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Probabilistic Safety Goals Phase 3 - Status Report

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Abstract

The first phase of the project (2006) described the status, concepts and history of probabilistic safety goals for nuclear power plants. The second and third phases (2007–2008) have provided guidance related to the resolution of some of the problems identified, and resulted in a common understanding regarding the definition of safety goals. The basic aim of phase 3 (2009) has been to increase the scope and level of detail of the project, and to start preparations of a guidance document. Based on the conclusions from the previous project phases, the following issues have been covered:

- Extension of international overview. Analysis of results from the questionnaire performed within the ongoing OECD/NEA WGRISK activity on probabilistic safety criteria, including participation in the preparation of the working report for OECD/NEA/WGRISK (to be finalised in phase 4).
- Use of subsidiary criteria and relations between these (to be finalised in phase 4).
- Numerical criteria when using probabilistic analyses in support of deterministic safety analysis (to be finalised in phase 4).
- Guidance for the formulation, application and interpretation of probabilistic safety criteria (to be finalised in phase 4)

Key words

Safety Goals, PSA, Safety Targets, ALARP, Decision criteria, Risk informed decision making

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Probabilistic Safety Goals

Phase 3 Status Report

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Abbreviations

ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
BWR	Boiling water reactor
CDF	Core damage frequency
CET	Containment event tree
CFF	Containment failure frequency
CLI	Criteria for limiting impact (in EUR)
CSNC	Canadian Nuclear Safety Commission
DBA	Design Basis Accident
DID	Defence-in-depth
DSA	Deterministic Safety Analysis
EOP	Emergency operating procedures
EPR	European Pressurized Reactor
ET	Event tree
EUR	European Utility Requirements
FKA	Forsmarks Kraftgrupp AB
FT	Fault Tree
HRA	Human reliability analysis
HSE	Health and Safety Executive (UK)
IAEA	International Atomic Energy Agency
IE	Initiating event
INES	International Nuclear Event Scale (IAEA)
JAEA	Japan Atomic Energy Agency
LERF	Large early release frequency
LOCA	Loss of coolant accident
LRF	Large release frequency
LWR	Light water reactor
NEA	Nuclear Energy Agency of OECD
NII	Nuclear Installations Inspectorate
NKS	Nordic nuclear safety research
NPP	Nuclear power plant
NPSAG	Nordic PSA Group
OECD	Organisation for Economic Co-operation and Development
PSA	Probabilistic safety assessment

PWR	Pressurised water reactor
RC	Release category
RPS	Reactor protection system
SAP	Safety assessment principle (UK HSE)
SAR	Safety Analysis Report
SG	Safety goal
SKI	Swedish Power Nuclear Inspectorate (Statens kärnkraftinspektion); <i>(until 2008 – now part of SSM)</i>
SSC	Systems, structures and components (of a nuclear power plant)
SSI	The Swedish Radiation Protection Authority (Statens strålskyddsinstitut); <i>(until 2008 – now part of SSM)</i>
SSM	Swedish Radiation Protection Authority (Strålsäkerhetsmyndigheten)
STUK	Radiation and Nuclear Safety Authority of Finland (Säteilyturvakeskus)
TVO	Teollisuuden Voima Oy
U.S.NRC	United States Nuclear Regulatory Commission
VTT	Technical Research Centre of Finland
WG	Working Group (of OECD/NEA)

Summary

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

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

The first phase of the project (2006) provided a general description of the issue of probabilistic safety goals for nuclear power plants, of important concepts related to the definition and application of safety goals, and of experiences in Finland and Sweden. The second and third phases (2007–2008) has been concerned with providing guidance related to the resolution of some of the problems identified, such as the problem of consistency in judgement, comparability of safety goals used in different industries, the relationship between different levels of criteria, and the utilisation of numerical criteria when using probabilistic analyses in support of deterministic safety analysis. In parallel, additional context information has been provided. This was achieved by extending the international overview by contributing to and benefiting from a survey on PSA safety criteria which was initiated in 2006 within the OECD/NEA Working Group Risk. Finally, work on providing general guidance concerning the formulation, application and interpretation of probabilistic criteria has been initiated on the basis of project experiences.

The results from the project can be used as a platform for discussions at the utilities on how to define and use quantitative safety goals. The results can also be used by safety authorities as a reference for risk-informed regulation. The outcome can have an impact on the requirements on PSA, e.g., regarding quality, scope, level of detail, and documentation. Finally, the results can be expected to support on-going activities concerning risk-informed applications.

Acknowledgements

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1 Introduction

The project “The Validity of Safety Goals” has been financed jointly by NKS (Nordic Nuclear Safety Research), SSM (Swedish Radiation Protection Authority) and the Swedish and Finnish nuclear utilities. The national financing went through NPSAG, the Nordic PSA Group (Swedish contributions) and SAFIR2010, the Finnish research programme on NPP safety (Finnish contributions).

The project is performed in four phases during 2006–2009, of which this project status report covers work carried out during phase 3 (mainly during 2008). An overview of the entire project is given in Figure 1.

BASIS	<ul style="list-style-type: none"> • CONCEPTS • DECISION THEORETIC BACKGROUND • EVOLVEMENT OF SAFETY GOALS • EXPERIENCES FROM APPLICATION AND INTERPRETATION IN SWEDEN AND FINLAND • LIMITED INTERNATIONAL OVERVIEW • ISSUES FOR FURTHER ANALYSIS <ul style="list-style-type: none"> ○ USE OF SAFETY GOALS IN DECISION MAKING ○ AMBIGUITIES IN DEFINITIONS OF SAFETY GOALS ○ TREATMENT OF UNCERTAINTIES IN THE APPLICATION OF SAFETY GOALS ○ AMBIGUITIES IN THE SCOPE OF SAFETY GOALS ○ SAFETY GOALS ON DIFFERENT LEVELS (SUBSIDIARY CRITERIA) ○ SAFETY GOALS FOR NEW/OPERATING PLANTS 	PHASE 1	OECD NEA WG Risk “PROBABILISTIC RISK CRITERIA FOR NPPs”
ELABORATION	<ul style="list-style-type: none"> • CONSISTENCY IN USAGE OF SAFETY GOALS • CRITERIA FOR ASSESSMENT OF RESULTS FROM PSA LEVEL 2 • SAFETY GOALS RELATED TO OTHER MAN-MADE RISKS IN SOCIETY • EXPANSION OF INTERNATIONAL OVERVIEW <ul style="list-style-type: none"> ○ WG RISK TASK ON SAFETY GOALS 	PHASE 2-4	
GUIDANCE	<ul style="list-style-type: none"> • USE OF SUBSIDIARY CRITERIA • USE OF PROBABILISTIC ANALYSES IN SUPPORT OF DETERMINISTIC SAFETY ANALYSIS • GUIDANCE FOR THE FORMULATION, APPLICATION AND INTERPRETATION OF PROBABILISTIC SAFETY CRITERIA 	PHASE 3-4	

Figure 1. Overview of the 4-year NKS project “The Validity of Safety Goals”(2006–2009)

The first phase of the project was largely carried out during 2006 With the aim to discuss and document current views, mainly in Finland and Sweden, on the use of safety goals, including both benefits and problems. The work has clarified the basis for

the evolvement of safety goals for nuclear power plants in Sweden and Finland and of experiences gained. This was achieved by performing a rather extensive series of detailed interviews with persons who are or have been involved in the formulation and application of the safety goals. Results of phase 1 have been published in two parallel reports issued by NKS [NKS-153], and SSM [SKI 2007:06]. The report presents the project context and a background to safety goals, as well as a historical review describing reasons for defining safety goals, context of goals and experiences. A number of specific issues related to the definition, interpretation and use of probabilistic safety goals were also identified and discussed. Towards the end of project phase 1, the OECD/NEA Working Group RISK started preparations for carrying out a task aimed at mapping probabilistic safety criteria in use in the member countries, and at collecting experiences from application of probabilistic criteria. The task was defined in cooperation with the NKS project, and the task leader (Phillip Hessel from the CNSC) has participated and presented status reports at the NKS Safety Goals project seminars in 2006-2008.

The second phase increased the scope and level of detail of the project by addressing a number of specific issues related to the application and use of safety goals, i.e.: consistency in the usage of safety goals, numerical criteria when using probabilistic analyses in support of deterministic safety analysis, criteria for assessment of results from PSA level 2 (criteria for off-site consequences). It also included the addition of a more systematic overview of international safety goals and experiences from their use, including participation in the OECD/NEA WGRISK Task 2006:2 “Probabilistic safety criteria” [NEA_2009], and a concise review of safety goals related to other man-made risks in society, with focus on the railway and oil and gas industries. Results were reported in [NKS-172].

The basic aim of phase 3 has been to increase the scope and level of detail of the project, and to start preparations for preparing a guidance document in the final project phase. Based on the conclusions from the previous project phases, the following issues were selected for analysis:

- Analysis of results from the questionnaire, performed within the ongoing OECD/NEA WGRISK activity on probabilistic safety criteria, including participation in the preparation of the working report for OECD/NEA/WGRISK (to be finalised in phase 4).
- Use of subsidiary criteria and relations between these (to be finalised in phase 4).
- Numerical criteria when using probabilistic analyses in support of deterministic safety analysis (to be finalised in phase 4).
- Guidance for the formulation, application and interpretation of probabilistic safety criteria (to be finalised in phase 4)

The results of this project phase were presented at a project seminar in Stockholm in December 2008 [SG_Semin_2008]. The project has also been presented with two papers at PSAM 9, an international conference on Probabilistic Safety and Management [PSAM9-0428 and PSAM9-0443], at the Nordic PSA Castle Meeting in June 2008 [PSA_CM_2008], and at the joint NKS-R and NKS-B seminar in Stockholm in March 2009 [NKS_Semin_2009].

This document includes the following parts:

Chapter 1. Introduction

Background; Aim and scope.

Chapter 2. WGRISK Task on Probabilistic Safety Criteria in member countries

Overview of the status within the task “Probabilistic Safety Criteria” conducted within the OECD/NEA Working Group Risk.

Chapter 3. Validation of subsidiary risk criteria

Discussion of subsidiary (lower level) criteria, including definitions of subsidiary criteria and approaches for validation of criteria.

Chapter 4. Numerical criteria when using probabilistic analyses in support of deterministic safety analysis

Description of some activities dealing with the relationship between the levels of defence in depth and PSA or other probabilistic analyses.

Chapter 5. Guidance for the definition and use of probabilistic safety criteria

Initiation of the most important phase 4 activity (development of guidance document) by summarising conclusions from the project workshop in December 2008.

Chapter 6. Conclusions

Conclusions, including a summary of planned activities for phase 4 of the project.

2 WGRISK Task on probabilistic safety criteria in member countries

2.1 Background of the task

OECD/NEA Working group RISK initiated in 2006 a task group on probabilistic safety criteria. The objective of the task is to review the rationales for definition, the current status, and actual experiences regarding the use of probabilistic safety goals and other PSA related numerical risk criteria in the member states.

