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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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Probabilistic Risk Criteria and Safety Goals

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FOREWORD

The main mission of the Working Group on Risk Assessment (WGRISK) is to advance the understanding and utilisation of Probabilistic Safety Assessment (PSA) in ensuring continued safety of nuclear installations in Member countries. In pursuing this goal, the Working Group shall recognise the different methodologies for identifying contributors to risk and assessing their importance. While the Working Group shall continue to focus on the more mature PSA methodologies for Level 1, Level 2, internal, external, shutdown, etc. It shall also consider the applicability and maturity of PSA methods for considering evolving issues such as human reliability, software reliability, ageing issues, safety goals, etc., as appropriate.

Considering the work of the COOPRA (Cooperative Research on PSA) working group on Risk-Informed Decision-Making and the Nordic Project "Validity of Safety Goals", the WGRISK initiated task (2006)-2 "Probabilistic Risk Criteria"

The scope includes the whole range of criteria from individual and societal risk, off-site release, core damage and lower level goals to numerical criteria used in various risk-informed applications. Sometimes, wording "safety goals" is used for the upper level criteria. The focus of this task will be on gathering information (methodological and rationales) related to the setting and technical application of the criteria, and to consequences for the status and use of PSA. Both regulatory criteria and those defined and used by utilities will be covered.

The Task has a direct coupling to the WGRISK task 2003-2 (to be finalised in winter 2006-7): "Use of risk information in the regulatory process" and is its follow-up. It partly continues the work of the International COOPRA project working group "Risk-Informed Decision Making (RIDeM)" (e.g. same chairperson and risk criteria carried on as a topic) and has a relation to WGRisk Task 2005-1 "The Use and Development of Probabilistic Safety Assessment (PSA) in Member and non-Member Countries".

This work represents the collective effort of the task group all of whom provided valuable time and considerable knowledge toward its production. In offering it thanks to these experts, the NEA Secretariat wishes to express to provide particular appreciation to M. Philippe Hessel (CNSC, Canada), who as task leader adeptly chaired the many meetings and provided overall co-ordination towards completing the report and to MM. Michael Knochenhauer (Relcon-Scandpower, Sweden) and Jan-Erik Holmberg (VTT, Finland) who provided the connection with the Nordic Project and provided excellent support.

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EXECUTIVE SUMMARY

Probabilistic Safety Criteria, including Safety Goals, have been progressively introduced by regulatory bodies and utilities. They range from high level qualitative statements (e.g., "The use of nuclear energy must be safe") to technical criteria (e.g., probability of fuel cladding temperature being higher than 1204 ^{o}C).They have been published in different ways, from legal documents to internal guides. They can be applied as legal limits (not meeting them is an offence) down to "orientation values".

The questionnaire produced for this tasks requested information on the above issues, with added questions on the basis for the criteria, the way they are applied and experience on their use.

Answers have been received from 13 nuclear safety organizations (Canada, Belgium, Chinese Taipei, Finland, France, Hungary, Japan, Korea, Slovakia, Sweden, Switzerland, UK and USA) and 6 utilities (Hydro-Québec, Fortum, OKG, Ontario-Power-Generation, Ringhals and TVO). Two of the regulatory bodies (Belgium and Chinese Taipei) declared they have not set (and do not intend to set) any Probabilistic Safety Criterion. Some supplementary information (three countries) has been taken from a questionnaire on Safety Goals during the 20-24 November 2006 IAEA Technical Meeting on the development of draft DS-394. This report is based on information given in the annexed questionnaire. More information that could be found in other CSNI reports is not considered here*.

The reported Probabilistic Safety Criteria can be grouped into 4 categories, in relation with the tools to be used for assessing compliance:

- Core Damage Frequency (CDF) – Level 1 PSA – 16 respondents
- Releases Frequency (LERF, LRF, SRF) – Level 2 PSA – 14 respondents
- Frequency of Doses – Level 3 PSA – 4 respondents
- Criteria on Containment Failure – System level – 2 respondents

Several respondents use more than one criterion (e.g., CDF and LERF) while some others use a range of values for a given criterion (e.g., frequency of doses to the public, to the workers, during accidents, during normal operations).

While originally set considering the state of the art of PSA, the CDF criterion is presently considered as based on Defence-In-Depth. Also, the Criteria on Containment Failure, newly introduced in Japan and USA, is an expression of Defence-In-Depth as new designs could meet the LERF without taking containment into account.

Releases Frequency and Frequency of doses address public safety. However, while the frequency of doses addresses directly public health, Releases Frequency considers that public safety is achieved for a given release (within a given time for LERF), taking into account Emergency Measures (such as evacuation).

* The intent of the Task group is that all information in this report can be traced back to the responses to the questionnaire. This could not be possible from existing CSNI reports.

The values associated with CDF vary from 5 E-4 per year to 1 E-5 per year. When indicated, this spread is reduced when considering new plants where all respondents but 2 set the CDF to 1 E-5.

The values associated to releases frequency show a wider spread, from 1 E-5 per year to 1 E-7 per year. As for the CDF, the spread is reduced when considering new plants, where all respondents but one set the LRF (or LERF) to 1 E-6 per year. It has to be noted that the results are highly related to the scope and detail of the reference PSA, so the numerical values cannot be compared without a complete definition of the scope covered by the PSA.

Generally, all respondents considered introduction of Probabilistic Safety criteria resulted in safety improvements.

Opinion is widespread on the benefits of using Probabilistic Safety Criteria for communication with the public, ranging from bad to good experiences. It seems that there is a strong relation with each country culture and the circumstances.

The responses to the questionnaires suggested that more work should be considered in the definition of Releases Frequencies: some regulators include a time range (generally 24 hours) in the criterion while others do not limit the time to be considered. It is suggested that, in the first case, the existing PSAs should be revisited to assess if long development accident sequences were considered.

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1. STATUS OF PROBABILISTIC RISK CRITERIA

Probabilistic safety criteria differing in formal status, levels and definitions have been in use for the past two to three decades in most member states, generating an extensive body of experience. The scope includes the whole range of safety criteria from societal risk, off-site release, core damage and lower level criteria to numerical criteria used in various risk-informed applications.

In most of the countries in which numerical safety criteria have been defined they have been defined as a "target", an "objective" or a "goal" where the recommendation is that the risk should be lower than the prescribed value. However, these criteria are not generally considered as a regulatory limit.

1.1 Status of probabilistic risk criteria

There are 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 can be seen:

- A legally strict value to be fulfilled. Design must be changed, if the criterion is not met. In some countries probabilistic risk 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 ALARP[1] principle. They denote a boundary that, if surpassed, will often lead to increased regulatory oversight. It is only used as one piece of information in the regulatory process (risk-informed not risk-based).

For most respondents probabilistic risk criteria are target values, orientation values or safety indicators.

1.2 Differences in criteria for existing plants, life extension, new builds, new design[2]

In several countries, different criteria apply to existing plants and for new plants or the criteria have different status. The following categories of statuses can be seen:

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

[2] While this section highlights the differences, the rest of this report would address also similarities.

- Probabilistic risk criteria are the same for existing and future plants[3]
- Probabilistic risk criteria use a similar metric for existing and future plants. The numerical values for the frequencies are a factor (typically 10) lower for future plants.
- Probabilistic risk criteria involve the same numerical values for the frequencies, considered as limits for future plants and targets for existing plants.
- Probabilistic risk criteria are defined only for existing plants[4].
- 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.

For modernisation and life extension, generally same criteria are applied as for operating plants.

[3] Japan uses the same criteria for existing and future plants. However, JNSC recommends utilities to make further efforts to develop new reactors that have a lower risk than existing ones.

[4] This is the case where new plants are presently expected.

2. WHAT PROBABILISTIC RISK CRITERIA EXIST?

The questionnaire defined three levels of Probabilistic Risk Criteria:

- 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)

However, the separation between society level and intermediate level is not always clear.

2.1 Society level criteria

None of these criteria is probabilistic. However, they are of interest as they are the basis for the probabilistic criteria.

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[5].

These criteria can mainly be abstracted into "Prevent **unreasonable** risk to the public and the environment".

In some countries, the criterion is strict:

- In Finland (STUK) where the criterion is given as "The use of nuclear energy **must be** safe; **it shall not cause**…".
- In France (ASN), where the criterion is 'Practical elimination of consequences'

2.2 Intermediate level

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"**

These "other risks", when defined, are characterized as one or several criteria:

- the background risk of cancer,
- the risk from other sources of energy production,
- the risk of fatality for all other sources (early or late),

[5] This criterion is very similar to the 'Intermediate level' and will be considered as such.

Some regulators define the "risk" in a quantitative way:

- with amount of radionuclide (for instance 100 TBq of Cs_{137}, in relation with the risk of long term relocation of public),
- with a given dose (for instance 0.1 mSv from normal operation of a nuclear power plant)
- with a probability of consequences (less than 1% of other causes of cancer),
- with a frequency of occurrence (should not exceed approximately 1E-6 per year)

2.3 Technical Level

Only at "Technical Level" the criteria are fully probabilistic.

All respondents but two have at least one Probabilistic risk Criterion.

Twelve Probabilistic Risk Criteria are indicated in the responses:

- Core Damage frequency (14 respondents use this criterion),
- Large Release frequency (12 respondents use this criterion),
- Small Release frequency (1 respondent uses this criterion),
- Individual risk of fatalities (3 respondents use this criterion),
- Systems reliability targets (2 respondents use this criterion),
- Containment Failure Frequency (1 respondent uses this criterion),
- General Objectives (1 respondent uses this criterion),
- Risk Related to Shutdown conditions (1 respondent uses this criterion),
- Objectives related to EPR (1 respondent uses this criterion),
- Instantaneous risk (1 respondent uses this criterion),
- Frequency of doses (1 respondent uses this criterion), and
- Societal risk (1 respondent uses this criterion).

Of these criteria, 2 are considered as out of the task scope (systems reliability targets and instantaneous risk). As they are addressed by other WGRisk tasks, they are not considered in the following analysis.

In practice, the remaining 10 criteria can be compounded into 4 Probabilistic Risk Criteria:

- Core Damage Frequency,
- Frequency of Releases
- Criteria on Containment Failure,
- Frequency of doses, and

Each of these Probabilistic Risk Criteria is discussed in the following chapters.

The Core Damage Frequency criterion is shared by 14 respondents. The associated measures are consistent, differing only according to the reactors' technology.

Frequency of Releases or Containment Failure Frequency is also shared by a large majority of respondents. They address the risk to the public from release of radionuclide at a high level. It does not consider directly the health effect of radiation. This criterion shows some conceptual differences between respondents, essentially on the timing of the releases. The task group has decided to consider them as an "Open Issue" (Chapter 8 of this report).

Frequency of doses is considered by only a few respondents. It is expressed either in terms of probability of health effects or in terms of doses.

3. ANALYSIS OF PROBABILISTIC RISK CRITERIA

For each identified Probabilistic Risk Criterion, the respondents were requested to:

- Define the criterion
- Explain how the criterion is expressed
- Define its applicability (to a plant…)
- State the scope of the analysis supporting compliance
- Provide the rationales supporting the criterion

3.1 Core Damage Frequency criterion

3.1.1 Definition of core damage frequency

The criterion core damage frequency is used by 14 of the respondents. However, the definition of the criterion differs considerably with the reactors technology. For instance, for reactors of CANDU type, the core damage is defined as loss of structural integrity of more than one fuel channel.

Some countries have very precise technical definitions of CDF, e.g. defining core damage as local fuel temperature above 1204 ºC, i.e., the limit defined in section 1b of 10 CFR 50.46 (Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors). Other countries have more general definitions referring, for instance to prolonged core uncover or long-term cooling.

The frequency limits regarding core damage vary between 1 E-4 and 1 E-6 per year.

Requirements for new plants are typically stricter (in terms of frequency) than for existing ones, and are mandatory as opposed to indicative. For instance, in Switzerland and Finland it is required by regulation that the applicant for a permit to build a new nuclear power plant shall demonstrate that the core damage frequency is below 1 E-5 per year.

Figure 1 summarises numerical criteria defined for core damage.

3.1.2 How is core damage frequency expressed

Of the 14 respondents that use CDF as a criterion, 13 express the frequency as a single value. Only Ontario Power Generation expresses it as a band. As mentioned above this frequency varies between 1 E-4 and 1 E-6 per year.

Where a single value was not used, the criterion was defined as a band with a limit and a target value.

The USNRC also noted that for decision-making purposes, there are several bands of varying CDF and ΔCDF values (see section 8.3).

