical levels. It is believed that the Guidance will be of great use in the further development of PSA and Risk Informed applications in Sweden and Finland.

Acknowledgements

We thank the NKS (Nordic nuclear safety research) and the members of NPSAG (Nordic PSA Group) and SAFIR-2010 (The Finnish Research Programme on Nuclear Power Plant Safety) for financial support of the project as well as for other input.

References

[1] Holmberg, J.-E. and Knochenhauer, M., *Guidance for the Definition and Application of Probabilistic Safety Criteria;* To be published 2010

[2] Holmberg, J-E. and Knochenhauer, M., *Probabilistic Safety Goals. Phase 1 — Status and Experiences in Sweden and Finland.* SKI Report 2007:06, Swedish Radiation Safety Authority (SSM), 2007, Stockholm.

[3] Holmberg, J.-E. and Knochenhauer, M., *Probabilistic Safety Goals – Final Report;* To be published 2010

[4] *Probabilistic Risk Criteria and Safety Goals*, NEA/CSNI/R(2009)16, OECD, Nuclear Energy Agency, Committee on the Safety of Nuclear Installations, 2009, Paris.

[5] HSE, *Nuclear Installations Inspectorate (NII) Safety Assessment Principles for Nuclear Facilities*, HSE 2006 Edition Revision 1

[6] SKI / SSI; *Utsläppsbegränsande åtgärder vid svåra härdhaverier (Release mitigating measures after severe accidents);* SKI ref 7.1.24 1082/85; SKI / SSI; 1985

[7] STUK; *Probabilistic safety analysis in safety management of nuclear power plants*; Guide YVL-2.8. ISBN 951-712-786-3; STUK; 2003

[8] IAEA; *Basic Safety Principles for Nuclear Power Plants*. 75-INSAG-3 Rev. 1. INSAG-12; IAEA Safety Series No. 75-INSAG-12. ISBN 92–0–102699–4; IAEA; 1999

[9] 10 CFR 50.46, *Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors*, 10 CFR 50.46, section 1b

[10] Larsson, Stig-Erik; *E.ON Nordic Säkerhetspolicy för kärnkraft (E.ON Nordic Safety Policy for Nuclear Power Plants)*; EON/ SKKÖT-050609-01; E.ON Kärnkraft Sverige AB; 2005