The scope includes the whole range of safety goals, i.e., societal risk, off-site release, core damage, and lower level goals. The focus is on experiences from actual use of the safety goals for existing installations, including procedures used, problems related to the technical application of the criteria, and consequences for the status and use of PSA. Both regulatory criteria and criteria defined and used by utilities are covered.

During 2007, a questionnaire was prepared and sent to the member countries. In total 19 responses have been received from 13 regulatory bodies and 6 utilities (Canada, Finland and Sweden). The responses have been analysed during 2008 and results will be reported to OECD/NEA during 2009 [NEA_2009].

2.2 Status of probabilistic criteria

There are considerable differences in the status of the numerical risk criteria that have been defined in different countries. Some have been defined in law or regulations and are mandatory, some have been defined by the regulatory authority (which is the case in the majority of countries where numerical risk criteria have been defined), some have been defined by an authoritative body and some have been defined by plant operators or designers. Hence there is a difference in the status of the numerical risk criteria which range from mandatory requirements that need to be addressed in law to informal criteria that have been proposed by plant operators or designers for guidance only.

The following categories of statuses of the criteria can be seen:

- A legally strict value to be fulfilled. Design must be changed, if the criterion is not met. In some countries probabilistic safety criteria are applied in this manner for new NPPs.
- A strict value but not legally bounding. The value should not normally be exceeded. Some utilities define this kind of status for their NPPs.
- Target value, orientation value, expectation, or safety indicator. If the target is not met, design improvements should be considered taking into account cost-benefit considerations or the ALARP¹ principle. Targets denote a boundary that, if surpassed, will often lead to increased regulatory oversight, but is used as one piece of information (out of several) in the regulatory process (risk-informed not risk-based).

¹ In the context of this report, the concepts ALARP and ALARA are considered to have the same meaning.

In most countries, probabilistic risk criteria are defined and applied as target values, orientation values or safety indicators. Strict criteria are applied for new NPPs in some countries, e.g., Finland, the Netherlands and Switzerland.

2.3 Comparison of criteria for new and operating plants

In several countries, different criteria apply to existing plants and new plants, or the criteria have different status. For modernization and life extension, generally the same criteria are applied as for operating plants. The following categories of statuses can be seen:

- Probabilistic risk criteria are the same for existing and future plants, e.g., Switzerland.
- Probabilistic risk criteria are defined similarly for existing and future plants, but the numerical values for the frequencies are a factor (typically 10) lower for future plants, e.g., Canada, Czech Republic, Hungary, Korea, and Slovakia.
- Probabilistic risk criteria involve the same numerical values for the frequencies, but are considered as limits for future plants and targets for existing plants, e.g., Finland.
- Probabilistic risk criteria are defined only for existing plants, since new plants are at the time not considered, e.g., Sweden.
- No numerical risk criteria have been defined for new plants. However, there is a general requirement that the level of risk should be comparable to (or lower than) the risk from existing plants, e.g., Japan.

Band criteria (limit and objective) are explicitly used only by few organisations, e.g. HSE in the UK. Band criteria have also been supported by PSA users in Nordic countries. The reasoning is that it can be useful to define several levels of criteria, and a limit and an objective have different usage. Objectives can be set at a more demanding level, e.g., to support design. However, strict limits may be easier to communicate with the public.

2.4 What probabilistic risk criteria exist?

The questionnaire defined three levels of probabilistic risk criteria, as done by e.g. U.S.NRC:

- at society level (such criteria are mainly qualitative),
- at an intermediate level (such criteria can be quantitative and/or qualitative)
- at a technical level (quantitative)

The separation between society level and intermediate level is not always clear.

Of the 13 responding regulatory bodies, 8 have defined society level criteria. These criteria are generally set in the mandate of the regulatory body. One out of the six responding utilities has declared having a society level criterion.

Of the 13 responding regulatory bodies, 8 have defined intermediate level criteria. One out of the six responding utilities has declared having an intermediate level criterion.

The criteria generally indicate that “The risk from use of Nuclear Energy shall/should be low compared to other risks to which the public is normally exposed”.

On the Technical level, a rather large number of different probabilistic risk criteria are indicated in the responses:

- Core damage criteria
 - Core Damage frequency
- Release criteria
 - Large Release frequency
 - Small Release frequency
- Health risk criteria
 - Individual risk of fatalities
 - Frequency of doses
 - Societal risk
- Containment criteria
 - Containment Failure Frequency
 - Conditional containment failure probability
- Out of scope for the WG RISK task
 - Systems reliability targets
 - Instantaneous risk

2.5 Consideration of uncertainty

The responses to the questionnaire show a large consensus, all respondents stating that the comparison with probabilistic safety criteria should use the “best estimate” of the PSA results. Several respondents note that setting the criteria with uncertainty would be equivalent to setting a goal at a different level, without any added value.

2.6 When and how do probabilistic risk criteria apply?

A main use of risk criteria for operating plants is when the study is updated:

- For six respondents, the PSA supporting evaluation of the risk criteria shall be updated within the framework of the periodic safety review (generally 10 years).
- One country (and its utilities) requires the PSA supporting evaluation of the risk criteria to be updated every 3 years, or after significant modifications to the plant.
- One country (and its utilities) requires the PSA supporting evaluation of the risk criteria to be kept up to date (on design modifications).
- One utility updates the PSA every year and on plant modifications.

Four regulatory bodies and five licensees use the risk criteria to assess the impact on risk of design modifications in the plant. Four of them indicate they use the risk criteria

for assessing the impact on risk (and the appropriate response) from incidents and/or on discovery of new information.

The received response show considerable differences between the different countries regulatory regimes. As the risk criteria are generally considered as indicators or orientation values, no regulatory actions are expected on non-compliance with a probabilistic safety criterion.

Practically, there is a consensus on finding the reasons for the non-compliance and identification on the way to overcome it. However, when indicated, there is also a consensus for new builds, where not meeting the probabilistic risk criteria would prevent the regulatory body from granting an operating license.

2.7 Experience on implementation of probabilistic risk criteria

The information obtained from the application of probabilistic risk criteria is often used for:

- general safety improvements
- plant modifications (including procedures)
- system upgrades
- decision making
- temporary configurations
- identification of functional dependencies

The general experience from the implementation of risk criteria is positive. Respondents who have implemented criteria have experienced various benefits. In a number of cases, design weaknesses or procedural weaknesses in NPPs have been identified using PSA and PSA criteria, resulting in the introduction of safety improvements. More than half of the respondents describe how the implementation of risk criteria and safety goals have lead to plant modifications in order to meet the probabilistic risk criteria. One of the respondents also described how, using PSA, changes suggested on a deterministic basis have been avoided.

Furthermore, the implementation of safety goals often emphasizes the need for more detailed and realistic PSA models, since conservative assumptions in the PSA often make the calculated risk unnecessarily high. It appears that the use of safety goals has increased the focus on the correctness and quality of PSA models. One problem that may be highlighted, is the scope of the PSAs, i.e., results from limited scope PSAs may be harder to assess and difficult to compare to probabilistic safety criteria.

Some respondents emphasize the importance of using PSA as an integrated part of the total safety analysis concept, i.e. as a complement to other relevant information such as deterministic analyses, human reliability analysis and operating experience.

Some respondents pointed out a general concern about using probabilistic risk criteria and defined safety goals as absolute limits, as this might indirectly have an impact on the quality and relevance of the PSA models. According to these respondents, the defined goals should rather be used as triggers for identifying potential deficiencies, and as indicators showing that changes made have a positive effect.

A number of the respondents express scepticism towards a strict application of quantified safety criteria, and the use of criteria does not appear to be prioritized within the over-all PSA activities of these respondents.

When it comes to the interpretation of the criteria, several of the respondents agree that more work is needed in the definition of the various criteria. Thus, there seems to be a need for a common definition as to what constitutes severe core damage and large release. A strict and common definition would facilitate comparison of risks and results between different plants.

2.8 Experience on communication of probabilistic risk criteria

Only few respondents report experiences from the communication to the public of probabilistic risk criteria and the responses varies widely between the respondents. Some respondents focus on the need for (and difficulty of) communicating very complex information, both regarding the analysis process and the definition of the risk criteria.

In those cases where safety goals are met, some respondents have found the results useful when communicating the level of safety to the public. In case the PSA results exceed the safety goals, communication would be more complex.

One experience is that public risk perception is more concerned with the consequence part of a criterion than with the frequency part, e.g., a “radioactive release” is perceived to be more easily understandable than a frequency of “1E-7 per year.” Another concern is with the complexity of the risk assessment process itself, and the ability of the general public to interpret results correctly.

If the results of PSA and safety goals should be made easier to understand to the public, it is important that it can be clearly demonstrated that PSA results and safety goals have lead to safety improvements in plants. However, the format in which PSA results and risk or safety criteria are presented needs to be carefully considered, in order to minimize the risk for misinterpretation or misunderstanding.

The U.S.NRC has developed guidelines for communicating risk information and risk decisions to the public. NUREG/BR-0308, “Effective Risk Communication, The Nuclear Regulatory Commission's Guideline for External Risk Communication” contains a comparative analysis of NRC’s risk communication needs and state-of-the-art risk communication practices.

3 Validation of subsidiary risk criteria

3.1 Background

A “subsidiary criterion” is a criterion on a lower technical level to assess in a simplified way the consequences on a higher level. Large release frequency (LRF) and large early release frequency (LERF) criteria are examples of subsidiary criteria used for risk of offsite consequences in many countries where level 3 PSA is not required. In some documents the term “surrogate criterion” is used instead.

The discussion of subsidiary criteria can be connected to two interrelated nuclear safety concepts: defence-in-depth (DID) [INSAG-10] and the international nuclear event scale (INES) [IAEA_INES]. DID calls for multiple successive methods or barriers to radioactive release to the environment (see also discussion in Chapter 4). INES is a scale for events corresponding to the safety impact of the event.

The general safety objective of a nuclear power plant is according to [INSAG-12] *to protect individuals, society and the environment by establishing and maintaining in nuclear power plants an effective defence against radiological hazard*. In the DID framework, safety objective of a nuclear power plant can be interpreted as reducing the risk of breaching all DID levels to an acceptable level. In the INES framework, it is related to the risk of events INES-4 to INES-7 (accidents). See Figure 2 for an overview of the levels of the INES scale.