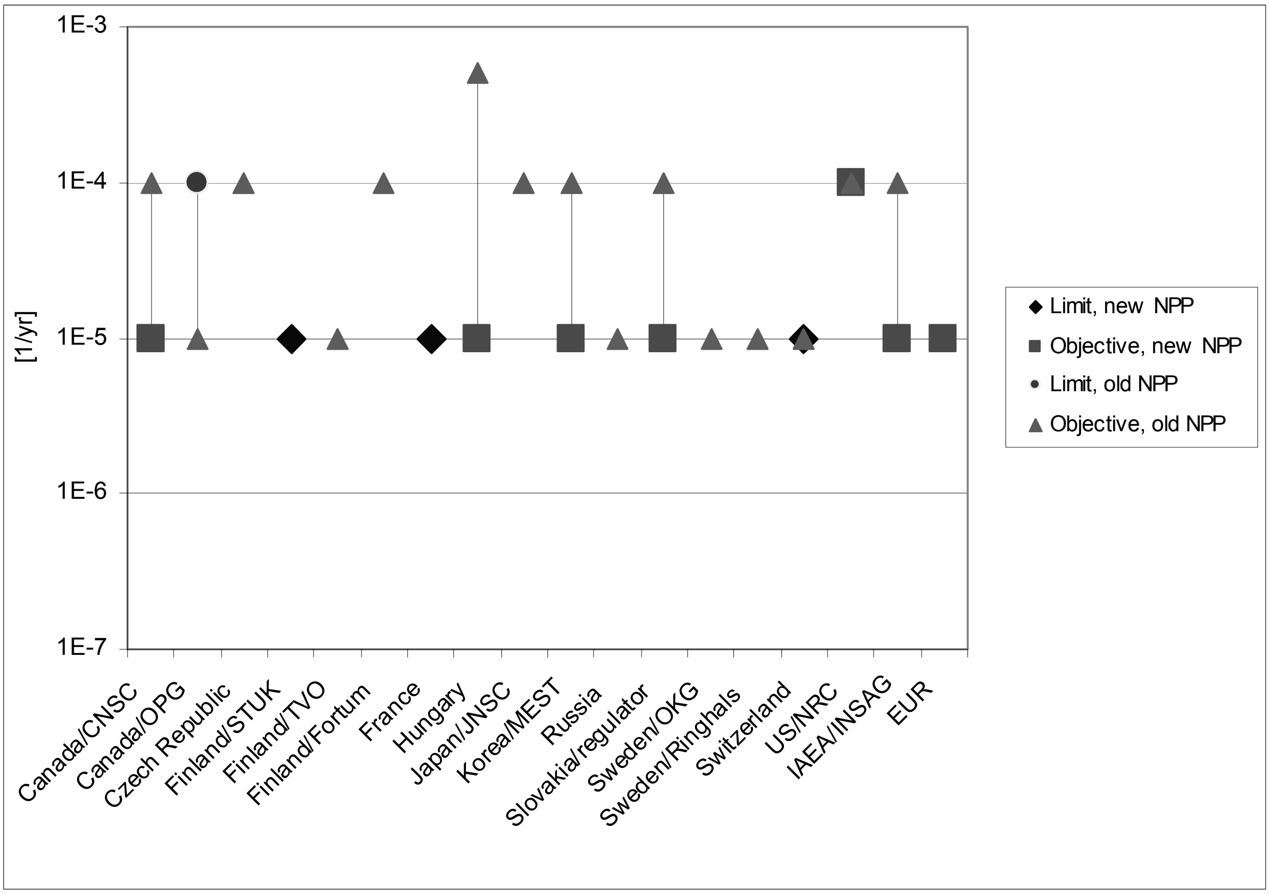

Figure 1: Numerical criteria defined for Core Damage.

3.1.3 The applicability of core damage frequency

In every country the criterion is applicable at reactor level.

In the Finnish answer it was added as a justification (for applying criteria on reactor level) that the plant units are completely separated and have no common safety related systems. The USNRC justified application on reactor level with the aim to be able to evaluate the safety of each individual reactor.

3.1.4 The scope of the analysis supporting compliance

In general, all countries aim at using full scope (internal and external events, full power and shutdown operating modes) PSA to assess CDF. The vast majority of the respondents require full scope PSA, but some comments are given on the degree of maturity for some parts of the analysis, and the degree of uncertainty associated with some initiating event categories.

The requirements among the responding countries differ somewhat. According to Swiss requirements, the CDF is to be calculated for full power operating mode of the reactor. Fuel damage during other operating modes is covered by a separate risk measure called Fuel Damage Frequency (FDF). The U.S. Nuclear Regulatory Commission just states that: *"The analysis must be of sufficient scope, depth, and quality to support the decision to be made. Generally, internal and external events are considered."* whereas the Canadian Nuclear Safety Commission requires full scope PSAs (including external events). However, the CNSC has prescribed that for external events, use of other methods to demonstrate low risk may be acceptable.

3.1.5 The rationales supporting core damage frequency

In most cases CDF had been selected as a criterion according to the defence-in-depth concept, e.g., to avoid the design of a plant whose safety relies too much on a strong containment.

Some respondents gave reasons to why they used CDF. The general main reason seems to be the achievement of a high safety level, whether set as an internal goal or a requirement from an external regulatory body. Furthermore some countries pointed out the benefits of the criteria being a well known measure that is used by several countries all over the world, thus allowing comparisons of results. The criterion is also frequently used for decision making regarding plant modifications.

Almost every country has documentation supporting the criterion. In some cases the documentation consists of USNRC and/or IAEA documents. For utilities, internal documents are sometimes complemented with national documents. In those cases where there is no documentation supporting the CDF, the proposed CDF has been compared with international practice.

3.2 Frequency of Releases Criterion

3.2.1 Definition of the Frequency of Releases Criterion

There is both a considerably larger variation in the frequency limits, and very different answers to the question of what constitutes an unacceptable release. As with the CDF, the magnitudes are sometimes based on IAEA safety goals suggested for existing plants, i.e., on the level of 1 E-5 per year (IAEA-INSAG-12). However, most countries seem to define much stricter limits, between 1 E-6 per year and 1 E-7 per year.

The definition of what constitutes an unacceptable release differs a lot, and there are many parameters involved in the definition, the most important ones being the time, the amount and the composition of the release. Additionally, other aspects may be of interest, such as the height above ground of the point of release. The underlying reason for the complexity of the release definition is largely the fact that it constitutes the link between the PSA level 2 results and an indirect attempt to assess health effects from the release. However, such consequence issues are basically addressed in PSA level 3, and can only be fully covered in such an analysis.

The release for which a numerical criterion is given is also defined in several different ways:

- Large release: This is defined as an absolute magnitude of activity and isotope released, e.g., 100 TBq of Cs_{137}.
- Large early release: These definitions are more qualitative, e.g., "Large off-site releases requiring short term off-site response," "Significant, or large release of Cs_{137}, fission products before applying the offsite protective measures," "Rapid, unmitigated large release of airborne fission products from the containment to the environment, resulting in the early death of more than 1 person or causing a severe social effect."
- Small release: CNSC from Canada has set criteria both for large and small release. A small release is defined as a release of 1000 TBq of I_{131}.
- Unacceptable consequence. This is a French definition which is fully open and rather old (1977). In the meantime a more precise definition was proposed by EDF and is still under discussion between EDF and the French safety authority. It should be noted that the performance of level 2 PSA was not required in France by the safety authority for existing reactors but has been carried out on a voluntary basis. In case if new reactors, a level 2 PSA is required.

- Containment failure. The Japanese Nuclear Safety Commission proposes a criterion for containment failure frequency. In Finland, STUK had defined, in the first version of the Guide YVL-2.8, a probabilistic criterion for containment isolation failure (conditional failure probability). This is a requirement that aims at assuring the robustness of the defence-in-depth.

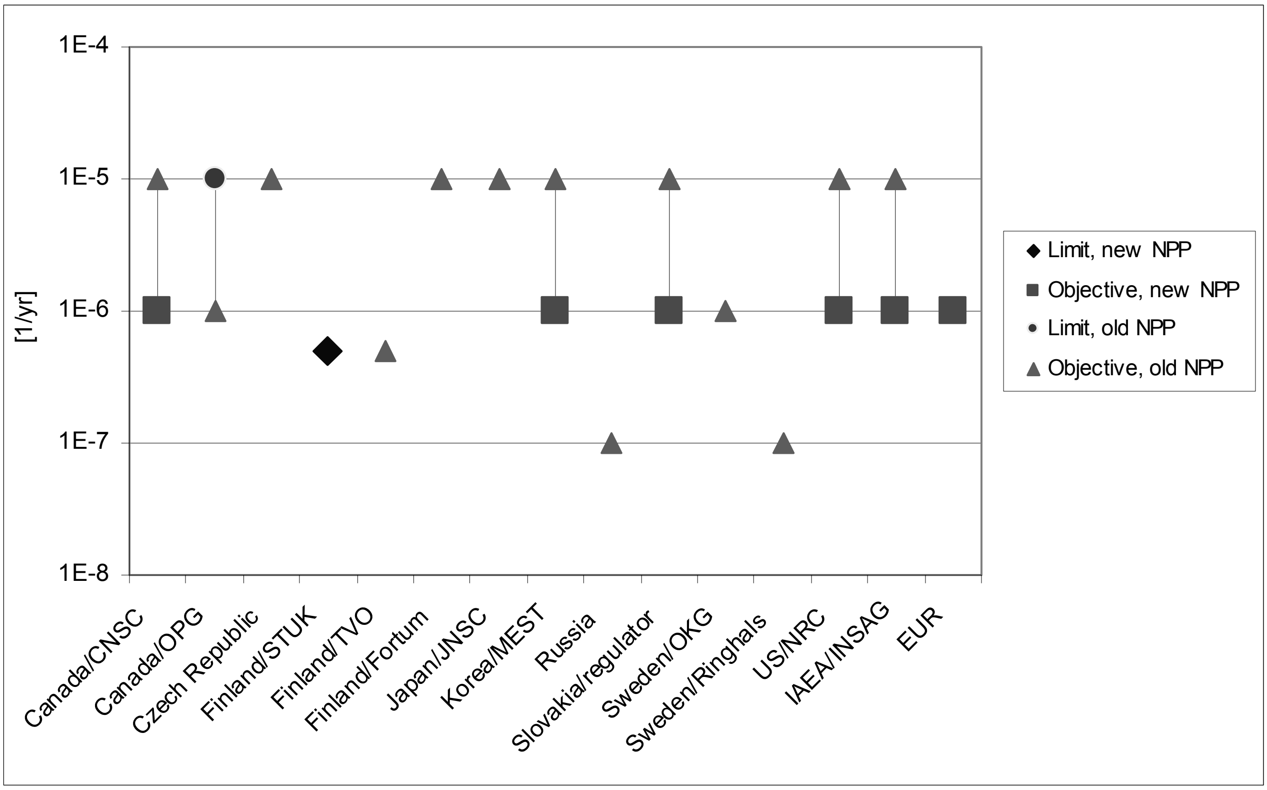

Figure 2: Numerical criteria defined for large release. Definition and timing of "large release" varies (see Compilation of responses)

Figure 2 summarises numerical criteria defined for large release. As explained above, the definitions for "large release" is not the same for all organisations. However, it can be seen that objectives vary from 1 E-7/year to 1 E-5/year, which is a quite large spread, larger than for core damage frequency, where objectives vary between 1 E-5/year and 1 E-4/year.

3.2.2 Expression of the Frequency of Releases criterion

In most cases, the criterion is expressed as a single frequency value for a large (early) release. Some organizations specify that the comparison is made against the mean value. One utility uses a band (limit and target).

3.2.3 Applicability of the Frequency of Releases criterion

In all cases, the criterion is applied to a reactor-unit.

3.2.4 Scope of the analysis supporting compliance

The PSA shall be full scope (all initiating events, all operating modes). In few cases, if the PSA is not a full scope PSA, a limited scope PSA is accepted, and qualitative methods or methods can be required to demonstrate that the risk is acceptable.

3.2.5 What are the rationales supporting Frequency of Releases Criterion

In most cases, the Frequency of Releases Criterion is based on protecting the public against prompt fatalities and radiological-induced cancers. The LERF criterion is based on the time being sufficient for public evacuation before a significant release occurs.

Some countries (such as Canada, Finland and Sweden) do not consider the releases timing and base their criteria on land contamination, i.e. the long term consequences of an accident. Canada explains this choice on societal effects, considering that impact on the public from relocation and land banning goes largely beyond the radiological effects.

3.3 Criteria on Containment Failure

The criterion for a containment failure frequency (CFF) is only defined by the Japanese Nuclear Safety Commission (NSC). In addition, for new or advanced nuclear power plants the US NRC has set a target for conditional containment failure probability.

3.3.1 Definition of the Containment Failure Frequency Criterion

In Japan, NSC stated that a performance goal for each type of nuclear facility be examined and set as safety benchmark to demonstrate compliance with these quantitative health objectives for measuring plant performance. For LWRs, the containment failure criterion E-5/year was derived from the quantitative health objective value, and the conditional average probabilities of acute fatality or cancer fatality of the individuals in the surrounding area determined for a hypothetical large release (< 0.1). Here the containment failure frequency means the sum of the frequencies of various containment function failure modes ranging from relatively small leaks to a large and early break of the containment. If the same number is used for the frequency criterion, this definition of containment failure frequency yields a stricter requirement for containment performance than a LERF criterion.

3.3.2 Expression of the Containment Failure Frequency Criterion

In Japan, NSC has proposed 1 E-5/year as a frequency criterion for LWRs.

3.3.3 Applicability of the Containment Failure Frequency criterion

In Japan, the CFF is set at reactor level.

These criteria are considered as safety indicator.

3.3.4 The scope of the analysis supporting compliance

These performance goals are also applied to nuclear accidents caused by both internal and external events except for intentional man-made events such as sabotage. However, there are large differences in maturity of risk assessment techniques and uncertainty bands of PSA results for external events such as tsunami and floods for which the experience of PSA and the development of database are not sufficient. In real applications of the performance goals, PSA will not be necessarily performed for all the initiating events.

3.3.5 Rationales supporting the Containment Failure Frequency criterion

The reason for using CFF and not using LERF (large early release frequency) is that, although the LERF has closer relationship to individual risks, the CFF gives more conservative assessment when the same value is taken for CFF and LERF and it is a way to cope with the uncertainties in the quantification of source terms and the effectiveness of emergency protective measures, etc.

3.3.6 Conditional Containment failure Probability

In the USA, the NRC expects new or advanced nuclear power plants to present a higher level of severe accident safety performance consistent with the NRC's Severe Accident Policy Statement.

	CDF	LERF	Conditional Containment Failure Probability
Operating Plants & License Renewal	<1E-04	<1E-05	n/a
New Plants	<1E-04	<1E-06	<0.1

3.4 Frequency of Doses Criterion

While being generally the basis for the "technical level" criteria, only two respondents (Canada-OPG and UK) use frequency of doses as a technical level criterion per-se.

3.4.1 Definition of the Frequency of Doses Criterion

Risks are divided into fatal acute or fatal late health risks and these can be calculated for an individual or a group. In both cases, risk is defined as the risk to the member of a critical group that receives maximum exposure from an accident.

Typically acute health effects have a threshold dose value under which the probability of health effect is zero, but above which the probability of acute health effect is increased with increasing dose. On the other hand most late health effects do not have threshold values for dose. Based on these assumptions acute health effects can be expected in the vicinity of the release point, whereas late health effects appear in the public exposed to radiation over larger areas.

3.4.2 Expression of the Frequency of doses criterion

Rate of exposure in Sv/yr to the individual and/or probability of latent health effects.

3.4.3 Applicability of the Frequency of doses criterion

For Canada-OPG, the criterion applies at station level[6], for the UK at site level.

3.4.4 The scope of the analysis supporting compliance

Generally full scope PSA.

[6] A station consists of multiple units of identical design, operated from a common control room and for which performance data is collected and reported.

3.4.5 Rationales supporting the Frequency of doses criterion

This criterion is generally considered as meeting directly the high level goal of "no added significant risk to the public".

3.4.6 Rationales supporting the Screening Criterion

When identified, the Screening Criterion is based on NUREG-0800-section 2.2.3

4. CONSIDERATION OF UNCERTAINTY

Several organisations and individuals have claimed that the results of the PSA used to evaluate compliance with Probabilistic Risk Criteria are subject to large uncertainties.

Therefore, the questionnaire included a question on the way the respondents address uncertainty in their assessment of compliance with the criteria.

The responses to the questionnaire show a large consensus, all respondents stating that the comparison with probabilistic risk 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 different goal, without any added value.

All respondents consider that uncertainty analysis is an integral part of a PSA, and that its results are considered in the decision-making process.

Sensitivity analysis (sensitivity to assumption and to data) is also noted by several respondents as an integral part of the PSA and one of the components of the decision-making process when assessing compliance with the criteria.

5. WHEN AND HOW DO PROBABILISTIC RISK CRITERIA APPLY

The original questionnaire included 2 questions on the issue:

- When do you require evaluation of plant performance against the Probabilistic Risk Criteria?
- What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

After analysing the received responses, the task group found that the responses to the first question addressed two different topics:

- When is the analysis supporting evaluation of the Risk Criterion evaluated?, and
- When is the risk criterion used?

Clarification was requested from the respondents. The received answers are used in this report.

5.1 When is the analysis supporting the Risk Criteria evaluated?

The responses show the differences between the different regulatory regimes:

Eight regulatory bodies and 5 utilities answered this question.

- For six of them, the PSA supporting evaluation of the Risk Criteria shall be updated within the framework of the Periodic Safety Review (generally 10 years).
- One (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 (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.

5.2 When is the Probabilistic Risk Criteria used?

4 regulatory bodies and 5 licensees answered this question.

All respondents 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.

5.3 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

The received response show considerable differences between the different countries regulatory regimes. As the Risk Criteria are generally considered as Indicators or Orientation Values (see chapter 3 above), no hard regulatory actions are expected on non-compliance with a Probabilistic Risk 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 granting an operating license.

The utilities from Sweden declare using rules on non-compliance to the Core Damage criterion. The action levels specified for core damage frequency are:

- PSA results >1 E-3/year – immediate shutdown
- 1 E-3/year > PSA results > 1 E-4/year – correction at next planned yearly shutdown
- 1 E-4/year > PSA results > E-5/year – long-term planning of actions

6. 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 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 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 probabilistic risk criteria often emphasises 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 probabilistic risk criteria 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 risk 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 a negative 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 risk 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 more precise and common definition would facilitate comparison of risks and results between different plants.

7. EXPERIENCE ON COMMUNICATION OF PROBABILISTIC RISK CRITERIA

Responses regarding communication of probabilistic risk criteria to the public vary 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 probabilistic risk criteria are met, many of the respondents have found the results useful when communicating the level of safety to the public. For example, Ontario Power Generation explains how the fact that their plants meet the probabilistic risk criteria has been cited to support the adequacy of plant safety at public hearings and public inquiries related to nuclear power.

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 “1 E-7/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 probabilistic risk criteria should be made easier to understand to the public, it is important that it can be clearly demonstrated that PSA results and probabilistic risk criteriahave lead to safety improvements in plants. However, the format in which PSA results and risk or safety criterion are presented needs to be carefully considered, in order to minimize the risk for misinterpretation or misunderstanding.

The US-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,” January 2004 contains a comparative analysis of NRC’s risk communication needs and state-of-the-art risk communication practices. The document provides the Risk Communication Guidelines and how the NRC can best incorporate risk communication principles throughout the agency.

8. OPEN ISSUES

When reviewing the responses to the questionnaire, the task group members noted that several responses put in light differences among the countries on definition and use of Probabilistic Risk Criteria. They suggested some of these differences be exposed in a specific chapter as information that could be considered by the different countries in the development of their Probabilistic Risk Criteria framework.

Three issues have been identified:

- not early releases, i.e., late releases
- small releases,
- use of band criterion

8.1 Not Early Release, i.e., late releases

At present, three countries[7] define the Probabilistic Risk Criteria on releases without limitation on the timing of these releases ("Large Release Frequency" Vs "Large Early Release Frequency").

Analysis of the responses and interrogation of these countries representatives shown that the non indication of time is based on the fact that a release would trigger public evacuation (and possible relocation) what would be the timing of the release. This would result, beyond the received doses, into disruption of public life. This can be linked to the fact that high-level criteria in these countries include consideration of effects on the environment.

During the Task Group meetings, several member countries representatives noted that, due to the Early characteristic of the Risk Criterion (LERF), some PSAs could not consider long development sequences that could lead to significant releases more than 24 hours after the Initiating Event. Consequently, the risk from some nuclear plants could be significantly under-evaluated.

The task group members recommend that:

Every country check if in the existing PSAs long development sequences are considered as leading to releases.

8.2 Small Releases

One country[8] has defined a Probabilistic Risk Criterion on frequency of Small Releases. This criterion is initially related to tube-type reactors (CANDU and Gas-Cooled reactors) where severe fuel damage could occur on single channel fuel content, the rest of the fuel being correctly cooled, concurrently with some containment impairment. For the UK, small releases are indirectly considered by the formulation of the dose criteria.

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[7] Canada (CNSC), Finland (STUK), Sweden (SSM and utilities)

[8] Canada

Several country representatives noted that such a risk could also exist on PWR and BWR reactors under some specific configurations.

As a small release would trigger emergency measures, the task group members recommend that

Every country consider the interest of for setting a Probabilistic Risk Criterion on Small Releases.

8.3 Use of Band Criterion

Among the respondents, only one country[9] uses a band-type Probabilistic Risk Criterion. However, some countries (and/or utilities)[10] have indicated practical use of band criteria for decision-making.

Some members of the task group noted that these band-type criteria could also be useful to address uncertainty in the PSA results.

The NKS project addresses the issue of the use of band criterion.

[9] UK

[10] Canada (CNSC and OPG), Sweden (Ringhals and Oskarshamn) and USA

9. REFERENCED DOCUMENTS

The respondents provided lists of reference documents, often from several sources. In order to make these references list practical, we have grouped them by originating country or organisation.

Canada/CNSC:

- Regulatory Document RD-337 – Design of New Nuclear Power Plants http://www.suretenucleaire.gc.ca/fr/lawsregs/regulatorydocuments/published/index.cfm
- "Safety Goals for S-337" CNSC reference 968790 (available on request)

Canada/OPG:

- "Dinnie, K.S., Experience with the Application of Risk-Based Safety Goals, 25th Annual Conference of the Canadian Nuclear Society, June 2004."
- OPG Nuclear Safety Policy, N-POL-0001 R00
- OPG Risk and Reliability Program, N-PROG-RA-0016 R05

Finland/STUK:

- Decision of the Council of State on the general regulations for the safety of nuclear power plants, 14 Feb. 1991 (395/1991) http://www.edilex.fi/stuklex/en/lainsaadanto/19910395/P29
- Guide YVL 1.0 Safety criteria for design of nuclear power plants, http://www.edilex.fi/stuklex/en/lainsaadanto/saannosto/YVL1-0
- Guide YVL 2.8 Probabilistic safety analysis in safety management of nuclear power plants http://www.edilex.fi/stuklex/en/lainsaadanto/saannosto/YVL2-8

Finland /Utilities:

- Vaurio, J.K.; Safety-related decision making at a nuclear power plant; Nuclear Engineering and Design 185 (1998) pp 335–345; Imatran Voima Oy, Loviisa Power Station; 1998
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France:

- Letter 1076/77 of the Nuclear Safety Division published in 1977.
- "Technical Guidelines for Future PWRs"- IPSN/GRS report 82 (Nov. 2000).

Hungary:

- Volume 3 of the Nuclear Safety Codes issued by the Hungarian Governmental Decree No. 89/2005

Japan:

- "Report on Performance Goals for Light Water Power Reactors, - on performance goals consistent with safety goals - (in Japanese)", Special Committee on Nuclear Safety Goals of NSC, March 2006. http://www.nsc.go.jp/anzen/shidai/genan2006/genan023/siryo1.pdf
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Korea:

- Policy on severe accident (2001)

Slovakia:

- Regulatory Guideline, Requirements for PSA performance, UJD SR, BNS I.4.2/2006, Bratislava, Slovakia, November 2006.

Sweden/SSM:

- SKI / SSI; Utsläppsbegränsande åtgärder vid svåra härdhaverier (in Swedish; Release limiting measures after severe accidents); SKI ref 7.1.24 1082/85; SKI / SSI; 1985

Sweden/Oskarshamn:

- Gunnarsson, Kerstin; Mål och strategi för modernisering avseende reaktorsäkerhet av OKG:s kärnkraftblock (in Swedish; Goals and strategies for modenisation regarding reactor safety in OKG NPP:s); OKG/96-00385; OKG AB; 1997
- Lindahl, Pär; Giltighet av probabilistiska säkerhetsmål (in Swedish; The Validity of Safety Goals); OKG/2005-14190; OKG AB; 2005
- Lindahl, Pär; Metodbeskrivning för probabilistisk analys av rumshändelser med avseende på risken för härdskada/bränsleskada (in Swedish; Method description for probabilstic analysis of area events); OKG Anvisning 2006-09475; OKG AB; 14
- Industridepartementet; Villkor för fortsatt tillstånd enligt 5 § lagen (1984:3) om kärnteknisk verksamhet för att driva kärnkraftreaktorerna Sweden - Oskarshamn I, II och III (in Swedish; Preconditions for allowing continued operation of the NPP:s Oskarshamn 1, 2 and 3);
- Industridepartementet 2717/85 1986-02-27 (dossier 8523); Industridepartementet; 1986

Sweden/Ringhals:

- Ringhals document "Övergripande mål och förhållningssätt för reaktorsäkerhet" (in Swedish; Over-all goals and apporaches to reactor safety), id 1839723.

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UK:

– "Safety Assessment Principles for Nuclear Facilities", 2006 Edition, HSE, www.hse.gov.uk/nuclear/saps/saps2006.pdf

– "Numerical targets and legal limits in Safety Assessment Principles for Nuclear Facilities, An explanatory note", December 2006, HSE, www.hse.gov.uk/nuclear/saps/explanation.pdf

– "Tolerability of risk from nuclear power stations" www.hse.gov.uk/nuclear/tolerability.pdf

– NII's Technical Assessment Guide (TAG) on PSA http://www.hse.gov.uk/foi/internalops/nsd/tech_asst_guides/tast030.pdf

USA:

– Appendix D to NUREG-1860, http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1860/v2/sr1860v2.pdf

– NRC's Severe Accident Policy Statement. (http://www.nrc.gov/reading-rm/doc-collections/commission/policy/50fr32138.pdf)

– Safety Goals for the Operations of Nuclear Power Plants, August 1986 http://www.nrc.gov/reading-rm/doc-collections/commission/policy/51fr30028.pdf

– Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities, August 1995 http://www.nrc.gov/reading-rm/doc-collections/commission/policy/60fr42622.pdf

– Commission White Paper on Risk-Informed and Performance-Based Regulation, March 1999 http://www.nrc.gov/reading-rm/doc-collections/commission/srm/1998/1998-144srm.html

– Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002 http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-174/01-174.pdf

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– Regulatory Guide 1.177 "An Approach for Plant-Specific, Risk-Informed Decision-Making: Technical Specifications," August 1998 http://adamswebsearch2.nrc.gov/idmws/ViewDocByAccession.asp?AccessionNumber=ML003740176

– Regulatory Guide 1.178 "An Approach for Plant-Specific, Risk-Informed Decision-Making for In-service Inspection of Piping," August 1998 http://adamswebsearch2.nrc.gov/idmws/ViewDocByAccession.asp?AccessionNumber=ML003740181

- Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," January 2007 http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-200/01-200r1.pdf

- Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006 http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-201/01-201r1.pdf

- Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," May 2006 http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-205/01-205.pdf

- Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants," June 2007 http://adamswebsearch2.nrc.gov/idmws/doccontent.dll?library=PU_ADAMS^PBNTAD01&ID=071710180

- NUREG-1855 Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making - http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1855/
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- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," June, 2007 http://adamswebsearch2.nrc.gov/idmws/doccontent.dll?library=PU_ADAMS^PBNTAD01&ID=071710059

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," June 2007 http://adamswebsearch2.nrc.gov/idmws/doccontent.dll?library=PU_ADAMS^PBNTAD01&ID=071710155

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- ANSI/ANS-58.23-2007, "Fire PRA Methodology" http://www.ans.org/store/vi-240270
- ANSI/ANS-58.21-2007, "External-Events PRA Methodology" http://www.ans.org/store/vi-240265

IAEA:

- The Role of Probabilistic Safety Assessment and Probabilistic Safety Criteria in Nuclear Power Plant Safety, Safety Series No.106, International Atomic Energy Agency, Vienna, May 1992
- Basic Safety Principles for Nuclear Power Plants. 75-INSAG-3; IAEA Safety Series No. 75-INSAG-3; IAEA; 1988 (superseded by INSAG-12)
- Basic Safety Principles for Nuclear Power Plants. 75-INSAG-3 Rev. 1. INSAG-12; IAEA Safety Series No. 75-INSAG-12. ISBN 92–0–102699–4; IAEA; 1999 http://www-pub.iaea.org/MTCD/publications/PDF/P082_scr.pdf
- Safety Evaluation of Operating Nuclear Power Plants Built to Earlier Standards – A Common Basis for Judgment : INSAG 8 http://www-pub.iaea.org/MTCD/publications/PDF/Pub991e_web.pdf

NKS (Nordic Nuclear Safety Research) / NPSAG (Nordic PSA Group):

- Holmberg, J.-E., Knochenhauer, M.; Probabilistic Safety Goals; Phase 1 – Status and Experiences in Sweden and Finland; Nordic Nuclear Safety Research Report NKS-153, 2006 http://www.nks.org/scripts/pdfsearchbackend.php?Mode=getpdf&id=111010000004762&hash=493b7ab165459d839f95de5415f28b8b
- Holmberg, J.-E., Knochenhauer, M., Probabilistic Safety Goals. Phase 2 Status Report. ; Nordic Nuclear Safety Research Report NKS-172, 2008 http://www.nks.org/scripts/pdfsearchbackend.php?Mode=getpdf&id=111010000004742&hash=9715d9b283ba525f5e91757a50ae4b8b

10. QUESTIONNAIRE

OECD/WGRISK TASK 2006-2

Questionnaire on Probabilistic Risk Criteria For Nuclear Power Plants

Presentation

Task 2006-2 addresses Probabilistic Risk Criteria. This scope is wider than is generally understood as "Safety Goals" and includes

- Lower level criteria, such as regulatory targets for the plant systems or requirements on Fussel-Vessely importance for equipment.
- Other non-prescriptive criteria, such as "orientation values"

For maintaining this questionnaire to a manageable level, the lower level Probabilistic Risk Criteria are excluded from its scope.