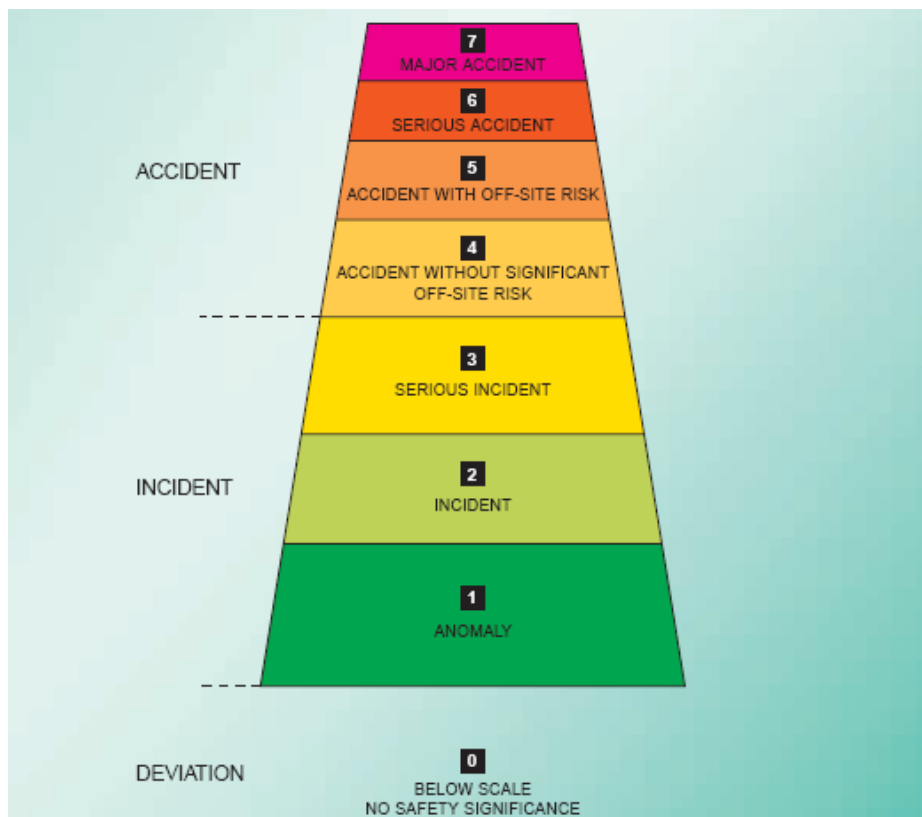


Figure 2. Overview of the INES scale [IAEA-INES-2001]

In both the frameworks, numerical criteria set to different levels are subsidiary risk criteria, the compliance of which, in principle, can be assessed by means of PSA. In practice, it depends on the scope and level of detail of the PSA model.

Figure 3 shows suggested links between PSA level 1–3 and DID levels 1–5. A safety goal framework based on INES is presented in [RESS_80(2003)143].

Initiating event Level 1 PSA		Safety functions Level 1 PSA	Safety functions Level 2 PSA	Consequence Level 3 PSA	
DID level 1 Prevention of abnormal operation and failures	DID level 2 Control of abnormal operation and detection of failures	DID level 3 Control of accidents within the design basis	DID level 4 Severe accident management	DID level 5 Mitigation of the radiological consequences	Consequence

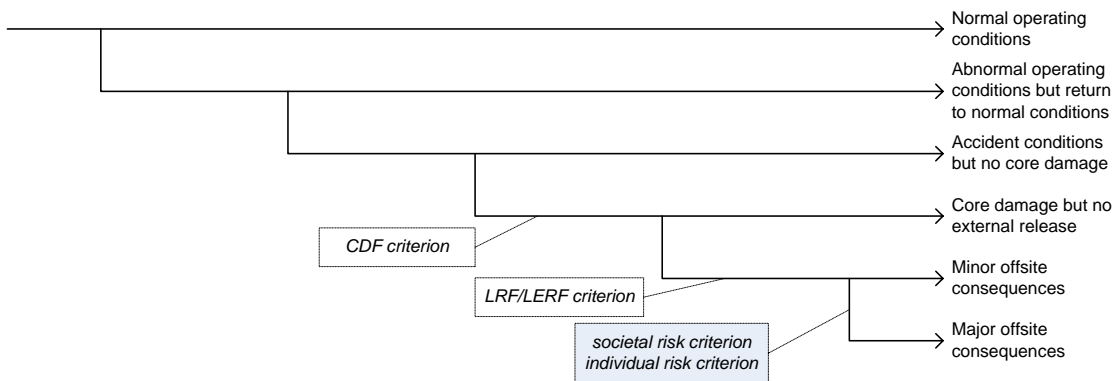


Figure 3. Simplified PSA event tree and corresponding levels of defence-in-depth (DID) linking event tree branches with different subsidiary risk criteria.

Subsidiary criteria are advocated for several reasons:

- To perform a full-scope level 3 PSA is a resource demanding effort, which can be avoided if the safety of a nuclear power plant can be demonstrated by a level 2 PSA.
- The uncertainties in the risk assessment of offsite consequences (e.g. societal and individual risk) are considerably larger than in the assessment of risk of large releases or risk of core damage. There are also fewer uncertainties in the assessment of compliance with subsidiary risk criteria.
- Subsidiary risk criteria put focus on defence-in-depth; in particular attention is paid to the accident prevention and mitigation.
- Subsidiary risk criteria can be used as a basis for the definition of safety function or system level reliability requirements, providing better support than higher level criteria to the actual design of safety functions and systems.
- Subsidiary risk criteria are closer to day-to-day operational safety management concerns of the utility, and they are closer to risk-informed applications.

The following concerns may be expressed in relation to the use of subsidiary criteria:

- The metric of different subsidiary risk criteria typically differ a lot (core damage – large release – off-site consequences), which complicates any tries to verify the assumed correspondence with higher level safety criteria.
- Technology dependency and site dependency can be difficult to take into account in subsidiary criteria.

- Subsidiary criteria (like CDF or LERF) can be difficult to compare to other risks of the society, which are typically expressed on a higher level (degree of damage to individuals or groups).
- In the communication with the public, subsidiary criteria (like CDF) may be seen as more abstract and harder to understand than top level risks (like off-site consequences).

The validity of subsidiary criteria is discussed from three perspectives: 1) validity with respect to primary safety goal for a nuclear power plant, 2) validity with respect to risk-informed applications and 3) validity with respect to capabilities of the PSA methodology.

3.2 Validity with respect to the primary safety goal for a nuclear power plant

Validity with respect to the primary safety goal for a NPP means that firstly the primary safety goal should be defined as one or more quantitative risk criteria. As an example of high level safety goals, [INSAG-12] can be used:

- To protect individuals, society and the environment by establishing and maintaining in nuclear power plants an effective defence against radiological hazard.
- To ensure in normal operation that radiation exposure within the plant and due to any release of radioactive material from the plant is as low as reasonably achievable, economic and social factors being taken into account, and below prescribed limits, and to ensure mitigation of the extent of radiation exposure due to accidents.
- To prevent with high confidence accidents in nuclear plants; to ensure that, for all accidents taken into account in the design of the plant, even those of very low probability, radiological consequences, if any, would be minor; and to ensure that the likelihood of severe accidents with serious radiological consequences is extremely small.

In order to determine the achievement of the safety goals, it is necessary to formulate quantitative risk criteria, which can be set e.g. to

- health risk
 - societal risk (consequences to the surrounding public expressed in terms of societal radiological detriment)
 - group (fatality) risk, subset of the societal risk
 - individual (fatality) risk
- environmental risk
 - restrictions in land use
 - damages to animals, cattle
- economical risk
 - cost to industry

- cost to society

An example of quantitative (intermediate) risk criteria are the quantitative objectives defined in the U.S.NRC safety goal policy [USNRC SECY-01-0009]:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes.

The first criterion is an individual risk criterion and the second one can be associated with a group mortality risk criterion. Such risk criteria can be derived by a comparison with other risks in society.

The individual risk is sometimes defined for a hypothetical most exposed person in the vicinity of a nuclear power plant, sometimes for an average person. The individual risk criterion, p^* , can be expressed like

$$p < P^*. \quad (1)$$

As a reference for the criterion, the general accidental death, which is about $1E-4$ per year, can be used. Using the factor of 100 or 1000, the safety goal for individual risk from a reactor accident should be $1E-6$ per year or $1E-7$ per year, meaning *no significant additional accident risk to an individual*.

For the group risk, references can be found e.g. from results from other risk analyses, legislation in other contexts and by comparison with radiation based cancer risk. The group mortality risk can expressed like

$$f(n) < F^*(n), \quad (2)$$

i.e., the frequency of a single accident causing n or more fatalities shall be less than $F^*(n)$. Examples for $F^*(n)$ are $1E-3/n^2$ per year used by Dutch authorities for hazardous installations, $1E-3/n$ per year, used by Australian authorities for existing dams ($1E-4/n$ for new dams) and “total risk of 100 or more fatalities”, limit $1E-5$ per year, objective $1E-7$ per year used by U.K. HSE.

To show the compliance with the individual and societal/group risk criteria a level 3 PSA should be performed. Another alternative is to derive consistent subsidiary risk criteria for the level 2 PSA. Number of fatalities should be interpreted in terms of doses, doses in terms of types of releases (source terms) and effectiveness of countermeasures. The procedure can be continued further to define criteria for level 1 PSA and for the reliability of safety functions.

It is quite evident that the use of a single criterion for level 2 PSA or level 1 PSA is a limited approach. In level 2, the use of a single frequency criterion for certain release can lead to a very strict criterion if the aim is to ensure the fulfilment of higher level criteria. On the other hand, it may be optimistic, if the criterion is only defined for an “early” release. Late releases are important for the control of societal risk.

A sufficient validity of level 2 criteria can be ensured by defining several release related criteria, as suggested e.g. in [RESS_80(2003)143], where criteria are defined for each INES-class event. Table 1 presents tentative criteria defined for a typical site in Japan, by making an interpretation of INES-classes in terms of release limits such that are coherent with the quantitative health objectives for individual and societal risk.

Table 1. Risk criteria with respect to INES classes 2 to 7 proposed in [RESS_80(2003)143].

INES-class		Release limit [TBq]			Frequency [1/yr]	
		Noble gas	Iodine	Caesium	Target	Limit
2	Incident	-30	-3	-0,3	1E-2	1E-4
3	Serious incident	-300	-30	-3	3E-3	3E-5
4	Accident mainly in installation	-3000	-300	-30	1E-3	1E-5
5	Accident with off-site risks	-30000	-3000	-300	1E-4	1E-6
6	Serious accident	-300000	-30000	-3000	1E-5	1E-7
7	Major accident	300000-	30000-	3000-	1E-6	1E-7

If the effort is put on the derivation of valid level 2 criteria, a CDF-criterion becomes irrelevant, from the health objective point of view. CDF-criterion can be used to control the defence-in-depth of the plant, e.g., reliability of DID levels 1–3. From the health objective point of view it could be worth considering plant damage state dependent criteria, at least distinguishing the containment by-pass sequence. CDF-criterion is also important for risk-informed applications, which is discussed in the next chapter.

3.3 Validity with respect to risk-informed applications

Validity with respect to risk-informed applications means validity from the decision theory point of view. It is assumed that risk (probabilities and consequences) can be assessed quantitatively and that decision maker can express preferences between different lotteries. A lottery is a decision theoretic construction defined by prizes x_1, \dots, x_n and probabilities of winning p_1, \dots, p_n , often denoted as $\{ \langle x_1, p_1 \rangle, \dots, \langle x_n, p_n \rangle \}$.

One naive interpretation of the PSA criteria, e.g., CDF* and LRF*, is a lottery $\{ \langle \text{'no accident'}, 1 - \text{CDP}^* \rangle, \langle \text{'core damage'}, \text{CDP}^* \rangle, \langle \text{'large release'}, \text{LRP}^* \rangle \}$, where CDP* and LRP* are probabilities of experiencing a core damage resp. large release during the life time of the plant and 'no accident' implies the benefits of operating an NPP for the corresponding life time. It represents a choice where the decision maker is indifferent with taking the risk or not. To use this interpretation as reference in risk-informed applications, it is necessary to quantitatively assess the cost of an accident and the benefits of operating of an NPP.