They are addressed by the "Performance Indicators" project NEA/CNRA/R(2006)-1.

It is noted that the definition and status of Probabilistic Risk Criteria could be quite different in the different Members States and, within a Member State, between utilities.

We request you consider, when responding to this questionnaire, the wider possible scope of criteria, defining them more precisely in the specific questions.

This questionnaire is divided into three parts:

- Questions 1 and 2, aimed at identifying the Probabilistic Risk Criteria relevant to your country or organisation,
- Questions X.3 to X.9 address a given Probabilistic Risk Criterion. They should be repeated X times, X being the number of Probabilistic Risk Criteria identified in response to question 2.
- Questions 10 to 15 address specific issues. In some cases, several response could apply to a given question, if the response is different according to some specific Probabilistic Risk Criteria

To help filling-in the questionnaire, we have provided, after each question, an example of what could be a response. These examples do not constitute a Member State position.

Identification

Please identify your organisation:

Name: ………………………………………………………………………………………

Address : ……………………………………………………………………………..

……………………………………………………………………………………..

Country : ……………………………………………………………………………..

Are you

- A Regulatory Body
- A Supporting organisation to a regulatory body
- A Utility
- A Vendor

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1. Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1. Society level (qualitative)

For instance
"protect the public and the environment from unreasonable risk"

1.2 Intermediate level (qualitative or quantitative)

For instance
"Avoid long term relocation of the public" (qualitative criterion) or "Ensure the risk for latent radiation-induced cancer is less than 0.1% of other causes of cancer" (quantitative criterion).

1.3 Technical level (quantitative)

For instance
"Core Damage Frequency shall be less than E-5 per reactor year"

2. What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

For instance:

- *Large early release,*
- *Core damage frequency,*
- *Reliability of safety systems*
- *Contribution of individual sequences to the higher level criterion*

Are these criteria the same for existing plants, life extension of existing plants, new builds, and new designs?

For instance

"The Small and Large Release Frequency and Core Damage Frequency are the same for existing plants and life extension and are different for new plants. Systems probability of failure when required to operate is identical for the three cases"

Please answer the following questions for each of the listed criteria, limiting the responses to criteria at or above system level.

For instance

Using the preceding criteria list, answer the following question for the 5 cases

- *LERF for existing plants and life extension*
- *LERF for new plants,*
- *CDF for existing plants and life extension,*
- *CDF for new plants,*
- *Systems reliability: (Do not enter into details for each considered system, and limit information at common issues)*

For Criterion X:

The following examples will consider the CDF criterion

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

For instance

The Core damage Frequency, while not directly related to public safety, has been set to ensure compliance with defence-in-depth, avoiding design of an unsafe reactor with an extremely robust containment.

Please elaborate on the reasons

X.4 What is the definition for this criterion (be precise)?

In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

For instance

Core damage is defined as loss of structural integrity of more than one fuel channel, or
The dose shall be evaluated for the most exposed member of the public, remaining at the most exposed place accessible to the public.

X.5 What is the supporting documentation for this criterion?

Please list and give a short abstract of each key publicly available document supporting the above criterion.

If no publicly available document exists, could you explain briefly how the criterion is supported?

For instance:

- *CMD 2006-89 "Justification for the choice of Core Damage Frequency". This document is public and can be delivered on demand.*
- *Internal discussions comparing the proposed CDF with international practice*

X.6 To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

For instance:

The Small and Large Release Frequency is set at plant level.

Please explain why you chose this applicability.

X.7 How is the criterion expressed?

- Single value
- Band (limit and target)

For instance:

Single value:

*The CDF shall be lower than 10-4/reactor*year*

Band:

*The core damage frequency is desired lower than E-5/ reactor*year and, in any case, should not exceed 10-4/reactor*year*

X.8 How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

For instance:

This criterion is considered as an orientation value. Meeting or not the criterion would result only in prioritizing regulatory requirements for safety improvements.

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

For instance:

The PSA used to assess the reactor Core Damage Frequency shall be full scope (Internal and external events, full power and shutdown operating modes). However, the CNSC can accept use of other methods. – CSNC standard S-294.

End of questions on each Probabilistic Risk Criterion

10. How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

For instance:

The PSA report shall include uncertainty analysis. However, the computed CDF will be compared to the criterion as a best estimate. This is based on the rationale that requiring consideration of uncertainty when comparing the PSA results to the criterion would be equivalent to set a best estimate criterion several times lower than selected.

11. When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

For instance:

The PSA used to compare plant compliance with the Probabilistic Safety Criteria shall be updated every third year, or on the discovering of unexpected deviation form the Safety Analysis assumptions
CNSC standard S-294

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

For instance:

- *If a Probabilistic Risk Criterion is exceeded, analysis of the reasons for this exceedance will be performed to identify the cause. Regulatory action will be set according to the cause of exceedance, ranging from strict requirement to correct the cause in a given time to increased inspections or acceptance of the situation.*
- *When the PSA is updated after plant modifications.*

12. In case of band-defined goals, how is handled the case where the results are inside the band?

For instance:

If the PSA results show the plant performance being between the limit and the target, the licensee will be required to assess ways to improve the situation. Cost-Benefit analysis could be considered to rank the possibilities, the ratio between cost and benefit varying with the situation of the initial results between objective and limit.

13. Have you defined other subsidiary criteria for PSA applications?

For instance:

Instantaneous risk
Accident precursor risk (risk follow-up indicator)

14. What is your experience with Probabilistic Risk Criteria?

Please provide a separate response for every experience you wish to share. These experience statements can address one or several of the following sub-criteria.
We do not expect a complete relation of your experience with probabilistic Risk Criteria, but a limited number of statements addressing the most important.

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

For instance:
Implementation of the Probabilistic Safety Criteria has been generally well received by the utilities. However, before formally accepting the associated values, the utilities have requested information on the sources and rationales.

14.2 On consequences of implementation

Have Probabilistic Risk Criteria lead to system upgrades?

For instance:
The first PSA for Pickering A showed a Core Damage Frequency higher than 10-4/year, exceeding the internal OPG safety limit. Analysis for the cause of this situation led to dramatic improvement in plant safety, now in the lower part of the band for CDF

14.3 On communication with the public

For instance:
The draft safety goals for new plants are such that a plant meeting them would be under the threshold for accidents to be considered in the required Environmental Assessment.

14.4 On interpretation

For instance:
There has been considerable internal discussion for defining what will be Core Damage for CANDU reactors.

Examples of other areas of interest might be beneficial:

15. Your organisation has not defined Probabilistic Risk Criteria:

If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

11. ANSWERS TO THE QUESTIONNAIRE

Response for Belgium

This is a common response from AVN and Tractebel Engineering (TE) on the Belgian practice. Responses with specific issues for either AVN or TE are explicitly indicated.

Identification

Please identify your organisation:

Name: ***AVN (Association Vinçotte Nuclear)***

Address: Avenue du roi 148 ; B-1070 Brussels

Country: Belgium

Are you

- A Regulatory Body □
- A Supporting organisation to a regulatory body: *Yes*
- A Utility □
- A Vendor □

Contact person for AVN: Pieter De Gelder

Name: ***TE (Tractebel Engineering)***

Address: Avenue Ariane 7, boite 1; B-1200 Brussels.

Country: Belgium

Are you

- A Regulatory Body □
- A Supporting organisation to a regulatory body □
- A Utility: *TE is the Architect-Engineering company, performing the PSAs on behalf of Electrabel, the Belgian utility*
- A Vendor □

Contact person for TE: Isabelle Hendrickx

1 Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1 Society level (qualitative)

No.

1.2 Intermediate level (qualitative or quantitative)

No.

1.3 Technical level (quantitative)

Except for the evaluation of the required protection against external events (where the probabilistic criteria of the USNRC SRP section 2.2.3 are used), no probabilistic safety criteria have been defined in Belgium to evaluate the safety of the operating nuclear power plants.

2. What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

None.

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

Not Applicable.

Please answer the following questions for each of the listed criteria, limiting the responses to criteria at or above system level.

The following questions are not applicable given the response above.

10 How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

Not applicable.

11 When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

Not applicable.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

Not applicable.

12 In case of band-defined goals, how is handled the case where the results are inside the band?

Not applicable.

13 Have you defined other subsidiary criteria for PSA applications?

Since several years, AVN has a program on PSA-based Event Analysis (PSAEA). Within that context, events with a CCDP greater than 1E-6 are considered to be precursors, and events with a CCDP greater than 1 E-4 are considered to be important precursors.

14 What is your experience with Probabilistic Risk Criteria?

Not applicable.

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

Not applicable.

14.2 On consequences of implementation
Have Probabilistic Risk Criteria lead to system upgrades?

Not applicable.

14.3 On communication with the public

Not applicable.

14.4 On interpretation

Not applicable.

Examples of other areas of interest might be beneficial:

15 Your organisation has not defined Probabilistic Risk Criteria:

If deliberate, please explain the reasons

Does it expect setting Probabilistic Risk Criteria in the future?

Response for AVN:

When the PSA program was started in Belgium (around 1986) in the framework of the periodic safety review of the existing NPPs, national rulemaking did not contain any probabilistic safety goals or criteria. Further, AVN considered that the objective of the PSAs was (and is) mainly to confirm the robustness of the deterministic design of the NPPs, to identify design and/or operational weaknesses (if any), and to address these weaknesses if necessary. In that way, the deterministic and probabilistic approaches are used in a complementary way. Comparison of, for instance, the estimated CDF with a probabilistic safety goal was not considered essential for success of the PSA application, although IAEA documents have been referenced for orientation values.

For the licensing of new plants, the present rulemaking (Royal Decree of 20/07/2001) requests that a probabilistic safety analysis is part of the safety analysis report to be submitted in the framework of the licensing procedure. This rulemaking does not specify any probabilistic safety goals or criteria. Because of the existing phase-out law in Belgium, discussions on probabilistic safety goals for new NPPs are not foreseen. Anyhow, whenever such a discussion would take place in future in case of a modified context, AVN considers that the FANC (Federal Agency of Nuclear Control) has the prime responsibility in defining probabilistic safety goals or criteria to be applied in a regulatory context.

Response for Canada (CNSC)

Identification

Please identify your organisation:

Name: **Canadian Nuclear Safety Commission**

Address: 280 Slater St. PO Box 1046 – Station B – Ottawa – ON – Canada - K1P 5S9

Country : Canada

Are you

- A Regulatory Body ×
- A Supporting organisation to a regulatory body ☐
- A Utility ☐
- A Vendor ☐

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1. Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1 Society level (qualitative)

(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)

1.2 Intermediate level (qualitative or quantitative)

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)

1.3 Technical level (quantitative)

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} are less than E-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} are less than E-6 per plant year.
iii) Core Damage Frequency
The sum of frequencies of all sequences that can lead to significant core degradation are less than E-5 per plant year.

(Regulatory Document RD-337)

2. What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

- Large release,
- Small Release
- Core damage frequency,
- Reliability of safety systems

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

The above criteria are the same for any plant (existing or future). However, the numerical values for the frequencies are expected to be one decade higher for existing plants.

Please answer the following questions for each of the listed criteria, limiting the responses to criteria at or above system level.

<u>For Criterion X1: Large release frequency</u>

X1.3 What is the reason why you developed this Probabilistic Risk Criterion?

The Large Release Frequency, has been chosen to avoid the large societal impact (permanent relocation of inhabitants) that would result of a release of a long term ground contaminant. Cs_{137} has been chosen as the largest contributor to long term ground contamination. It is assumed that the health effects of public permanent relocation go largely beyond the very effects of radiation.

The numerical value (100 TBq) has been chosen as it would not contaminate a surface significantly larger that the plant exclusion zone to a level higher than defined by the ICRP (ICRP publication 63, "Principles for intervention for protection of the public in a radiological emergency" section C.2.).

X1.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

This criterion is defined as a release of more than 100 TBq of Cs_{137}, all possible causes included
It applies to the average yearly risk.

X1.5 What is the supporting documentation for this criterion?
Please list and give a short abstract of each key publicly available document supporting the above criterion:

"Safety Goals for S-337" CNSC reference 968790 (available on request)

X1.6 To what is the criterion applicable?

- *A single reactor, or*
- *A plant (multiple reactors sharing at least one safety system),*

Several plants in Canada are "Multi-Units", i.e., several reactors share the Turbine Hall and the Containment.

Please explain why you chose this applicability.

As a large release results from containment impairment, for multi-unit plants containment failure resulting from an accident in one unit makes the containment unavailable for the other units.

X1.7 How is the criterion expressed?

A target

The target is E-6 per year:

X1.8 How is this criterion considered?

Legal position on this criterion would depend on the way Regulatory Document RD-337 is treated: If included as a part of a Licence, the criteria is legally bounding. If not included in a licence, it would be an indicator.

X1.9 What is the scope of the analysis used for measuring plant performance against the criterion?

The PSA used to assess the plant large Release Frequency shall be full scope (Internal and external events, full power and shutdown operating modes). However, the CNSC can accept use of other methods. – CSNC standard S-294.

For Criterion X2: Small release frequency

X2.3 What is the reason why you developed this Probabilistic Risk Criterion?

The Small Release Frequency, has been chosen to avoid the societal impact (evacuation of inhabitants) that would result of a release of a short term contaminant. I_{131} *has been chosen as the largest contributor to short term contamination. It is assumed that the health effects of public evacuation go largely beyond the very effects of radiation.*

The numerical value (1000 TBq) has been chosen as it would not contaminate a surface significantly larger that the plant exclusion zone to a level higher than defined by the ICRP (ICRP publication 63, "Principles for intervention for protection of the public in a radiological emergency" table3).

X2.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

This criterion is defined as a release of more than 1000 TBq of I_{131}*, all possible causes included*
It applies to the average yearly risk.

This criterion originates from the CANDU technology, where accidents involving a single fuel channel are possible. In this case, the amount of releases can be significant even if the reactor core can survive.

X2.5 What is the supporting documentation for this criterion?
Please list and give a short abstract of each key publicly available document supporting the above criterion.:

"Safety Goals for S-337" CNSC reference 968790 (available on request)

X2.6 To what is the criterion applicable?

- *A single reactor, or*
- *A plant (multiple reactors sharing at least one safety system),*

Several plants in Canada are "Multi-Units", i.e., several reactors share the Turbine Hall and the Containment.

Please explain why you chose this applicability.

As a small release results from small containment impairment or containment venting, for multi-unit plants containment leakage resulting from an accident in one unit makes the containment unavailable for the other units.

X2.7 How is the criterion expressed?

A target
The target is E-5 per year.

X2.8 How is this criterion considered?

Legal position on this criterion would depend on the way Regulatory Document RD-337 is treated: If included as a part of a Licence, the criteria is legally bounding. If not included in a licence, it would be an indicator.

X2.9 What is the scope of the analysis used for measuring plant performance against the criterion?

The PSA used to assess the plant large Small Frequency shall be full scope (Internal and external events, full power and shutdown operating modes). However, the CNSC can accept use of other methods. – CSNC standard S-294.

For Criterion X3: Core damage frequency

X3.3 What is the reason why you developed this Probabilistic Risk Criterion?

The Core Damage Frequency, has been selected according to the defence-in-depth concept, to avoid the design of a plant whose safety is made only by a very strong containment.

X3.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

This criterion is defined as the failure of more than one fuel channel.
It applies to the average yearly risk.
This criterion originates from the CANDU technology, where accidents involving a single fuel channel are possible. In this case, the reactor core can survive. However, when more than one fuel channel is involved, the probability of calandria deformation leading to a non-coolable geometry is non negligible.

X3.5 What is the supporting documentation for this criterion?
Please list and give a short abstract of each key publicly available document supporting the above criterion.:

"Safety Goals for S-337" CNSC reference 968790 (available on request)

X3.6 To what is the criterion applicable?

– *A single reactor*

X3.7 How is the criterion expressed?

A target

The target is E-5 per year.

X3.8 How is this criterion considered?

Legal position on this criterion would depend on the way Regulatory Document RD-337 is treated: If included as a part of a Licence, the criteria is legally bounding. If not included in a licence, it would be an indicator.

X3.9 What is the scope of the analysis used for measuring plant performance against the criterion?

The PSA used to assess the plant large Small Frequency shall be full scope (Internal and external events, full power and shutdown operating modes). However, the CNSC can accept use of other methods. – CSNC standard S-294.

For Criterion X4: Systems reliability targets

X4.3 What is the reason why you developed this Probabilistic Risk Criterion?

The systems reliability targets were established more than 30 years ago, as a way to meet the high level goal (risk less or equal to other electricity production sources) when computational power was low.

X4.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

Probability that the system does not meet its minimum performance criteria shall be lower than 10^{-3}.

X4.5 What is the supporting documentation for this criterion?
Please list and give a short abstract of each key publicly available document supporting the above criterion.:

Regulatory documents R-7, R-8 and R-9

X4.6 To what is the criterion applicable?

- *A single reactor*

X4.7 How is the criterion expressed?

A single value

X4.8 How is this criterion considered?

It is practically considered as an indicator, the CNSC requiring efforts from the utility if the criterion is not met.

X4.9 What is the scope of the analysis used for measuring plant performance against the criterion?

Internal events only

10. How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

The PSA report shall include uncertainty analysis. However, the computed CDF will be compared to the criterion as a best estimate. This is based on the rationale that requiring consideration of uncertainty when comparing the PSA results to the criterion would be equivalent to set a best estimate criterion with another value.

11. When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

Evaluation of plant performance against the probabilistic criteria is required:

- *Before an operating licence is granted (new builds)*
- *Before a refurbished plant can return to operation,*
- *Every third year for operating plants (CNSC standard S-294)*

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

This is not yet defined.

12. In case of band-defined goals, how is handled the case where the results are inside the band?

It is expected that, if the PSA results show the plant performance being between the target and one decade under the target, the licensee will be required to assess ways to improve the situation. Cost-Benefit analysis could be considered to rank the possibilities, the ratio between cost and benefit varying with the situation of the initial results between objective and limit.

13. Have you defined other subsidiary criteria for PSA applications?

Not formally.

In communicating with utilities, the CNSC has indicated that it will consider first the short cutsets (indicating weaknesses in defence-in-depth) and the cutsets containing more than one human error (possible non-independence of events in one cutset)

14. What is your experience with Probabilistic Risk Criteria?

Please provide a separate response for every experience you wish to share. These experience statements can address one or several of the following sub-criteria.
We do not expect a complete relation of your experience with probabilistic Risk Criteria, but a limited number of statements addressing the most important.

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?

Implementation of the Probabilistic Safety Criteria has been generally well received by the utilities. However, before formally accepting the associated values, the utilities have requested information on the sources and rationales.

- Have you identified practical benefits of setting Probabilistic Risk Criteria?

While the CNSC has not yet implemented Probabilistic Safety Criteria for plants to be refurbished, the relevant utilities have internally defined criteria in line with those the CSNC is considering.
In this case, the utility has analysed and committed several safety improvements on the basis that, without these improvements, they would have been unable to meet some of the criteria.

14.2 On consequences of implementation
Have Probabilistic Risk Criteria lead to system upgrades?

Yes: For instance, The first PSA for Pickering A showed a Core Damage Frequency higher than 10^{-4}/year, exceeding the internal OPG safety limit. Analysis for the cause of this situation led to dramatic improvement in plant safety, now in the lower part of the band for CDF

14.3 On communication with the public

The CNSC has received several applications for new plants. According to the Canadian Law, an Environmental Assessment (EA) is required in this case. However, the utilities have not yet indicated what reactor technology and vendor they intend to build. In this case, commitment to meet the Probabilistic Safety Criteria could be used as a surrogate for the definition of accidents to be considered in the EA..

14.4 On interpretation

Considerable discussion has occurred, both inside the CNSC and with the licensees on the definition of Core Damage (CANDU reactors design is such that there is a fair number of possible damages) and on the difference between a "reactor" and a "plant" for application of the release criteria

Examples of other areas of interest might be beneficial:

15 Your organisation has not defined Probabilistic Risk Criteria:

Not applicable for Canada.

Response for Canada (Hydro-Québec)

Identification

Please identify your organisation:

Name: **Raynald Vaillancourt Reliability Supervisor, Gentilly-2 NPP, Hydro-Quebec**

Address : 4900, boulevard Bécancour, Bécancour, QC, Canada, G9H 3X3

Country : Canada

Are you

A Utility

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1 Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria?

Now, we don't have a PSA but we develop one at technical level for refurbishing the plant.

2 What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria?

Core damage frequency and reliability of safety system

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

The Small and Large Release Frequency and Core Damage Frequency are the same for existing plants and life extension and are different for new plants. Systems probability of failure when required to operate is identical for the three cases

Please answer the following questions for each of the listed criteria, limiting the responses to criteria at or above system level.

For instance,

Using the preceding criteria list, answer the following question for the 5 cases

- *LERF for existing plants and life extension,*
- *LERF for new plants,*
- *CDF for existing plants and life extension,*
- *CDF for new plants,*
- *Systems reliability: (Do not enter into details for each considered system, and limit information at common issues)*

For Criterion X:

The following examples will consider the CDF criterion

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

The Core damage Frequency has been set to ensure compliance with defence-in-depth, avoiding design of an unsafe reactor with an extremely robust containment.

X.4 What is the definition for this criterion (be precise)?

Core damage is defined as loss of structural integrity of more than one fuel channel

X.5 What is the supporting documentation for this criterion?

Internal discussions comparing the proposed CDF with international practice

X.6 To what is the criterion applicable?

A reactor

Please explain why you chose this applicability.

We have only one reactor

X.7 How is the criterion expressed?

*Single value 10-4/reactor*year*

X.8 How is this criterion considered?

Orientation values

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

The PSA used to assess the reactor Core Damage Frequency shall be full scope (Internal and external events, full power and shutdown operating modes)

End of questions on each Probabilistic Risk Criterion

10. How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

The PSA report shall include uncertainty analysis. However, the computed CDF will be compared to the criterion as a best estimate. This is based on the rationale that requiring consideration of uncertainty when comparing the PSA results to the criterion would be equivalent to set a best estimate criterion several times lower than selected.

11. When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

The PSA used to compare plant compliance with the Probabilistic Safety Criteria shall be updated every third year, or on the discovering of unexpected deviation form the Safety Analysis assumptions

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

- *If a Probabilistic Risk Criterion is exceeded, analysis of the reasons for this exceedance will be performed to identify the cause.*
- *When the PSA is updated after plant modifications.*

12 In case of band-defined goals, how is handled the case where the results are inside the band?

13 Have you defined other subsidiary criteria for PSA applications?

No

14 What is your experience with Probabilistic Risk Criteria?

14.1 On implementation

No experience

14.2 On consequences of implementation

No experience

14.3 On communication with the public

The criteria was presented to a public audience for the refurbishing of the station

14.4 On interpretation

There has been considerable internal discussion for defining what will be Core Damage for CANDU reactors.

15 Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

Response for Canada (Ontario Power generation)

Identification

Please identify your organisation: Ontario Power Generation Inc.

Name: **Michael O'Neill**

Address: 889 Brock Road, 6th floor, Pickering, Ontario, L1W 3J2

Country: Canada

Are you

- A Regulatory Body □
- A Supporting organisation to a regulatory body □
- A Utility √
- A Vendor □

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1. Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1 Society level (qualitative)

"to ensure the radiological risks arising from nuclear accidents associated with operation of nuclear reactors will be low in comparison to risks to which the public is normally exposed.

1.2 Intermediate level (qualitative or quantitative)

To ensure the risk for latent radiation-induced cancer to the most exposed individual is less than 1% of other causes of cancer

To limit the frequency of a release of >1% of core inventory of Cs-137 (purpose is to limit the requirement for long-term relocation of population living close to the plant)

To limit the reliance on containment by placing a limit on the severe core damage frequency

1.3 Technical level (quantitative)

Latent health effect shall be less than E-4and should be less than E-5 latent health effects per site per year

Large release frequency shall be less than E-5 and should be less than E-6 per reactor year

Severe core damage frequency shall be less than E-4and should be less than E-5 per reactor year"

Basis for and experience with use of probabilistic safety goals in OPG is described in "Dinnie, K.S., Experience with the Application of Risk-Based Safety Goals, 25th Annual Conference of the Canadian Nuclear Society, June 2004." Specific guidance of using "risk" information and criteria have been developed in OPG Nuclear Safety Policy, N-POL-0001 R00 and OPG Risk and Reliability Program, N-PROG-RA-0016 R05.

2. What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

1) *Latent health effect*
2) *Large release frequency,*
3) *Severe core damage frequency,*
4) *Unavailability targets for risk-significant systems (typically 10-15 systems)*
5) *Instantaneous risk limits on severe core damage and large release*

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

Current safety goals in OPG governance were developed by the utility for existing plants. The regulator (CNSC) is in the process of developing safety goals for new plants but these have not yet been published.

Please answer the following questions for each of the listed criteria, limiting the responses to criteria at or above system level.

3. What is the reason why you developed this Probabilistic Risk Criterion?

3.1 Multi-unit CANDU stations with shared containment have relatively higher containment leakage than single-unit station, that could result in exposure of the public to low levels of radiation release from any accident involving fuel failures. A pressure tube reactor has identifiable accident sequence end-states that involve limited release of fission products to containment in absence of severe core damage sequences. There is a need to have in place an overall measure of risk that includes the contribution of accidents that do not involve severe core damage.

3.2 To limit the frequency of any requirement for prolonged relocation of the public living close to the site boundary to a very low level

3.3 To provide defence in depth by limiting the reliance on containment to protect the public. Also to provide a basis for comparison with the results of other PSAs

3.4 To ensure that the availability of plant systems is maintained consistent with the plant PSA as required by CNSC regulatory standard S-98.

3.5 To ensure that plant operating configurations do not pose an unacceptably high risk even for a short period of time, and that compensatory actions are taken where necessary to achieve this objective.