The decision theoretic approach may be criticized for several reasons. Firstly, it is difficult to capture all essential elements, especially uncertainties, affecting the decision making in a model. Secondly, people do not follow the rules of decision theory in practical decision making and have difficulties in the interpretation of probabilities. Thirdly, there are many stakeholders whose different interests should be taken into account. However, the decision theoretic framework can be used as a reference in the definition of surrogate criteria (e.g. CDF*, LRF*), and in the definition of PSA

application specific criteria. It guarantees the consistency of decision making provided that the conditions for the decision model are accepted.

The subsidiary criteria can be derived by considering an investment problem to build an NPP or not. First decision criteria are defined, such as accident risk (probability, consequences), costs (investment, operation) and income from operation. Then cost and income parameters are estimated. Accident risk (e.g. CDF*, LRF*) making the options to build or not equally preferable is the highest acceptable risk. The application specific, such as allowed outage time optimisation, can be derived analogously.

3.4 Validity with respect to capabilities of the PSA methodology

Validity with respect to capabilities of the PSA methodology means that the limitations of PSA are acknowledged. To main issues are the scope of PSA and uncertainties of PSA. The scope of PSA is always limited meaning that not all accident scenarios affecting societal and individual risk are accounted. Results of PSA include a lot of uncertainties due to several simplifications, engineering judgements, lack of statistics, and use of conservative assumptions.

Despite of limitations of PSA, ‘valid’ criteria may be defined, if there is an agreement on the role of PSA in decision making. To reach an agreement, it is necessary to define

- objectives with PSA
- requirements on PSA
- applications of PSA
- how PSA criteria and safety goals are used in decision making.

While in the previous validity considerations (with respect to overall safety goals and with respect to risk-informed applications) the approach to define valid subsidiary criteria is top-down, here the approach is bottom-up. Based on experience from present PSAs CDF and L(E)RF for different reactors can be used as references. In this consideration, it is important to know the scope and limitations of the studies. It is also important to look at the contributing factors for the numeric results, and compare the risk information with the conception of the safety of a plant.

In fact, the CDF and L(E)RF criteria used in many countries and e.g. proposed by IAEA, have their basis on the experience with PSAs. $CDF^* = 1E-5$ per year is generally regarded as an achievable target for a well designed plant. Regarding large release, the issue is more open due to varying and vague definitions for large release.

3.5 Summary

Safety goals typically express primary objectives in a qualitative sense. In order to make them fit for practical use and application, the safety goals must be translated into quantitative risk criteria such as societal risk and individual risk. Sometimes these quantitative risk criteria are called intermediate criteria, since they need to be further translated into numeric criteria for the interpretation of results from a PSA study. These criteria are called subsidiary or surrogate criteria, at least when used for level 1 and 2 PSA.

There are several aspects in the validation of subsidiary risk criteria, such as validity with respect to definition of risk, validity with respect to rational decision making under risk and validity with respect to use of PSA. Taking into account the several aspects means a combination of top-down and bottom-up approaches in the derivation of subsidiary risk criteria. For the top down approach, several references exist for societal and individual level risk criteria to be used as the basis. The decision theory provides the framework for the definition of rational risk criteria. For the bottom-up approach, it is necessary to define the objectives with PSA and how safety goals are used. Also experience from present PSAs are valuable in this process.

4 Numerical criteria when using probabilistic analyses in support of deterministic safety analysis

4.1 Overview of potential areas of interaction between deterministic and probabilistic analysis

The issue of existing and potentially relevant connections between deterministic and probabilistic analysis was touched on in a general manner at a Technical Meeting arranged by the IAEA in 2006 [IAEA_TM2006], dealing with the possibility to combine Deterministic Safety Analysis (DSA) and PSA in the plant safety management. A number of issues were discussed and explored. The focus was on providing a broad status description based on input from the participating countries and to make suggestions for further work that could improve the possibilities of interaction between DSA and PSA. Therefore, the meeting procedures provide a broad overview of areas where integration of DSA and PSA might be possible and beneficial. The main areas of discussion are listed and shortly outlined below:

- Insights from the DSA that can be used in PSA
 - Determination of the set of initiating events.
 - Analysis of success criteria for safety systems and accident progression following an initiating event.
 - Analysis of internal and external hazards.
 - Severe accident phenomena, containment performance and source term calculations for level 2 PSA.
 - High level requirements that relate to defence in depth and provision of adequate safety margins.
 - Lower level requirements for safety systems, e.g., regarding redundancy, diversity, separation/segregation, fail safe actuation, and equipment qualification.
- Insights from the PSA that can be used in DSA
 - Examples of possible areas are, e.g., maintenance optimisation, design support, assessment of impact of model and data uncertainties, and Technical Specifications development.
- Relationship between the deterministic analysis and PSA in addressing defence in depth
 - A variety of PSA parameters and output might be used as tools for assessing the strength of the different DID levels, see Figure 4. (The DID levels have already been described in Figure 3.)
 - This issue is analysed in detail in two Swedish projects initiated by the Swedish Radiation Protection Authority (SSM).

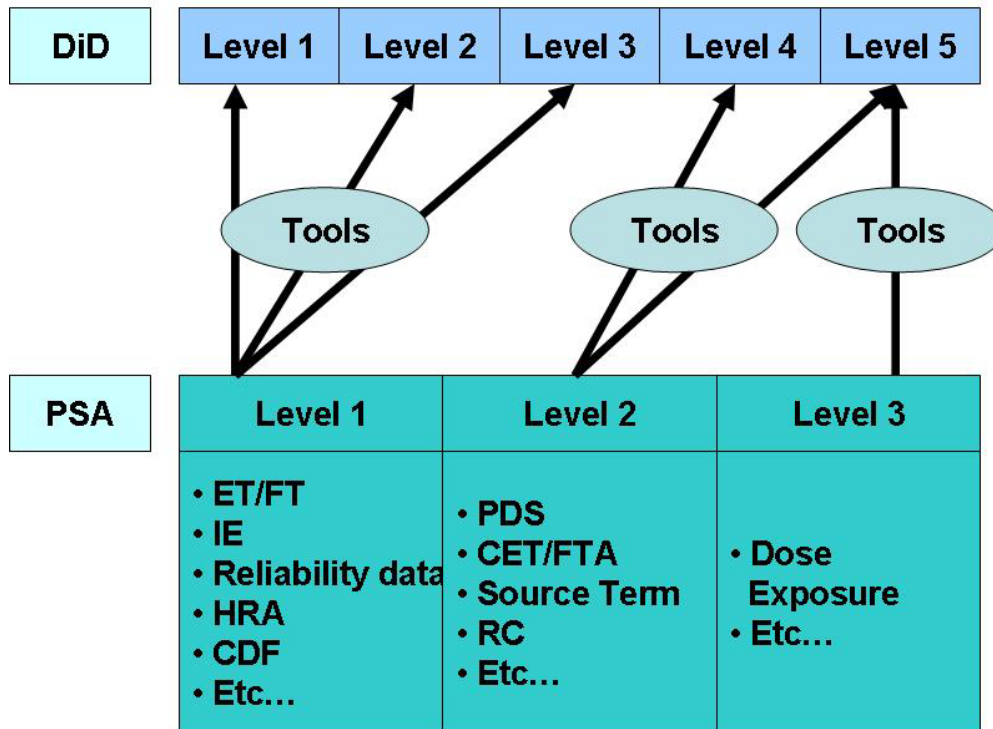


Figure 4. Overview of the relation between PSA and the levels of defence in depth

4.2 Assessment of defence in depth using probabilistic methods

The Swedish Radiation Protection Authority regulations concerning Safety in Nuclear Facilities [SSMFS 2008:1] stresses the importance of systematic implementation and verification of defence in depth in nuclear facilities. It also requires that in addition to deterministic analyses, the plant shall be analysed using probabilistic methods in order to obtain as comprehensive a view as possible of safety. This is the background to the two project described in this chapter.

4.2.1 Assessment of defence in depth using probabilistic methods

There are a number of risk-informed applications where parts of the defence in depth are analysed and risk assessed with PSA – this is in fact one of the basic aims of PSA. PSA results can generally be seen as an assessment of the overall safety of a plant, giving information about the capability of the plant as such and of its various safety functions to handle various types of disturbances, both relatively frequent ones and disturbances that are expected to occur extremely infrequently. However, there is at this time no explicit connection between PSA and the various levels of defence in depth as defined in [SSMFS 2008:1] and [IAEA_INSAG-10].

An SSM project with the title “Assessment of defence in depth using probabilistic methods” is on-going, and will be finalised during 2009. The expected outcome is in part a systematic mapping of conditions related to the various levels of defence in depth, and the suggestion of quantitative measures (if any) that can be associated with each identified condition. A second part of the expected outcome is the connection of the suggested quantitative measures with PSA, i.e., the generation of a set of methods for using PSA models and results in a way that allows assessment and ranking of the

structures, systems, components (SSC) and operating procedures which form part of the defence in depth of a nuclear power plant. The project is described in [PSAM9-0336].

The research project is divided into five phases:

1. Survey of which qualitative parameters of each level of defence in depth that should be considered in the method. This includes identification and structuring of the SSCs that belong to each DID level and that should thus be considered for potential PSA evaluation.
2. Survey of different quantitative parameters of each level of defence in depth based on 1).
3. Development of methods and models and need for adjustment of current PSA models to evaluate each of the quantitative parameters defined in 2).
4. Quantitative analyses including trial applications.
5. Analyses for each of the quantitative parameters including qualitative and quantitative safety evaluation of the results.

Within the project, a systematic and focused analysis has been performed of the connections between the levels of defence in depth, and the risk measures utilised in existing applications in order to make efficient use of available information on risks. This review will lead to the definition of new risk measures that may be used in the risk assessment procedure, and which may be of use in assessing the safety level of a plant, evaluation of occurred events with safety impact, and evaluation of proposed plant changes, including changes in SAR or Technical Specifications.

The basis for the evaluation will be current PSA studies for Swedish BWR and PWR plants including planned further work for these. Methods suggested should thus not imply a need for unreasonable modifications of the studies. The first two phases of the project have been completed and included a literature survey, description and definition of the DID levels, a preliminary DID-PSA discussion and development of a proposal and survey of qualitative parameters to be used for evaluation of defence in depth.

4.2.2 Mapping of conditions in the defence in depth levels 1 and 2 against LOCA

The second subproject of the SSM project focused on the conditions that should be considered when analyzing the DID level 1 and level 2 and to define quantitative measures for these conditions [SKI 2008:33]. The DID level 3 to 4 are quite well handled in today's PSA-studies and the DID level 5 is related the level 3 PSA, which is not a requirement in many countries. Meanwhile in the DID levels 1 and 2 there are a large number of activities which not necessarily are explicitly modelled in PSA-studies but that may be of interest from a risk assessment point of view. Many PSA-studies practically use a condition where a reactor scram should be actuated as the definition for an initiating event during power operation. This definition omits the significance of the DID levels 1 and 2 for reactor safety.