4 What is the definition for this criterion (be precise)?

4.1 The frequency of Latent Effects corresponds to the increase in the probability of fatality or serious irreversible injury arising from the release of radiation to the environment due to operation of a nuclear generating station when averaged over one year. The calculation accounts for releases due to normal operation and those due to postulated accident events. It is based on a hypothetical individual living at a fixed location close to the facility boundary 24 hours per day, 365 days per year and is calculated conservatively as a rate of exposure in Sv/yr to the individual multiplied by the probability of a latent health effect/Sv.

4.2 The frequency of Large Off-Site Release is the sum of the mean frequencies of events that can lead to the release of greater than 1 percent of the equilibrium core inventory of Cs-137 to the environment due to the operation of a nuclear reactor when averaged over a one year period. .

4.3 The Severe Core Damage Frequency is the sum of the mean frequencies of events due to operation of a nuclear reactor that can lead to failure of both fuel and fuel channels when averaged over one year.

4.4 Unavailability is the probability that, at some future time, a poised system will fail to adequately respond to meet its credited safety function. This is calculated using representative component failure rate data and testing frequencies in a fault tree based unavailability model.

4.5 Instantaneous risk is the plant risk assessed for a given configuration assuming zero-maintenance.

5 What is the supporting documentation for this criterion?

There are no publicly-available basis documents; guidance is prescribed by internal documents.

If no publicly available document exists, could you explain briefly how the criterion is supported?

The safety goals are supported by OPG Governance, Nuclear Program N-PROG-RA-0016, Risk and Reliability Program and associated Nuclear governing documents. The purpose of the program is to establish a framework for the development and use of Probabilistic Risk Assessment (PRA) at Ontario Power Generation (OPG), Nuclear as a means to manage radiological risks from nuclear accidents and to contribute to safe operation of OPG Nuclear reactors. Program elements have been developed to meet the intent of OPG Nuclear Safety Policy and CNSC Standards S99, S98 and S294. The Risk and Reliability Program consists of Safety Goals, station-specific PRAs, associated risk models, unavailability models of systems important to safety and software applications,

6 To what is the criterion applicable?

6.1 The criterion applies to the station as a whole and therefore includes the contribution from all reactor units that constitute the station. A station consists of multiple units of identical design, operated from a common control room and for which performance data is collected and reported.

6.2 The criterion applies to individual reactor units. The safety goal acts a surrogate for early fatality and offsite impacts other than health effects, and serves as an international benchmark when estimated at the reactor level.

6.3 The criterion applies to individual reactor units. The safety goal acts a means of limiting the degree to which containment is relied upon to control public health risk and serves as an international benchmark when estimated at the reactor level.

6.4 The criteria apply to systems of individual reactor units, except where the system provides a shared function (containment, emergency power etc.). This is used as the basis for annual performance reporting.

6.5 The criteria apply to any actual unit operating configuration; especially where equipment credited in the PSA has failed or is taken out of service beyond the maintenance and test outage times assumed in the PRA.

7 How is the criterion expressed?

7.1 Band (limit and target), compared to mean estimated value
7.2 Band (limit and target), compared to mean estimated value
7.3 Band (limit and target), compared to mean estimated value
7.4 Single value compared to mean estimated value
7.5 Single value compared to mean estimated value

8 How is this criterion considered?

8.1 The limit is a strict but not legally-binding value that should not normally be exceeded. The goal is a target, above which design improvements should be considered taking into account cost-benefit considerations.

8.2 The limit is a strict but not legally-binding value that should not normally be exceeded. The goal is a target, above which design improvements should be considered taking into account cost-benefit considerations.

8.3 The limit is a strict but not legally-binding value that should not normally be exceeded. The goal is a target, above which design improvements should be considered taking into account cost-benefit considerations.

8.4 The unavailability targets are considered as performance measures, not meeting the criterion would result in a report to the regulator and development of a plan to establish compliance.

8.5 The instantaneous risk limits are used as part of plant operational risk management to indicate the need for compensatory actions up to and including reactor shutdown.

9 What is the scope of the analysis used for measuring plant performance against the criterion?

9.1–9.3 The PSAs used to assess the plants against safety goals are currently limited to internal events (full power and shutdown operating modes). Recent CNSC requirements (S-294) require full scope PSAs (including external events). However, the CNSC has prescribed that for external events use of other methods to demonstrate low risk may be acceptable, although no such alternative methods have yet been established formally.

9.4 The system unavailability models assess the condition of being available (e.g. within minimum performance standards) and is equated to the probability that a system is in the required state, or able to change to the required state on demand. As a result most of the mission-related failures that may appear in the PSA system model are excluded from the unavailability model

9.5 The operational risk models use a zero-maintenance version of the PSA internal events model.

10 How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

The PSA report is required to include uncertainty analysis. However, the computed CDF will be compared to the criteria as a best estimate. This is based on the rationale that the safety goals are derived and applied in a conservative manner (e.g., the hypothetical most at-risk individual) and that further conservatism in applying the results of PSA would be unreasonable and unnecessary.

11 When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

The OPG governance requires that the PSA used to compare plant compliance with the Probabilistic Safety Criteria be updated every three years or when warranted by a major design change.

OPG governance requires that outage planning and on-line maintenance work plan is risk-informed using risk monitors developed based on plant specific PSA.

For instantaneous risk, an assessment by the operational risk model is performed for every non-standard plant configuration entered.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

If a Probabilistic Risk Criterion is exceeded, the reason is likely to be a discovery issue related to equipment unable to perform as assumed in the PSA, or new analysis or experimental information that affects the failure criteria in the PSA. Compensatory actions must be taken to bring the risk below the limits or the plant shut down while repairs or design changes are implemented. Processes to deal with operability or discovery issues are used for this purpose.

If significant changes to design or failure criteria are required, these will usually be implemented in the PSA as part of the recovery plan to confirm compliance with safety goals.

12 In case of band-defined goals, how is handled the case where the results are inside the band?

If the PSA results show the plant performance being between the limit and the target, the PSA will identify the major contributors and options to further reduce the risk. Implementation of any changes deemed cost-effective will be considered but is non-mandatory. Such changes are more likely to be implemented to address system unavailability in excess of limits.

13 Have you defined other subsidiary criteria for PSA applications?

Instantaneous risk as discussed above and associated risk increments used as part of operational risk management to guide the need for compensatory measures, set allowable operating times and level of authorization for continued operation.

14 What is your experience with Probabilistic Risk Criteria?

Please provide a separate response for every experience you wish to share. These experience statements can address one or several of the following sub-criteria.

We do not expect a complete relation of your experience with probabilistic Risk Criteria, but a limited number of statements addressing the most important.

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

The initial set of Probabilistic Safety Criteria proposed for OPG included 5 goals: early and delayed fatality, large and severe release (>10% Cs release) as well as severe core damage frequency. Experience has indicated that early fatality and severe release are adequately controlled by meeting the large release frequency goal.

Safety goals have been accepted as representing a more comprehensive and realistic assessment of safety adequacy when used in the management of plant operations.

14.2 On consequences of implementation

The first PSA for Pickering A showed a Core Damage Frequency higher than 10-4/year, exceeding the internal OPG safety limit. Analysis for the cause of this situation and plant modifications led to a reduction in severe core damage frequency, now in the lower part of the band.

Later PSAs have identified the importance of systems to reduce the impact of high energy pipe ruptures in the powerhouse of multi-unit stations. Changes to improve the reliability of these systems have been implemented.

14.3 On communication with the public

The fact that the OPG plants meet safety goals (and that these safety goals are generally consistent with international practice) has been cited to support the adequacy of plant safety at public hearings and public inquiries related to nuclear power. Interveners have pointed out that the current PSAs do not currently include fire and external events in their scope.

14.4 On interpretation

There is not an agreed definition between OPG and regulator as to what constitutes severe core damage. The industry defines it as the onset of loss of core structural integrity (i.e., collapse of fuel channels) whereas the CNSC uses "failure" of more than one channel. This could be important if event sequences were identified in which more than one channel was physically damaged but overall core cooling capability was maintained.

Examples of other areas of interest might be beneficial:

15 Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

Not Applicable.

IAEA
Atoms for Peace: The First Half Century
1957–2007

J4-TM-29260

Technical Meeting on

DEVELOPMENT OF A SAFETY GUIDE ON LEVEL 1 PROBABILISTIC SAFETY ASSESSMENT (PSA) AND APPLICATIONS FOR NUCLEAR REACTORS

IAEA, Vienna, 20 – 24 November 2006

Questionnaire on National Safety Goals and PSA Regulations

Item of Inquiry	YOUR Reply	Comments
Country:	Czech Republic	
Is PSA required for license?		
Is PSA required as part of PSR?	Yes	
Do you have numerical risk limits established in the national regulatory documents? If yes, please give the title(s) of the documents:	No, but INSAG-3 objective is expected to be met	
Are the values formulated as safety TARGETS/GOALS or CRITERIA (i.e. to strictly comply with?):		
What is the nature of the estimate (i.e. point estimate, mean, median, other)?		
What is the scope of PSA for which safety goals/criteria are specified?		
Please give the numerical values for:		
CDF:		
LERF:		
Level-2 (if any):		
Values related to risk for population (i.e. outcome of a Level-3 PSA) – please specify what value is meant:		
Other (please specify what are the values):		
Do you have requirements for uncertainty bounds for the safety goals/criteria? If yes, what are they?		
Do you have safety goals/criteria for a multiple unit site? If yes please specify this:		

Response for Finland (STUK)

Identification

Please identify your organisation:

Name: **Radiation and Nuclear Safety Authority, STUK**

Address: Laippatie 4 / P.O. BOX 14, FI-00881 Helsinki, Finland

Country: Finland

Are you

- A Regulatory Body ☑
- A Supporting organisation to a regulatory body ☐
- A Utility ☐
- A Vendor ☐

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1 Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1 Society level (qualitative)

Nuclear Energy Act, 11 Dec. 1987 (990/1991)

Section 6, Safety

The use of nuclear energy must be safe; it shall not cause injury to people, or damage to the environment or property.

1.2 Intermediate level (qualitative or quantitative)

Decision of the Council of State on the general regulations for the safety of nuclear power plants, 14 Feb. 1991 (395/1991)

Section 3, General objective

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.

Section 9, Limit for normal operation

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.

Section 10, Limit for an anticipated operational transient

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.

Section 11, Limit for a postulated accident

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.

Section 12, Limit for a severe accident

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 caesium-isotopes shall not cause, in the long term, starting three months from the accident, a hazard greater than would arise from a caesium 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.

Guide YVL 1.0, Ch. 3

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.

Note: Finnish nuclear legislation and the regulatory YVL Guides issued by STUK are available on STUK's web site www.stuk.fi (In English > Publications)

1.3 Technical level (quantitative)

Guide YVL 2.8

Ch. 2.1, During the design and construction of a NPP

The following numerical design objectives cover the whole nuclear power plant:

> *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 (LRF).
>
> *Should substantial risk factors not recognised earlier appear during operation, the licensee shall upgrade the safety of the plant.*
>
> *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.*

Ch. 2.2, second paragraph requires balanced risk profile:

The risks associated with various initiators and accident sequences, taking into account their uncertainties, shall be compared with the numerical safety objectives and with each other in order to ensure that no single or few prevailing risk factors will stay at the plant.

2. What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

See 1.3.

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

For new NPPs, the fulfilment of quantitative criteria shall be demonstrated during design, construction and operating phase.

For the currently operating plants (Loviisa 1&2, Olkiluoto 1&2), no quantitative criteria are applied. However, the principle of continuous safety enhancement is observed and several plant modifications have been implemented based on PSA.

For Criterion X:

Answers given below consider both the CDF and LRF criterion.

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

In the 1980's, there was a need to develop regulatory guides for licensing of a new NPP. In this context, first numerical safety objectives were defined including requirements on performing a design phase PSA. After that, the definitions for the numerical safety objectives have been updated a few times to better reflect the conception of qualities of PSA and conceivable reactor designs and their safety features.

X.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

Numerical safety objectives of Guide YVL 2.8 apply to the mean of annual risk distribution, uncertainty analysis is required.

Core damage: 1204 °C which leads to a fuel cladding failure. This is an old U.S. design criterion for LOCA. 1204 °C corresponds to failed core cooling.

Release: the target value defined in section 12 of the Government Resolution 359/1991:

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 caesium-isotopes shall not cause, in the long term, starting three months from the accident, a hazard greater than would arise from a caesium release corresponding to the above-mentioned limit.

X.5 What is the supporting documentation for this criterion?

The core damage definition is an old U.S. design criterion for LOCA. 1204 °C corresponds to failed core cooling.

The Cs-137 criterion is related to restrictions on the use of land. The value100 TBq has its origin in a Swedish study for the Barsebäck NPP in the FILTRA-project in the 1980's. If 100 TBq spreads to an area of 100 km^2, the fall-out would be a small. The Finnish definition of a large release was originally implemented as a deterministic criterion for severe accident management strategy. For example, in Loviisa NPP, 100 TBq corresponds to 0.1% of the core inventory. In U.S.A. and Germany, 1% criterion is applied.

The other part of the release criterion is acute health effects. If the Cs-137 criterion is satisfied, there should be no acute health effects. Therefore the Cs-137 criterion is regarded sufficient for the analyses.

X.6 To what is the criterion applicable?

The CDF and LRF criteria are applied to nuclear reactors.

According to the draft for updated regulatory guide, the large release frequency criterion is applied to all nuclear fuel at the site, i.e., reactor, spent fuel pool, intermediate spent fuel storage.

Please explain why you chose this applicability.

The updated guide should cover all potentially significant release sources.

X.7 How is the criterion expressed?

The mean value of CDF shall be less than 1E–5/a, for new NPPs.
The mean value of LRF (100 TBq, Cs-137)shall be smaller than 5E–7/a.

X.8 How is this criterion considered?

If these criteria are not met for a new NPP, the design has to be modified.

The quantitative criteria are expressed in Guide YVL 2.8 issued by STUK. Exemptions from the requirements of YVL Guides can be made by a decision of STUK's Director General under special circumstances. The general "Safety As High As Reasonably Achieved" would be taken into consideration.