In order to effectively study and demonstrate the idea of risk-informed assessment of DID, the work was limited to loss-of-coolant-accidents (LOCA). The means to DID can be considered from several perspectives. In the subproject, not only the system barrier perspective was considered but also the plant lifetime cycle, as presented in Figure 5.

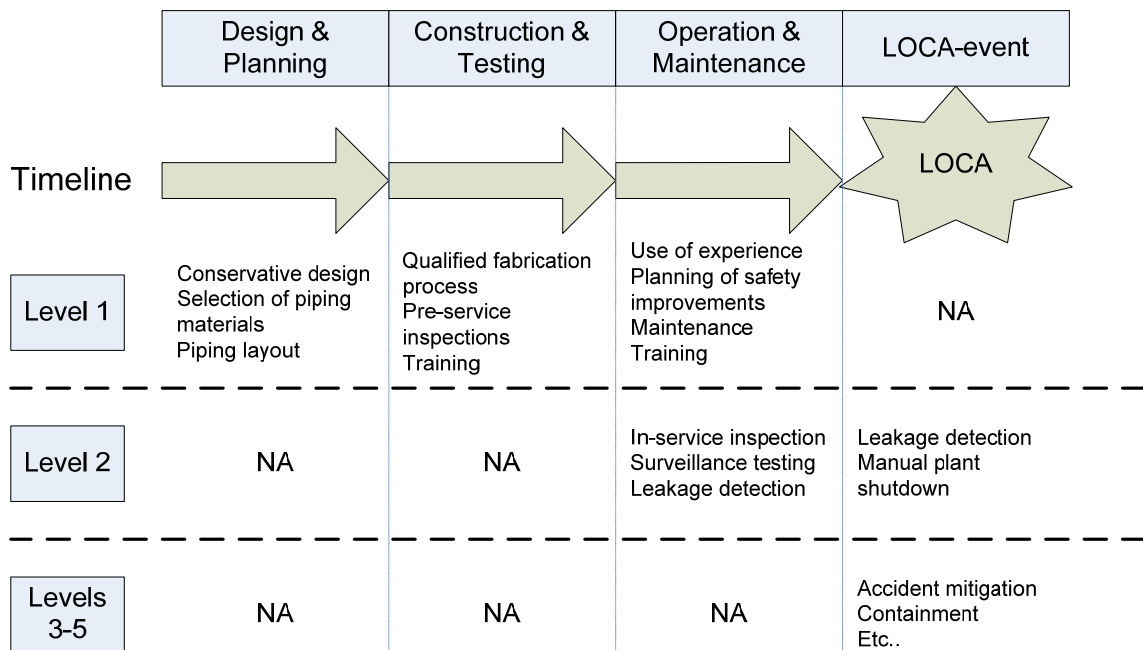


Figure 5. Means for defence-in-depth against LOCA during different phases of the nuclear power plant's lifetime [SKI 2008:33]

Means to DID in level 1 and 2 against LOCA were identified for different types of LOCA (both during power operation and during shutdown period). Many of those means are not considered explicitly in present PSAs, such as effectiveness of in-service-inspection and the reliability of leak detection methods. To expand the PSA-model by a more detailed modelling of the DID levels 1 and 2 is seen fully realistic, and this would facilitate the quantification of the risk importance of the DID level 1 and 2 activities.

5 Guidance for the definition and use of probabilistic safety criteria

5.1 Introduction

During the final project phase (Phase 4, 2009), one of the aims is to prepare a guidance document. The document aims at summing up, on the basis of the work performed throughout the previous three project phases, issues to consider when formulating, applying and interpreting probabilistic safety criteria. This includes issues like:

- Use of safety goals in a strict manner (limiting values) vs. as targets (orientation values).
- Treatment of uncertainties in the application of safety goals.
- Effect of the scope of the safety goal.
- Needs to internationally harmonize probabilistic safety criteria for NPPs.
- Qualification of PSA for the application of probabilistic safety criteria.
- Differences/similarities in the use of safety goals for new and operating NPPs.

5.2 Workshop results

In order to involve Nordic utilities and authorities in the planning of this important sub-task, a workshop was held at the project seminar in December 2008 [SG_Semin_2008]. The workshop discussions were held along two lines:

- Guidance related to the definition of a safety goal or criterion
- Guidance related to the application and interpretation of a safety goal or criterion

Workshop participants represented CNSC-CCSN (Canada); ES-konsult; FKA, (Forsmark NPP), Fortum Nuclear Services (Loviisa NPP), JAEA (Japan), NKS, Relcon Scandpower, Ringhals AB (Ringhals NPP), The Swedish Radiation Protection Authority (SSM), TVO (Olkiluoto NPP), Vattenfall Generation Nordic, and the Technical Research Centre of Finland (VTT).

At the workshop, two groups of about 10 people each were formed, each with a chairman and a secretary:

- Group 1 "Definition of Safety Goals"
 - Chair: Lars Gunsell, SSM
 - Secretary: Michael Knochenhauer
- Group 2 "Application and Interpretation of Safety Goals"
 - Chair: Risto Himanen, TVO
 - Secretary: Jan-Erik Holmberg

Table 2 presents the items discussed at the workshop. As seen from the table, some issues were considered relevant both for the definition and for the application and interpretation of safety goals.

Table 2. Overview of issues discussed at the Safety Goals Workshop

Discussion items	Definition of safety goal or criterion	Application and interpretation of safety goal or criterion
1. What should be the scope of safety goals? <ul style="list-style-type: none"> • sources of radioactive release • classes of initiating events • operational states • life cycle phases 	X	
2. What definition of risk should we use as primary goal? <ul style="list-style-type: none"> • Risk to the environment/offsite population • Risk to the plant, e.g., to the core (level 1 PSA) or to the containment (level 2 PSA) 	X	
3. Should safety goals be comparable to safety goals for other industries?	X	X
4. Should safety goals be site-dependent or reactor size dependent?	X	
5. Should safety goals depend on number of reactors at site?	X	
6. On what levels should safety goals be defined? <ul style="list-style-type: none"> • Off-site consequences • Release • Core damage 	X	X
7. Shall the same or different goals apply to <ul style="list-style-type: none"> • new vs. operating plants? • licensing of new plant vs. regulation of same plant during operation? 	X	
8. Should safety goals be harmonised (definition of common goals)?	X	
9. Who should define safety goals?	X	
10. Use of criteria as limits vs. target		X
11. Handling of violations of safety goals		X
12. Treatment of uncertainties		X
13. Communication of the safety goals and PSA results		X
14. What kinds of applications of safety goals can be foreseen?		X

Attachment 2 summarises the outcome of the workshop. This material will be used as part of the input for the guidance document to be developed during the concluding project phase.

6 Conclusions

6.1 Outcome of project phase 3

The first phase of the project (2006) described the status, concepts and history of probabilistic safety goals for nuclear power plants. The second and third phases (2007–2008) have provided guidance related to the resolution of some of the problems identified, and resulted in a common understanding regarding the definition of safety goals.

The basic aim of phase 3 has been to increase the scope and level of detail of the project, and to start preparations of a guidance document. Based on the conclusions from the previous project phases, the following issues covered:

- Extension of international overview. Analysis of results from the questionnaire performed within the ongoing OECD/NEA WGRISK activity on probabilistic safety criteria, including participation in the preparation of the working report for OECD/NEA/WGRISK (to be finalised in phase 4).
- Use of subsidiary criteria and relations between these (to be finalised in phase 4).
- Numerical criteria when using probabilistic analyses in support of deterministic safety analysis (to be finalised in phase 4).
- Guidance for the formulation, application and interpretation of probabilistic safety criteria (to be finalised in phase 4)

Some conclusions from these activities are summarised below.

Overview of international safety goals

There are considerable differences in the status of the numerical risk criteria that have been defined in different countries. Some have been defined in law or regulations and are mandatory, some have been defined by the regulatory authority or by an authoritative body, and some have been defined by plant operators or designers. Hence the status of the criteria ranges from mandatory requirements to informal.

Three levels of risk criteria exist, i.e., at society level (mainly qualitative), at an intermediate level (quantitative and/or qualitative) and at a technical level (quantitative). The higher level criteria are typically concerned with the actual risk to society or individuals, or to the environment. Technical criteria are generally quantitative (probabilistic) and mostly on lower levels (subsidiary). They typically concern core damage, unacceptable release, and unacceptable health risks. In later years, some countries have defined separate criteria to address robustness in defence in depth, e.g., by having a separate criterion for containment integrity.

In most countries that have criteria both for existing and new reactors, the criteria are more strict for new plants. In some cases this is expressed by using the same numerical values for the frequencies, but applying them as limits for new plants and targets for existing plants.

Regarding consideration of uncertainties, there is consensus that the comparison with probabilistic safety criteria should use the “best estimate” of the PSA results.

In most cases, risk criteria for operating plants are applied when the PSA is updated, which results in a very large spread, as a considerable part of the respondents only

update the PSA within the framework of the periodic safety review (generally 10 years), while others do updates on a yearly basis. Risk criteria are also used to assess the impact on risk of design modifications in the plant.

Risk criteria are mostly considered as indicators or orientation values, meaning that no regulatory actions are expected on non-compliance with a probabilistic safety criterion. Practically, there is a consensus on finding the reasons for the non-compliance and identification on the way to overcome it. However, for new builds application of risk criteria would be stricter.

When it comes to the interpretation of the criteria, several of the respondents agree that more work is needed in the definition of the various criteria. Thus, there seems to be a need for a common definition as to what constitutes severe core damage and large release. A strict and common definition would facilitate comparison of risks and results between different plants.

The general experience from the implementation of risk criteria is positive. Respondents who have implemented criteria have experienced various benefits. In a number of cases, design weaknesses or procedural weaknesses in NPPs have been identified using PSA and PSA criteria, resulting in the introduction of safety improvements. In many cases, the implementation of risk criteria and safety goals has led to plant modifications in order to meet the probabilistic risk criteria. The implementation of safety goals often emphasizes the need for more detailed and realistic PSA models, and it appears that the use of safety goals has increased the focus on the correctness and quality of PSA models.

Use of subsidiary criteria and relations between these

Goals related to CDF and LRF are surrogates to societal and individual risk level criteria. To fully validate these goals, calculations of environmental consequences of release sequences would need to be made. In a few countries, the performance of level 3 PSAs is required, which would enable a direct evaluation of the compliance against societal and individual risk level criteria.

There are several aspects in the validation of subsidiary risk criteria, such as validity with respect to the definition of risk, validity with respect to rational decision making under risk, and validity with respect to the use of PSA. Taking these aspects into account means a combination of top-down and bottom-up approaches in the derivation of subsidiary risk criteria. For the top down approach, several references exist for societal and individual level risk criteria, and can be used as a basis, with decision theory providing the framework for the definition of rational risk criteria. For the bottom-up approach, it is necessary to define the objectives with PSA and how safety goals are used. Also experience from present PSAs are valuable in this process.

Numerical criteria when using probabilistic analyses in support of deterministic safety analysis

This sub-task mainly monitors the outcome from other on-going work regarding the relation between PSA and defence in depth.

Guidance for the formulation, application and interpretation of probabilistic safety criteria

This issue is an important part of the final project phase (2009). At the project seminar in December 2008, representatives from Nordic utilities and authorities participated in a workshop with discussions held along two lines, i.e., guidance related to either the

definition or to the application and interpretation safety goals or criteria. Valuable input was received, and will be used as part of the basis for the guidance document to be developed during the concluding project phase.

6.2 Continued work during phase 4

During phase 4, the international overview (OECD/NEA WGRISK task on safety criteria) will be finalised, and results and conclusions from this project will be considered. In addition, focus will specifically be put on questions related to application and communication. Guidance to the definition of valid subsidiary criteria will be developed, i.e., lower level criteria that are indirectly related to the societal and individual risk criteria but that are directly applicable with present PSAs. Finally, some additional aspects related to the issue of consistency over time in the usage of safety goals will be explored.

Each of these sub-activities is shortly described below.

Participation in the finalization of the OECD/NEA WGRISK task on use of probabilistic criteria

This sub-activity consists in participating and contributing to the OECD/NEA WGRISK task on use of probabilistic criteria. The project was initiated in connection with the OECD/NEA WGRISK meeting in April 2007. A task group was formed, including representatives from about 10 countries, and a questionnaire was designed, partly based on the questionnaire used in the first phase of the NKS project.

The current status is that responses to the questionnaire have are being received and analysed and the preparation of the task report is in progress. There will be a task group meeting in early 2009 for the finalisation of the report which will be presented at the annual meeting of WGRISK in March 2009, and finalised thereafter.

Consistency of in the usage of safety goals (continuation)

This issue was dealt with in phase 2, but will be addressed once more from a slightly different angle, including consideration of issues related to the quality and scope of a PSA, and the question of whether PSA safety goals can be allowed to vary over time. In addition, it will be attempted to derive some more details from the comparison between three generations of the Forsmark 1 PSA which was initiated during phase 2.

Definition of valid subsidiary criteria

In the first three phases, issues related to connections between different levels of criteria have been encountered and discussed on a basic level. In this phase, it will be attempted to provide guidance on definition of valid subsidiary criteria especially at the levels of core damage risk (for level 1 PSA) and large release risk (for level 2 PSA). This is based on e.g.:

- Results from WGRISK task, which compiles information from the use of safety criteria on all levels from societal risk to core damage risk.
- On-going investigations in Finland on relations between acceptance criteria for level 2 PSA (large release) and level 3 PSA (off-site consequences).
- Relevant conclusions from on-going SKI projects on the relation between defence in depth and PSA.

- Use of conditional probability criteria (barrier analysis) for judging the acceptability of the safety level for initiators which have major uncertainties in their frequency of occurrence.

Guidance for the application and communication of the probabilistic safety criteria

This sub-activity aims at summing up, on the basis of the work performed throughout the three project phases, issues to consider when formulating, applying, and interpreting probabilistic safety criteria. This includes questions like:

- Use of safety goals in a strict manner (limiting values) vs. as targets (orientation values).
- Treatment of uncertainties in the application of safety goals.
- Effect of the scope of the safety goal.
- Needs to internationally harmonize probabilistic safety criteria for NPPs.
- Qualification of PSA for the application of probabilistic safety criteria.
- Differences/similarities in the use of safety goals for new and operating NPPs.

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Attachment 1. Safety goals and PSA risk criteria defined by nuclear safety authorities

Country	Safety goals	PSA risk criteria
<p>Canada Canadian Nuclear Safety Commission</p>	<p>(i) Prevent unreasonable risk, to the environment and to the health and safety of persons, associated with that development, production, possession or use, (ii) Prevent unreasonable risk to national security associated with that development, production, possession or use, [Nuclear Safety and Control Act]</p> <p>i) Individual members of the public shall be provided a level of protection from the consequences of nuclear power plant operation such that there is no significant additional risk to the life and health of individuals, and ii) Societal risks to life and health from nuclear power plant operation shall be comparable to or less than the risks of generating electricity by viable competing technologies, and should not be a significant addition to other societal risks. [Regulatory Document RD-337]</p>	<p>i) Small Release Frequency, The sum of frequencies of all event sequences that can lead to release to the environment of more than 10^{15} Bq of I-131 should not exceed $1E-5$ per plant year. ii) Large Release Frequency The sum of frequencies of all event sequences that can lead to release to the environment of more than 10^{14} Bq of Cs-137 should not exceed $1E-6$ per plant year. iii) Core Damage Frequency The sum of frequencies of all sequences that can lead to significant core degradation should not exceed $1E-5$ per plant year. [Regulatory Document RD-337]</p>
<p>Finland Radiation and Nuclear Safety Authority</p>	<p>The general objective is to ensure nuclear power plant safety so that nuclear power plant operation does not cause radiation hazards which could endanger safety of workers or population in the vicinity or could otherwise harm the environment or property.</p> <p>The limit for the dose commitment of the individual of the population, arising from normal operation of a nuclear power plant in any period of one year, is 0.1 mSv. Based on this limit, release limits for radioactive materials during the normal operation of a nuclear power plant are to be defined.</p> <p>The limit for the dose of the individual of the population, arising, as the result of an anticipated operational transient, from external radiation in the period of one year and the simultaneous radioactive materials intake, is 0.1 mSv.</p> <p>The limit for the dose of the individual of the population, arising, as the result of a postulated accident, from external radiation in the period of one year and the simultaneous radioactive materials intake, is 5 mSv.</p> <p>The limit for the release of radioactive materials arising from a severe accident is a release which causes neither acute harmful health effects to the population in the vicinity of the nuclear power plant nor any long-term restrictions on the use of extensive areas of land and water. For satisfying the requirement applied to long-term effects, the limit for an atmospheric release of cesium-137 is 100 TBq. The combined fall-out consisting of nuclides other than cesium-isotopes shall not cause, in the long term, starting three months from the accident, a hazard greater than would arise from a cesium release corresponding to the above-mentioned limit. The possibility that, as the result of a severe accident, the above mentioned requirement is not met, shall be extremely small.</p> <p>[Decision of the Council of State (395/1991)]</p> <p>The safety level of a nuclear power plant shall be raised as high as practicable to achieve the objectives presented in section 6 of the Nuclear Energy Act and in section 3 of the Council of State Decision (395/1991). The more severe an accident's consequences to man, the environment and property could be, the smaller the likelihood of its occurrence shall be.</p> <p>[Guide YVL 1.0, Ch. 3]</p>	<p>The following numerical design objectives cover the whole nuclear power plant:</p> <ul style="list-style-type: none"> - The mean value of the probability of core damage is less than $1E-5/a$. - The mean value of the probability of a release exceeding the target value defined in section 12 of the Government Resolution (359/1991) must be smaller than $5E-7/a$. <p>Should substantial risk factors not recognised earlier appear during operation, the licensee shall upgrade the safety of the plant.</p> <p>In conjunction with the design of safety upgrades the licensee shall demonstrate that the safety of the plant assessed after the upgrades is substantially at the same level or better than the objectives presupposed for the design phase.</p> <p>[Guide YVL 2.8, Ch. 2.1]</p>

Country	Safety goals	PSA risk criteria
<p>Hungary Hungarian Atomic Energy Authority</p>	<p>It is a general nuclear safety objective that the protection of individuals and groups of the population as well as that of the environment has to be in place against the dangers of ionising radiation. This has to be ensured by effective protection and its appropriate level maintenance within the nuclear power plant. [Vol. 3 of the Nuclear Safety Codes issued by the Hungarian Governmental Decree No. 89/2005 in paragraph 2.002]</p> <p>It is a radiation protection objective that the exposure of the operating personnel and the population during the operation of the nuclear power plant has to be kept under the prescribed limit, and at the reasonably achievable lowest level. This has to be ensured in cases of exposure during design malfunctions (Anticipated Operational Occurrence and Design Basis Accidents) and the exposure has to be reduced to a reasonably possible extent during severe operational accidents (Beyond Design Basis Accidents). [Vol. 3 of the Nuclear Safety Codes issued by the Hungarian Governmental Decree No. 89/2005 in paragraph 2.003]</p> <p>It is a technical safety objective that operational incidents have to be prevented to a reasonable extent, the possible consequences considered in the design phase of the facility as anticipated initiating event have to be within the prescribed limit and that the probability of accidents has to be reasonably low. [Vol. 3 of the Nuclear Safety Codes issued by the Hungarian Governmental Decree No. 89/2005 in paragraph 2.004]</p>	<p>During the probabilistic safety assessment of the nuclear power plant design it has to be an objective that the core damage frequency coming from the level 1 PSA taking into account all anticipated initiating events and design malfunction, as an annual average should not be higher than 1E-5 per year, and in any planned operating condition of the nuclear power plant, within the lifecycle of the operations the core damage frequency should not exceed the 5E-4 per year average value. [Volume 3 of the Nuclear Safety Codes in paragraph 3.072]</p>
<p>Japan The Japanese Nuclear Safety Commission</p>	<p>The likelihood of occurrence of health detriment to the public due to emission of radiation or release of radioactive materials from activities for nuclear energy utilization should be controlled to such a level that members of the public bear no significant additional risk to their daily life.</p> <p>The average risk of early fatality for members of the public in the vicinity of the site boundary of a nuclear facility due to radiation exposure from nuclear accidents should not exceed approximately one in 1000000 a year.</p> <p>The average risk of cancer fatality for members of the public within a certain distance from a nuclear facility due to radiation exposure from nuclear accidents should not exceed approximately one in 1000000 a year. [NSC_2006]</p>	<p>Core Damage Frequency (CDF): approximately 1E-4 per reactor year Containment Failure Frequency (CFF): approximately 1E-5 per reactor year</p> <p>Both of the two goals are required to be met at the same time for all events including internal and external initiating events. [NSC_2006]</p>
<p>Korea The Korean Nuclear Safety Commission</p>	<p>The main objectives of the policy on severe accident are to assure that the possibility of a severe accident occurrence is extremely low and its risk to the public is sufficiently reduced.</p> <p>The prompt fatality risk resulting from the accidents to an average individual in the vicinity of a NPP should not exceed one-tenth of one percent of the sum of those risks resulting from other accidents which members of the population might generally be encountered.</p> <p>The cancer fatality risk resulting from nuclear power plant operation to the population in the area near a NPP should not exceed one-tenth of one percent of the sum of cancer fatality risks resulting from all other causes. [KNSC_2001]</p>	<p>The performance goals (Core Damage Frequency, Large Containment Release Frequency) should be established in near future, and so far the official Probabilistic Risk Criteria does not exist in Korea.</p> <p>Tentative criteria are:</p> <ul style="list-style-type: none"> - CDF for existing plants and life extension : less than 1E-4 per reactor year - CDF for new plants : less than 1E-5 per reactor year - LERF for existing plants and life extension, : less than 1E-5 per reactor year - LERF for new plants : less than 1E-6 per reactor year
<p>Slovakia Nuclear Regulatory Authority of the Slovak Republic</p>	<p>Protect the public and the environment from unreasonable risk.</p>	<p>For existing plants. Criteria for the new plant are lower by one order of magnitude</p> <p><u>Large early release</u>: Significant, or large release is defined through the release of Cs -137. Early release is the release of fission products before applying the offside protective measures.</p> <p>Target $f < 1E-5$ per year [BNS I.4.2/2006]</p>