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

Full scope Level and Level 2 PSA: all initiating events (internal IEs, fires, floods, external hazards including seismic), all operating modes.

10. How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

The computed CDF and LRF will be compared to the criterion as mean values. The PSA report shall include uncertainty and sensitivity analysis. These analyses are considered in the regulatory review, but no specific requirements have been defined.

11. When and how do Probabilistic Risk Criteria apply?

11.1. When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

For new plants, the fulfilment of the Probabilistic Risk Criteria is assessed in the review of construction licence and operating licence applications. In addition, PSA shall be kept up-to-date during operation and it is reviewed by STUK.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

A new NPP shall fulfil the target values. If new significant risks are identified during licensing process or operation, the plant design has to be modified to meet the criteria.

12. In case of band-defined goals, how is handled the case where the results are inside the band?

N/A

13. Have you defined other subsidiary criteria for PSA applications?

<u>Guide YVL 2.8., ch. 2.2, second paragraph requires balanced risk profile:</u>

The risks associated with various initiators and accident sequences, taking into account their uncertainties, shall be compared with the numerical safety objectives and with each other in order to ensure that no single or few prevailing risk factors will stay at the plant.

14. What is your experience with Probabilistic Risk Criteria?

Please provide a separate response for every experience you wish to share. These experience statements can address one or several of the following sub-criteria.

We do not expect a complete relation of your experience with probabilistic Risk Criteria, but a limited number of statements addressing the most important.

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

The emphasis is on comprehensive, systematic, best estimate analysis, using state-of-the-art methodology. For new reactor designs, the fulfilment of quantitative criteria should be plausible with good margin.

The OL3 NPP project is the first time when these criteria are used in practice. Experience so far, has shown that the criteria are practicable and facilitate comparison of various types of risks. These criteria complement and support deterministic design

14.2 On consequences of implementation

Safety goals affect also the quality of PSA by requiring more detailed modelling of some issues. Conservatism can make the numbers look too bad. Safety goals motivate to make better analyses.

Safety goals are mandatory requirements for a new NPP. They are evaluated in the decision making process.

For old NPPs, safety goals set an ambitious goal for safety improvements. Safety goals mean also that plant changes and exemptions from licensing conditions need to be assessed numerically.

Have Probabilistic Risk Criteria lead to system upgrades?

A thorough, full scope PSA study is required to adequately demonstrate the fulfilment of quantitative criteria. Several PSA based design and plant modifications have been implemented in Finnish NPPs, including OL3.

14.3 On communication with the public

So far, quantitative risk estimates have not had an important role in public discussion or in the media.

14.4 On interpretation

The Finnish criterion for release is very stringent (100 TBq Cs-137). For instance, for Olkiluoto 3 NPP, it corresponds to 0,015% of core inventory, which means that the containment must be tight.

Examples of other areas of interest might be beneficial:

Risk analysis for nuclear waste facilities and decommissioning of NPPs is under discussion.

15 Your organisation has not defined Probabilistic Risk Criteria:

N/A

Response for Finland (FORTUM)

Identification

Please identify your organisation:

Name: **Fortum Power and Heat**

Address : POB 100, FI-00048 FORTUM, Finland

Country : Finland

Are you

- A Regulatory Body ☐
- A Supporting organisation to a regulatory body ☐
- A Utility ☑
- A Vendor ☐

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1 Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1 Society level (qualitative)

Not in addition to regulation?

1.2 Intermediate level (qualitative or quantitative)

Not in addition to regulation?

1.3 Technical level (quantitative)

Unofficial goal 1E-4/yr for core damage risk.

For release frequency, a goal can be derived 1E-5/yr based on 10% value of the CDF applied in the planning of SAM-strategies (severe accident management).

In plant modifications, the goal is that the new plant must be better than the old one.

There are economic criteria for justification of safety improvements, i.e., price of ΔCDF and ΔLERF. Criteria could be used also for justification of plant modifications that can increase core damage risk. In practice, compensating measures are often applied, especially if the risk level is not below the limit (1E-4/a CDF). Criteria have been defined based on estimation of value of core damage and large release.

2. What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

See 1.3

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

No.

For new plants, Guide YVL 2.8 would be applied.

Please answer the following questions for each of the listed criteria, limiting the responses to criteria at or above system level.

For Criterion X:

The following examples will consider the CDF criterion

The following answers will consider the CDF and LERF criteria.

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

Please elaborate on the reasons

PSA has been used in decision making on plant modifications since the first version of PSA completed 1989 for Loviisa NPP. When using PSA for decision making, criteria are needed. Therefore guidelines for using PSA in decision making haves been developed.

X.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

Core damage: Core melt fuel cladding temperature ~1200 °C. In practical analysis work this no precise definition is not needed because the situation is either safe or clearly melting core.

Release: There is a real spectrum of releases, which was used in the economical criterion. For average LERF the value is larger than YVL 2.8 (the Government Resolution) limit 100 TBq Cesium-137. In practical analysis work no precise definition needed. The definition of LERF should be different from 100 TBq, since international studies show that it is far too low, and the government regulation never defined LERF. It only stated that probability of >100 TBq release should be very small, without defining "small" either.

X.5 What is the supporting documentation for this criterion?

Please list and give a short abstract of each key publicly available document supporting the above criterion.

If no publicly available document exists, could you explain briefly how the criterion is supported?

IAEA; Basic Safety Principles for Nuclear Power Plants. 75-INSAG-3; IAEA Safety Series No. 75-INSAG-3; IAEA; 1988

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

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

USNRC; An approach for using probabilistic risk assessment in risk-informed decisions in plant-specific changes to the licensing basis; Reg.Guide 1.174; USNRC; 2002

USNRC; An Approach for Plant-Specific, Risk-Informed Decision making: In-service Testing (ML003740149) (Issued with SRP Chapter 3.9.7) (Draft DG-1062, ML003739158, issued 06/1997); Reg.Guide 1.175; USNRC; 1998

USNRC; An Approach for Plant-Specific, Risk-Informed Decision making: Graded Quality Assurance (ML003740172) (Draft DG-1064 , ML003739212, issued 06/1997); Reg.Guide 1.176; USNRC; 1998

USNRC; An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications (ML003740176) (Issued with SRP Chapter 16.1) (Draft DG-1065, ML003739150, issued 06/1997); Reg.Guide 1.177; USNRC; 1998

USNRC; An Approach for Plant-Specific Risk-Informed Decision making for In-service Inspection of Piping (9/98, ML003740181) (Issued with SRP Chapter 3.9.8) (Draft DG-1063, ML003739154, issued 10/1997) (Revision 1, ML032510128, issued 09/2003) (SRP, ML03251

Vaurio, J.K.; Safety-related decision making at a nuclear power plant; Nuclear Engineering and Design 185 (1998) pp 335–345; Imatran Voima Oy, Loviisa Power Station; 1998

Vaurio, J.K.; Risk-informed decision making at Loviisa NPP; Proceedings of the NKS/SOS-2 Seminar on Risk Informed Principles. Bergendal,Sweden, 13 – 14 April 1999. Ed. U. Pulkkinen and K. Simola. Nordic Nuclear Safety Research (NKS).NKS-6, ISBN 87-7893-05

X.6 To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

A reactor

Please explain why you chose this applicability.

?

X.7 How is the criterion expressed?

- Single value
- Band (limit and target)

Single values

X.8 How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

Targets. Aim is to fulfil the safety goal. Investments have been made to improve safety, major plant design changes, new safety systems have been built.

Loviisa NPP has also used an economical criterion. Safety improvements have been compared with the value of the plant, value of production and cost of improvement.

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

Full scope PSA

10. How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

Uncertainty and sensitivity analyses are made. The computed CDF and LERF will be compared to the criterion as a best estimate.

11. When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

Evaluation is made in connection to plant modifications and PSA updates.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

Before this year the CDF has been above the CDF goal. Therefore mainly risk decreasing proposals have been handled.

12. In case of band-defined goals, how is handled the case where the results are inside the band?

N/A

13. Have you defined other subsidiary criteria for PSA applications?

In exemptions from Tech.Spec., small risk increases, about 1%, could be allowed.

14. What is your experience with Probabilistic Risk Criteria?

Please provide a separate response for every experience you wish to share. These experience statements can address one or several of the following sub-criteria.

We do not expect a complete relation of your experience with probabilistic Risk Criteria, but a limited number of statements addressing the most important.

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

Aim is to achieve as realistic PSA-result as possible. It is worth analysing as precisely as possible. PSA and the goals have been used mainly to design and prioritise safety modifications, and to justify temporary configurations. In some cases PSA helped to avoid unnecessary changes suggested on deterministic basis. PSA has gained acceptance, although in some case a later PSA update has shown that earlier modifications were not quite enough or optimal. But no reversals have been necessary, and The safety authorities have been convinced about the quality and use of PSA. In recent years PSA has become more important for decision making at STUK (YVL 2.8), contrary to less emphasis and dependency on initiatives and management on the part of the utility.

14.2 On consequences of implementation
Have Probabilistic Risk Criteria lead to system upgrades?

Expansions of PSA use to lead to need for safety improvements.

14.3 On communication with the public

?

14.4 On interpretation

Definition for large release is unreasonable. It is very difficult to fulfil even by a new design.

Numerical objectives defined in Guide YVL-2.8 are too hard achieve.

Examples of other areas of interest might be beneficial:

15. Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

N/A

Response for Finland (TVO)

Identification

Please identify your organisation:

Name: **Teollisuuden Voima Oy**

Address: FI-27160 Olkiluoto, Finland

Country: Finland

Are you

- A Regulatory Body ☐
- A Supporting organisation to a regulatory body ☐
- A Utility ☑
- A Vendor ☐

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1. Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1. Society level (qualitative)

?

1.2 Intermediate level (qualitative or quantitative)

?

1.3 Technical level (quantitative)

For the operating units Olkiluoto 1/2, there no formal average CDF or LERF targets. Informally, the numerical objectives of STUK's Guide YVL 2.8 (CDF < 1E-5/yr, LERF (>100 TBq Cs137) < 5E-7/yr) are desired target values.

TVO has an internal guide for plant modifications based on STUK's Guide YVL-2.8 and NRC regulatory guide 1.174:

- *Permanent design change may not increase core damage frequency and frequency for radioactive release more than 1% from the target value. Target values are same as in Guide YVL 2.8. Higher risk increase must be justified.*

- *Temporary work (done only once in plant lifetime) may not cause more than 40% risk increase impact compared to the annual target value (= YVL-2.8). 40% criterion comes from the planned lifetime for the plant, i.e., 40 years.*

For an exemption from Tech.Spec. STUK requires a PSA-evaluation. TVO applies 1% risk increase criterion. Higher risk increases must be justified.

For the new unit Olkiluoto 3, which is under construction, the guide YVL-2.8 is applied.

The following numerical design objectives cover the whole nuclear power plant:

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.

2. What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

See 1.3

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

No, See 1.3.

Please answer the following questions for each of the listed criteria, limiting the responses to criteria at or above system level.

For Criterion X:

The following answers will consider PSA application criteria (see 1.3) used for an operating units.

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

Please elaborate on the reasons

TVO started development of numerical criteria while developing PSA applications in the early 1990s. The first applications were planning of preventive maintenance during power operation, optimisation of allowed outage times, and test interval optimisation. Very tight and fuzzy criteria were used at the beginning. Then, the U.S.NRC's regulatory guides 1.174-1.178 were published, and parts of them were adopted. The criteria are now based on results from PSA, numerical objectives defined in Guide YVL-2.8 and the U.S.NRC's regulatory guides 1.174-1.178. They are formulated in an internal PSA guide to be used in safety-related decision making.

X.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

Core damage: 1204°C. In practical analyses, hottest point temperatures 600–900 °C have been used. Release: YVL 2.8. limit 100 TBq Cesium-137. Regarding acute health effects, iodine (and undelayed noble gases) shall not be released, and the release of Iodine shall be small (not accurately defined).

X.5 What is the supporting documentation for this criterion?

Please list and give a short abstract of each key publicly available document supporting the above criterion.

If no publicly available document exists, could you explain briefly how the criterion is supported?

VNP 395/91, Guide YVL-2.8, NRC Reg.guide 1.174

X.6 To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

A reactor

Please explain why you chose this applicability.

?The identical plant units, including nuclear island and turbine island, are completely separated. They have no common safety related systems.

X.7 How is the criterion expressed?

- Single value
- Band (limit and target)

Single value

X.8 How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

Target values.

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

Full scope.

10 How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

Normally, uncertainty and sensitivity analyses are included in a PSA application. In the comparison to a criterion, however, best estimates are used.

11 When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

Application specific criteria are applied in connection to each application.

Changes in CDF and LERF are monitored in connection to each plant modification and PSA model update, which is done several times a year, e.g. after finalisation of each plant modification affecting PSA results..

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

Acceptance of exceedance of a criterion requires a justification. We try to prevent the exceedance of the Probabilistic Risk Criteria by analyzing the risk change involved in plant modifications in advance, before implementation on the plant. Core damage risk involved in safety related events reported to the authority is calculated using conditional core damage frequency. INES classification of the event is supported with PSA calculations according to the internal PSA guide.

12. In case of band-defined goals, how is handled the case where the results are inside the band?

N/A

13. Have you defined other subsidiary criteria for PSA applications?

?No criteria defined. However, all safety related events – including accident precursors - are analyzed also by means of PSA, if they are reported to the authority. See 11.2.

14. What is your experience with Probabilistic Risk Criteria?
Please provide a separate response for every experience you wish to share. These experience statements can address one or several of the following sub-criteria.

We do not expect a complete relation of your experience with probabilistic Risk Criteria, but a limited number of statements addressing the most important.

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

In-house, PSA results affects decision making. PSA-criterion is accounted in decision making. Risk comparisons are made in shutdown decision making. 1% criterion is well-known at the plant.

In communication with the safety authority, PSA criteria are directive for OL1/OL2 operating units. and mandatory for OL3. For OL3, must be fulfilled with a marginal.

14.2 On consequences of implementation

Have Probabilistic Risk Criteria lead to system upgrades?