Country	Safety goals	PSA risk criteria
<p>Sweden SSM (previous SKI)</p>	<p>The focus of the SKI is on avoidance of radiological accidents, i.e., safety goals are directed towards protection of the public rather than towards avoidance of core damage.</p> <p>Long-term ground contamination of large areas shall be avoided. This is judged to be fulfilled if the radioactive release after a severe accident is limited to below 0,1 % of the inventory of the caesium isotopes Cs-134 and Cs-137 in a core of 1800 MW, excluding noble gases.</p> <p>There shall be no short-term fatalities in acute radiation syndrome. This is judged to be fulfilled if the radioactive release after a severe accident is limited to below 1 % of the inventory of a core of 1800 MW, excluding noble gases.</p> <p>The radioactive release after a severe accident is limited to below 0,1 % of the inventory of the cesium isotopes Cs-134 and Cs-137 in a core of 1800 MW, excluding noble gases.</p> <p>The radioactive release after a severe accident is limited to below 1 % of the inventory of a core of 1800 MW, excluding noble gases .</p> <p>Release of more than 0,1 % of the inventory of Cs-134 and Cs-137 in a core of 1800 MWt shall be "extremely unlikely" (Interpreted as < 1E-7 per year).</p> <p>The containment shall remain intact for 10-15 hours after a core melt. This requirement implies that the core that mitigating measures protecting the containment from over-pressurisation and by-pass shall be designed in a way that practically eliminates the possibility of early releases.</p> <p>A number of acceptance criteria for the mitigating systems after a severe accident are defined: Events with extremely low probabilities (extremt låga sannolikheter) can be neglected. It is accepted that the filtered venting system cannot handle a reactor vessel rupture.</p> <p>[SKI_SSI_1985]</p>	<p>"Extremely unlikely" interpreted as 1E-5 per year</p> <p>Release of more than 0,1 % of the inventory of the caesium isotopes Cs-134 and Cs-137 in a core of 1800 MWt shall be "extremely unlikely" (Interpreted as < 1E-7 per year).</p> <p>[SKI_SSI_1985]</p>
<p>Switzerland HSK</p>	<p>General qualitative requirements on the safety level are expressed by the term safety and not by risk. Risk is considered only as one element of the safety. An overall qualitative safety requirement is that in the utilisation of nuclear energy, human beings and the environment must be protected against harm due to ionising radiation.</p> <p>The nuclear energy law requires that sufficient preventive and mitigative measures shall be considered in order to ensure the safety of nuclear power plants in Switzerland. In order to demonstrate that sufficient measures have been taken, the accidents are categorized according to their frequencies. Dose limits are defined for accidents with frequencies larger than 1E-6 per year.</p>	<p>The legal basis for the implementation of PSA in the regulatory safety oversight process is defined in the nuclear energy law and an accompanying nuclear energy ordinance in Switzerland. The ordinance stipulates that for the construction permit of a new nuclear power plant, the applicant is required to demonstrate that the core damage frequency is below 1E-5 per year. This risk criterion is also expected to be fulfilled by the existing plants, to the extent that is reasonably achievable. Risk criteria for assessment of operational events and determination of safety significance of active components are under discussion.</p>

Country	Safety goals	PSA risk criteria																				
<p>UK HMI</p>	<p>Protection must be optimized to provide the highest level of safety that is reasonably practicable.</p> <p>Limitation on risks to individuals: “Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm”</p> <p>Prevention of accidents: “All reasonably practicable steps must be taken to prevent and mitigate nuclear or radiation accidents”</p> <p>Protection of present and future generations: “People, present and future, must be protected against radiation risks” HSE’s SAPs (2006 Edition) (paragraph 42)</p>	<p>Target 5: Individual risk of death from on-site accidents – any person on the site, and</p> <p>Target 7: Individual risk to people off the site from accidents - Limit 1E-4 per year - Objective 1E-6 per year</p> <p>Target 6: Frequency dose targets for any single accident – any person on the site, and</p> <p>Target 8: Frequency dose targets for accidents on an individual facility – any person off the site</p> <table border="1" data-bbox="1464 480 1960 670"> <thead> <tr> <th>On site, mSv</th> <th>Off-site, mSv</th> <th>Limit</th> <th>Objective</th> </tr> </thead> <tbody> <tr> <td>2–20</td> <td>0,1–1</td> <td>1E-1</td> <td>1E-2</td> </tr> <tr> <td>20–200</td> <td>1–10</td> <td>1E-2</td> <td>1E-4</td> </tr> <tr> <td>200–2000</td> <td>10–100</td> <td>1E-3</td> <td>1E-5</td> </tr> <tr> <td>>2000</td> <td>>100</td> <td>1E-4</td> <td>1E-6</td> </tr> </tbody> </table> <p>Target 9: Total risk of 100 or more fatalities - Limit 1E-5 per year - Objective 1E-7 per year</p> <p>The targets are not mandatory but, rather, they are guides to inspectors to indicate where there is the need for consideration of additional safety measures.</p>	On site, mSv	Off-site, mSv	Limit	Objective	2–20	0,1–1	1E-1	1E-2	20–200	1–10	1E-2	1E-4	200–2000	10–100	1E-3	1E-5	>2000	>100	1E-4	1E-6
On site, mSv	Off-site, mSv	Limit	Objective																			
2–20	0,1–1	1E-1	1E-2																			
20–200	1–10	1E-2	1E-4																			
200–2000	10–100	1E-3	1E-5																			
>2000	>100	1E-4	1E-6																			
<p>USA U.S.NRC</p>	<p>Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.</p> <p>Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.</p> <p>The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.</p> <p>The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes.</p> <p>[NRC’s Severe Accident Policy Statement, 1986] [USNRC SECY-01-0009]</p>	<p><u>NRC. Safety Goal Policy Statement</u></p> <p>Although not part of the Safety Goals, the NRC established measures for core damage frequency (CDF) and large early release frequency (LERF) that are widely used to evaluate the safety of operating nuclear power plants. The CDF measure is 1E-4 and the LERF measure is 1E-5. Using the vast body of severe accident progression and PRA research that has been performed for current LWRs, it has been calculated that satisfying these measures will almost certainly satisfy the Safety Goals.</p> <p>[Appendix D to NUREG-1860]</p>																				

Country	Safety goals	PSA risk criteria
<p>IAEA International Atomic Energy Agency</p>	<p>To protect individuals, society and the environment by establishing and maintaining in nuclear power plants an effective defence against radiological hazard.</p> <p>To ensure in normal operation that radiation exposure within the plant and due to any release of radioactive material from the plant is as low as reasonably achievable, economic and social factors being taken into account, and below prescribed limits, and to ensure mitigation of the extent of radiation exposure due to accidents.</p> <p>To prevent with high confidence accidents in nuclear plants; to ensure that, for all accidents taken into account in the design of the plant, even those of very low probability, radiological consequences, if any, would be minor; and to ensure that the likelihood of severe accidents with serious radiological consequences is extremely small.</p> <p>[IAEA_INSAG-12]</p>	<p>The target for existing nuclear power plants consistent with the technical safety objective is a frequency of occurrence of severe core damage that is below about 10–4 events per plant operating year. Severe accident management and mitigation measures could reduce by a factor of at least ten the probability of large off-site releases requiring short term off-site response. Application of all safety principles and the objectives of para. 25 to future plants could lead to the achievement of an improved goal of not more than 10–5 severe core damage events per plant operating year. Another objective for these future plants is the practical elimination of accident sequences that could lead to large early radioactive releases, whereas severe accidents that could imply late containment failure would be considered in the design process with realistic assumptions and best estimate analyses so that their consequences would necessitate only protective measures limited in area and in time.</p> <p>[IAEA_INSAG-12]</p>
<p>EUR European Utility Requirements</p>	<p>The general objective of nuclear safety is to protect individuals, society and the environment by establishing and maintaining an effective defence against radiological hazards. radiological consequences, if any, would be minor.</p> <p>To ensure that in normal operation, radiation exposure within the plant and radiation doses due to any release of radioactive material from the plant are kept As Low As Reasonably Achievable (ALARA) and below prescribed limits.</p> <p>To ensure that, for all accidents addressed in the design of the plant, radiological consequences, if any, would be minor.</p>	<p>A Core Damage cumulative frequency of less than 1E-5 per year and, A cumulative frequency of less than 1E-6 per year of exceeding the Criteria for Limiting Impact*,</p> <p>A significantly lower cumulative frequency to get either earlier or much larger releases.</p> <p>These frequency targets shall include shutdown states which have been shown to be a significant contributor in assessments of present reactor designs.</p> <p>* <u>Criteria for limiting impact (CLI)</u>: An acceptance criterion, given by a comparison of a linear combination of families of isotope releases, versus a maximum value. Each criterion is associated with a specific kind of limited consequence to the public.</p> <p>[EUR_2002]</p>

Attachment 2. Workshop notes – Guidance on definition, interpretation and application of safety goals

Discussion items	Definition of safety goal or criterion	Application and interpretation of safety goal or criterion
<p>1. What should be the scope of safety goals?</p> <ul style="list-style-type: none"> • sources of radioactive release • classes of initiating events • operational states • life cycle phases 	<ul style="list-style-type: none"> • Sources of radioactive release <ul style="list-style-type: none"> • Basically everything needs to be included. • Screening possible. • Classes of initiating events <ul style="list-style-type: none"> • Everything needs to be included. • Screening possible (considering risk for cumulative effects from frequency screening). • Operational states <ul style="list-style-type: none"> • Every state challenging a safety function must be included. • Some simplification may be acceptable • Life cycle phases <ul style="list-style-type: none"> • Safety goals need to be known and considered during design. • After that the focus is on the operating phase (including operation, shut-down, start-up and cold shut-down). 	
<p>2. What definition of risk should we use as primary goal?</p> <ul style="list-style-type: none"> • Risk to the environment/of fsite population • Risk to the plant, e.g., to the core (level 1 PSA) or to the containment (level 2 PSA) 	<ul style="list-style-type: none"> • Depends a lot on the application. What are the goals to be used for, what kinds of decisions shall be supported? • Different needs may result from regulatory and industry points of view. <ul style="list-style-type: none"> • Regulator: Goals related to level 2 PSA (LRF/LERF) • Industry: Goals related to level 1 PSA (CDF) • On the other hand... <ul style="list-style-type: none"> • Core melt is beyond design, i.e. should be of interest for authorities as well. • Level 2 PSA considerations should be involved in any application. • Surrogate criteria acceptable. • Will result in different requirements depending on site location. Handling of this is basically a political issue. • Lower level criteria may be of interest (fuel dry-out, H2 events); PSA modelling may be of interest to support deterministic analysis. • Additional safety goals would be possible to define for specific operating situations (allowed risk increase). This would be an addition to yearly average information on CDF and LRF. 	

Discussion items	Definition of safety goal or criterion	Application and interpretation of safety goal or criterion
3. Should safety goals be comparable to safety goals for other industries?	<ul style="list-style-type: none"> • Some countries apply the same requirements for all sources of risk. For this, consideration of individual risk from a NPP would be needed. • Problem of consistency. • Should basically be used. • Risk perception and communication may be a problem (risk from NPP:s seen as more negative) 	<ul style="list-style-type: none"> • Authority decision making – is comparability to other industries meaningful? • Are consequences of nuclear accidents the same or different than in other industries? <ul style="list-style-type: none"> • health risks (cancer) • People tend to compare • Risk metric • Safety goals should be defined so that risk are comparable (US definition) • How to explain risk of NPP to the public? • To whom will we compare (to public, to authority)? • To get perspective for the people working in the nuclear industry. • To study what the other industries do. • Presently different industries have different safety goals • What is the aim with goals? • Risk has two dimensions (probability, consequence) • Society acceptance varies between different risks: <ul style="list-style-type: none"> • Traffic risk versus nuclear risk • What is the benefit of the comparison of safety goals? For utility, safety authority: <ul style="list-style-type: none"> • Could be useful for communication. • Goal for the overall safety work, limits for different actors to make decisions • Background for regulatory decision making • Conclusion: <ul style="list-style-type: none"> • Useful to compare safety goals
4. Should safety goals be site-dependent or reactor size dependent?	<ul style="list-style-type: none"> • See question #5. 	
5. Should safety goals depend on number of reactors at site?	<ul style="list-style-type: none"> • For severe accidents requirements are on individual reactors. The goal is the same for all reactors independent of size (in Sweden 0,1% Cs-137 for a reactor of 1800 MWt; in Finland corresponding absolute amount). • Larger uncertainty in calculation than number of plants on site? 	

Discussion items	Definition of safety goal or criterion	Application and interpretation of safety goal or criterion
<p>6. On what levels should safety goals be defined?</p> <ul style="list-style-type: none"> • Off-site consequences • Release • Core damage 	<ul style="list-style-type: none"> • Not possible to perform a L3 PSA for every application. • See also question #2 for comments to this issue. 	<ul style="list-style-type: none"> • In Sweden, at least offsite consequences and release level • Combination of goals is good • CDF is easier to apply for the utility • Almost everybody has a level 1 PSA • Internationally, there are only few level 3 PSAs • Large uncertainties in level 2 and level 3 PSA • How often are different safety goals needed? • Different uses for different levels • Regulation demands strong safety barriers, consequently probabilistic requirements can be set. • CDF is not understandable to the public, offsite risk is most meaningful to the public. • What conclusions can be drawn from CDF w.r.t release frequency and offsite risk? e.g. in case of explaining effects of plant modifications • Conclusion: <ul style="list-style-type: none"> • Safety goals are needed in all levels • Used for different purposes
<p>7. Shall the same or different goals apply to</p> <ul style="list-style-type: none"> • new vs. operating plants? • licensing of new plant vs. regulation of same plant during operation? 	<ul style="list-style-type: none"> • Current SG:s were defined after existing plants were in operation. • Basically same goals should apply to new and existing plants. • It can be expected that new plants are safer than older, based on technology development. • New plants meet SG:s by wide margins. • Confidence in the design important. • Public expect new reactors to be safer. 	
<p>8. Should safety goals be harmonised (definition of common goals)?</p>	<ul style="list-style-type: none"> • Yes 	
<p>9. Who should define safety goals?</p>	<ul style="list-style-type: none"> • Depends on the usage. • Government, political level. • Several levels, government => regulator (other level) => utility (other level) • International co-ordination. 	

Discussion items	Definition of safety goal or criterion	Application and interpretation of safety goal or criterion
10. Use of criteria as limits vs. target		<ul style="list-style-type: none"> • Targets are more flexible, can be met or not, while limits are more rigid • From public point of view, a target can be more difficult to explain • c.f. radiation protection limits (ALARP) • Targets are values in design • Targets can be set at a more demanding level • How about use of band approach (limit and target)? • Would a band approach promote safety? • NRC reg.guide 1.17x RI-applications targets (limits?) for acceptable changes in risk level • Conclusion: <ul style="list-style-type: none"> • Both needed • Other targets such as proportional risk changes are useful • Soft criteria: balanced risk profile
11. Handling of violations of safety goals		<ul style="list-style-type: none"> • Leads easily to questions from the regulatory side regarding how the utility works with the violation, i.e., is it a conservatism in the PSA or weakness of the plant? • Plant improvements or improvements of the analysis are needed. • When studies become more detailed, it may become more difficult to fulfil the target. • Interpretation of results from PSA is important part in handling of violations. • Violation of a limit versus violation of a target? • Compare to rules and limits in the technical specifications: <ul style="list-style-type: none"> • Should risk limits be written in tech.spec.? If yes, then they are demanding. • Are we going in that direction? (e.g. definition of allowed outage times are checked with PSA). • Need for risk monitor? <ul style="list-style-type: none"> • Not yet considered in Sweden and Finland. • How to qualify PSA for risk monitor? • Risk monitor may help in understanding situation with defence-in-depth.
12. Treatment of uncertainties		<ul style="list-style-type: none"> • Needed/required in the demonstration for the regulator • In PSA, assumptions should be realistic but this is not possible in all details. • Use of sensitivity studies. • In case of plant modifications, relative change is a less uncertain risk measure than absolute risk. • Mean values should be compared to the target values. <ul style="list-style-type: none"> • Uncertainty analysis separate from the comparison with safety goals.

Discussion items	Definition of safety goal or criterion	Application and interpretation of safety goal or criterion
13. Communication of the safety goals and PSA results		<ul style="list-style-type: none"> • Difficult area • Internationally very different experiences • Relative to whom do we have safety goals? • What issues should be communicated to the public? There are a lot of nuclear safety related, and technically demanding issues. <ul style="list-style-type: none"> • safety targets for communication with the public • offsite consequences
14. What kinds of applications of safety goals can be foreseen?		<ul style="list-style-type: none"> • Risk-informed applications, to be able to measure the value of RI applications <ul style="list-style-type: none"> • Optimisation of technical specifications. • Justification of plant modifications, possibility for risk trade-offs, • Effect of modernised EOPs, development of EOPs, • Design of new builds,

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Abstract The first phase of the project (2006) described the status, concepts and history of probabilistic safety goals for nuclear power plants. The second and third phases (2007–2008) have provided guidance related to the resolution of some of the problems identified, and resulted in a common understanding regarding the definition of safety goals. The basic aim of phase 3 (2009) has been to increase the scope and level of detail of the project, and to start preparations of a guidance document. Based on the conclusions from the previous project phases, the following issues have been covered:

- Extension of international overview. Analysis of results from the questionnaire performed within the ongoing OECD/NEA WGRISK activity on probabilistic safety criteria, including participation in the preparation of the working report for OECD/NEA/WGRISK (to be finalised in phase 4).
- Use of subsidiary criteria and relations between these (to be finalised in phase 4).
- Numerical criteria when using probabilistic analyses in support of deterministic safety analysis (to be finalised in phase 4).
- Guidance for the formulation, application and interpretation of probabilistic safety criteria (to be finalised in phase 4)

Some conclusions from these activities are summarised below.

Overview of international safety goals

There are considerable differences in the status of the numerical risk criteria that have been defined in different countries. Some have been defined in law or regulations and are mandatory, some have been defined by the regulatory authority or by an authoritative body, and some have been defined by plant operators or designers. Hence the status of the criteria ranges from mandatory requirements to informal.

Three levels of risk criteria exist, i.e., at society level (mainly qualitative), at an intermediate level (quantitative and/or qualitative) and at a technical level (quantitative). The higher level criteria are typically concerned with the actual risk to society or individuals, or to the environment. Technical criteria are generally quantitative (probabilistic) and mostly on lower levels (subsidiary). They typically concern core damage, unacceptable release, and unacceptable health risks. In later years, some countries have defined separate criteria to address robustness in defence in depth, e.g., by having a separate criterion for containment integrity.

In most countries that have criteria both for existing and new reactors, the criteria are more strict for new plants. In some cases this is expressed by using the same numerical values for the frequencies, but applying them as limits for new plants and targets for existing plants.

Regarding consideration of uncertainties, there is consensus that the comparison with probabilistic safety criteria should use the “best estimate” of the PSA results.

In most cases, risk criteria for operating plants are applied when the PSA is updated, which results in a very large spread, as a considerable part of the respondents only update the PSA within the framework of the periodic safety review (generally 10 years), while others do updates on a yearly basis. Risk criteria are also used to assess the impact on risk of design modifications in the plant.

Risk criteria are mostly considered as indicators or orientation values, meaning that no regulatory actions are expected on non-compliance with a probabilistic safety criterion. Practically, there is a consensus on finding the reasons for the non-compliance and identification on the way to overcome it. However, for new builds application of risk criteria would be stricter.

When it comes to the interpretation of the criteria, several of the respondents agree that more work is needed in the definition of the various criteria. Thus, there seems to be a need for a common definition as to what constitutes severe core damage and large release. A strict and common definition would facilitate comparison of risks and results between different plants.

The general experience from the implementation of risk criteria is positive. Respondents who have implemented criteria have experienced various benefits. In a number of cases, design weaknesses or procedural weaknesses in NPPs have been identified using PSA and PSA criteria, resulting in the introduction of safety improvements. In many cases, the implementation of risk criteria and safety goals has led to plant modifications in order to meet the probabilistic risk criteria. The implementation of safety goals often emphasizes the need for more detailed and realistic PSA models, and it appears that the use of safety goals has increased the focus on the correctness and quality of PSA models.

Use of subsidiary criteria and relations between these
Goals related to CDF and LRF are surrogates to societal and individual risk level criteria. To fully validate these goals, calculations of environmental consequences of release sequences would need to be made. In a few countries, the performance of level 3 PSAs is required, which would enable a direct evaluation of the compliance against societal and individual risk level criteria.

There are several aspects in the validation of subsidiary risk criteria, such as validity with respect to the definition of risk, validity with respect to rational decision making under risk, and validity with respect to the use of PSA. Taking these aspects into account means a combination of top-down and bottom-up approaches in the derivation of subsidiary risk criteria. For the top down approach, several references exist for societal and individual level risk criteria, and can be used as a basis, with decision theory providing the framework for the definition of rational risk criteria. For the bottom-up approach, it is necessary to define the objectives with PSA and how safety goals are used. Also experience from present PSAs are valuable in this process.

Numerical criteria when using probabilistic analyses in support of deterministic safety analysis

This sub-task mainly monitors the outcome from other on-going work regarding the relation between PSA and defence in depth.

Guidance for the formulation, application and interpretation of probabilistic safety criteria
This issue is an important part of the final project phase (2009). At the project seminar in December 2008, representatives from Nordic utilities and authorities participated in a workshop with discussions held along two lines, i.e., guidance related to either the definition or to the application and interpretation safety goals or criteria. Valuable input was received, and will be used as part of the basis for the guidance document to be developed during the concluding project phase

Key words

Safety Goals, PSA, Safety Targets, ALARP, Decision criteria, Risk informed decision making