The probabilistic criteria applied to operating plants are not strict, which allows more flexible handling of risk. Almost always, when extending PSA, new risks have been identified and plant modifications done, if the risk is "remarkable" compared with the old risks. The criteria have affected on the plant modifications by motivating to select the most effectively risk decreasing alternative. Also suggested modifications that would have increased the CDF, have been prohibited.

14.3 On communication with the public

?Not used since late eighties, when very negative feedback was experienced.

14.4 On interpretation

Definition for large release could be developed. The reasonableness of 100 TBq limit has been discussed.

Examples of other areas of interest might be beneficial:

15. Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons.
Does it expect setting Probabilistic Risk Criteria in the future?

N/A

Draft Answer from France (IRSN)

Introduction

Although the safety of French PWRs relies firstly on deterministic principles, the probabilistic approach was considered since the 1970s as an important complement for safety analysis.

PSA is not a regulatory requirement, and probabilistic studies have been carried out without any formal requirement from the Safety Authority. However PSA results led to several specific requirements for improving the plants design or operation. Due to the interest of the PSA applications, the role of PSA in safety analysis becomes more and more important.

In principle the French Nuclear Safety Authority (ASN) considers that Probabilistic Risk criteria have not to be defined by any regulatory document. The reasoning is developed in the answer to question 15. The main reason is that the aim of ASN is always to improve safety (not only to maintain it), and if the compliance with Probabilistic Criteria was demonstrated, this could lead to a low motivation for supplementary safety improvements.

However the increasing use of PSA as a necessary complement to the traditional deterministic safety analysis led to define, case by case, some absolute or relative Probabilistic orientation values.

These particular examples are the basis of the answers to the questionnaire, because they can bring some useful information to the task group. But once more in France there are NO PROBABILISTIC RISK CRITERIA.

In particular it has to be noted that ASN required a Basic Safety Rule relating to the acceptable methods for PSA use and development. This PSA Basic Safety Rule (RFS-2002/1 published in 2002) does not give any numerical criterion, but indicates that case by case orientation values could be defined.

Identification

Please identify your organisation:

Name: **IRSN**

Address: BP 17–92262 Fontenay-aux-Roses Cedex

Country: France

Are you

- A Regulatory Body ☐
- A Supporting organisation to a regulatory body ☐**x**
- A Utility ☐
- A Vendor ☐

1. Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

General level

In 1977, during the examination of the major technical options for the 1300 MWe plants, the Safety Authority set an overall probabilistic objective expressed as follows:

«In general terms, the design of a plant which includes a pressurised water nuclear reactor should be such that the overall probability that the plant could be the source of unacceptable consequences should not exceed E-6 per year .

This implies that, whenever a probabilistic approach is used to assess whether a family of events must be taken into account in the reactor design, the family must effectively be taken into account if its probability to lead to unacceptable consequences exceeds 1 E-7 per year..»

It has to be noted that :

a. *The overall objective is stipulated in terms of « unacceptable consequences », but these « unacceptable consequences » are not specified by legislation or regulation.*

b. *The E-6 value is an « objective » for a PWR plant, but EDF is not required to demonstrate that this objective has been achieved.*

c. *The 1 E-7 per year value is more practical for operational uses and is used in the approach to determine if the risks generated by external events have to be considered in the design ; for example, the value is applied to families of air crash events.*

1.2 Intermediate level

During the PSA applications, the acceptability of the utility proposals are not based on formal criteria, but some orientation values (relative or absolute) can be given case by case. Some examples are the following:

- *A probabilistic target of E-6 per reactor*year for the CMF related to shutdown conditions was set by the Safety Authority (considering in particular that during shutdown containment integrity is not guaranteed).*
- *In the framework of each of the French PWR series Periodic Safety Reviews, the level 1 PSA is updated and the overall CMF compared to the previous value. Moreover each functional sequence of the PSA with a CMF contribution > 10-8 per year was analysed, in order to investigate the interest and the feasibility of plant improvements. Particular attention was paid to sequences potentially resulting in early containment failure.*
- *A probabilistic analysis of operating events is carried out in France since 1994. The aim of the quantitative analysis is to assess the risk increase (in term of core damage probability) due to the incident. An incident is considered as a precursor if the risk increase is higher than E-6 per event. The Safety Authority required to take particular measures if the risk increase is higher than 10-4, and to assess the benefit of these measures.*
- *For the French-German project EPR (European Pressurised Reactor), the French and German Safety Authorities gave the following very general probabilistic objectives:*
  - *a reduced CMF compared to existing plants*
  - *« practical elimination » of sequences with potential for large early releases.*

In order to fulfill these objectives, the designers have proposed probabilistic safety objectives as orientation values which give useful guidance but are not strict limits and do not correspond to a requirement of the Safety Authorities. Examples of these probabilistic objectives are a value of E-6 per year for the CMF due to internal events, respectively for power states and for shutdown states.

Generally speaking, the French Safety Authority (ASN) considers PSA as a fruitful tool, notably for improving the safety of French PWRs by identifying where design and operating modifications are worthwhile, and for ranking problems in order of importance. However, they are not in favour of setting probabilistic criteria.

As the French Safety Authority's aim is to improve safety, they believe that the use of PSAs for relative consideration is more efficient than the use of absolute criteria.

2. What technical level Probabilistic Risk Criteria have you developed?

2.1 and 2.2: it is very difficult to answer to these precise questions. We have only partial and generally relative orientation values (See answer to question 1).

However, in order to provide some information which could be useful for the task, the answers below are given for the following orientation values:

1. *General Objectives*
2. *Risk related to shutdown states*
3. *Objectives relating to Periodic Safety Assessments*
4. *Objectives relating to EPR*

It has to be noted that the orientation values used for the EPR project are more precise than for existing plants, and it is clearly indicated that the CMF has to be lower than for existing plants.

For Criterion 1 (General objectives):

In 1977, during the examination of the major technical options for the 1300 MWe plants, the Safety Authority set an overall probabilistic objective expressed as follows :

« In general terms, the design of a plant which includes a pressurized water nuclear reactor should be such that the overall probability that the plant could be the source of unacceptable consequences should not exceed E-6 per year .

This implies that, whenever a probabilistic approach is used to assess whether a family of events must be taken into account in the reactor design, the family must effectively be taken into account if its probability to lead to unacceptable consequences exceeds 1 E-7 per year.. »

It has to be noted that :

- a. *The overall objective is stipulated in terms of « unacceptable consequences », but these « unacceptable consequences » are not specified by legislation or regulation.*
- b. *The E-6 value is an « objective » for a PWR plant, but EDF is not required to demonstrate that this objective has been achieved.*
- c. *The 1 E-7 per year value is more practical for operational uses and was used in the approach to determine the risks generated by external events ; for example, the value was applied to several families of air crash events.*

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

Not expressed clearly (WASH 1400 results?)

X.4 What is the definition for this criterion (be precise)?

Yearly risk

X.5 What is the supporting documentation for this criterion?

The letter 1076/77 of the Nuclear Safety Division published in 1977.

X.6 To what is the criterion applicable?

Reactor level

X.7 How is the criterion expressed?

Single value.

X.8 How is this criterion considered?

This criteria are considered as orientation value. The demonstration is not required.

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

The PSAs were not performed and not required.

For Criterion 2 (Risk related to Shutdown conditions):

*A probabilistic target of E-6 per reactor*year for the CMF related to shutdown conditions was set by the Safety Authority (considering in particular that during shutdown containment integrity is not guaranteed).*

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

The first French PSA results indicated the high contribution of shutdown conditions to the overall CMF. The Safety Authority required from the utility to improve the situation and to demonstrate that the CMF related to shutdown conditions did not exceed E-6 per year.

X.4 What is the definition for this criterion (be precise)?

Yearly (taking into account the duration of each shutdown situation).

X.5 What is the supporting documentation for this criterion?

A Safety Authority letter sent to EDF in may 1990.

X.6 To what is the criterion applicable?

The criterion applies to a reactor. But in fact, due to the standardised series of French plants, the criterion applies to each of the reactors.

X.7 How is the criterion expressed?

A single value.

X.8 How is this criterion considered?

An orientation value.

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

The PSA used for the demonstration was a level 1 PSA, including only internal initiators but with all the plant conditions.

For Criterion 3 (Objectives relating to periodic Safety Reviews):

In the framework of each of the French PWR series Periodic Safety Reviews (PSR), the level 1 PSA is updated and the overall CMF compared to the previous value. Moreover each functional sequence of the PSA with a CMF contribution > 1 E-7 per year was analysed, in order to investigate the interest and the feasibility of plant improvements.

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

The PSR of the French PWR series are carried out every 10 years. The basis of these reviews is mainly deterministic, but PSA became progressively an important complement to the deterministic approach.

In the framework of each PSR, the level 1 PSA ("reference PSA") is updated and the overall CMF compared to the previous value. Moreover each functional sequence of the PSA with a CMF contribution > 10-8 per year is analysed, in order to investigate the interest and the feasibility of plant improvements. The functional sequences are analysed with particular attention in the following cases:

- *The sequence is new (compared to the previous PSA).*
- *The sequence contribution is more than 10% of the overall result.*
- *The sequence frequency increased by at least a factor 2 since the previous PSA.*

X.4 What is the definition for this criterion (be precise)?

The criteria apply to yearly risk. A functional sequence is defined as a group of PSA sequences which could be improved by the same plant modification.

X.5 What is the supporting documentation for this criterion?

The method for using PSA in PSR is described in the PSA Basic Safety Rule (RFS-2002/1). But there is no documentation giving the numerical criteria, which were set during technical discussions between EDF and the Safety Authority.

X.6 To what is the criterion applicable?

The criterion applies to a reactor. But in fact, due to the standardised series of French plants, the criterion applies to each of the reactors.

X.7 How is the criterion expressed?

A single value.

X.8 How is this criterion considered?

Orientation values.

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

For the first use of PSA for PSR, the "reference" PSA was a level 1 PSA, including only internal events but all the plant conditions.

Several studies are in progress for extending the scope of the "reference PSA" (especially level 2). The application for PSR will be extended accordingly, with the definition of appropriate criteria.

For Criterion 4 (Objectives relating to EPR):

- *For the French-German project EPR (European Pressurized Reactor), the French and German Safety Authorities gave the following very general probabilistic objectives:*
- *a reduced CMF compared to existing plants*
- *«practical elimination » of sequences with potential for large early releases.*

In order to fulfil these objectives, the designers have proposed probabilistic safety objectives as orientation values which give useful guidance but are not strict limits and do not correspond to a requirement of the Safety Authorities. Examples of these probabilistic objectives are a value of E-6 per year for the CMF due to internal events, respectively for power states and for shutdown states.

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

The French and German Safety Authorities considered that for a new project a PSA has to be carried out since the beginning of the project and that this PSA has to show a significant safety improvement of the new design compared to existing plants.

X.4 What is the definition for this criterion (be precise)?

A significant reduction of the global core melt frequency must be achieved for the nuclear power plants of the next generation. Implementation of improvements in the "defence-in-depth" of such plants should lead to the achievement of a global frequency of core melt of less that E-5 per plant operating year, uncertainties and all types of failures and hazards being taken into account.

Moreover, an important objective is to achieve a significant reduction of potential radioactive releases due to all conceivable accidents, including core melt accidents.

Accident situations with core melt which would lead to large early releases have to be "practically eliminated": if they cannot be considered as physically impossible, design provisions have to be taken to design them out. The demonstration cannot rely only on a probabilistic assessment.

X.5 What is the supporting documentation for this criterion?

The main document supporting the safety objectives is the "Technical Guidelines for Future PWRs"-IPSN/GRS report 82 (nov. 2000).

X.6 To what is the criterion applicable?

Reactor level.

X.7 How is the criterion expressed?

Single value (uncertainties taken into account).

X.8 How is this criterion considered?

Orientation values

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

During the design phase, only level 1/internal events/ all plant conditions were considered. The final PSA version will include level 2 and internal and external hazards.

10 How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

When a contribution is considered as important (compared to orientation values), several sensitivity studies are always performed. In particular independent assessments are carried out by EDF and by IRSN. This double assessment contributes to PSA quality and credibility.

11. When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

For the existing plants the PSA results are used at least during each periodic safety review of a PWR series (every 10 years).

For the EPR project the PSA is used during the analysis of the safety report.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

Since there are no strict criteria, we cannot answer exactly to this question. However when a contribution is important there is always a requirement for investigating the interest and the feasibility of plant safety improvements.

12. In case of band-defined goals, how is handled the case where the results are inside the band?

N/A

13. Have you defined other subsidiary criteria for PSA applications?

A probabilistic analysis of operating events is carried out in France since 1994. The aim of the quantitative analysis is to assess the risk increase (in term of core damage probability) due to the incident. An incident is considered as a precursor if the risk increase is higher than E-6 per event.

The Safety Authority required to take particular measures if the risk increase is higher than 10-4, and to assess the benefit of these measures.

When EDF requires an exemption from technical specifications for a particular event, an orientation value is that the risk related to this event does not exceed 1 E-7 per event.

14. What is your experience with Probabilistic Risk Criteria?

Until now we have no experience with real Probabilistic Risk Criteria. The orientation values introduced case by case for PSA applications were set after some discussions with EDF, but in fact the discussions were rather applied to the consequences of the risk results (clearly: is a modification possible? Necessary? Prioritary?)

In practice, in spite of the absence of Probabilistic Risk Criteria, PSA results were widely used for identifying safety improvements.

15. Your organisation has not defined Probabilistic Risk Criteria:

The French regulators consider that setting Probabilistic Safety Criteria is not necessary, and could have a negative effect.

The main reason (in addition to the classical lack of trust in PSA results because of uncertainties...) is that the aim of ASN is always to improve safety (not only to maintain it), and if the compliance with Probabilistic Criteria was demonstrated, this could lead to a low motivation for supplementary safety improvements.

Moreover it has to be noted that the French situation is somewhat particular, with a rather large reactor fleet nearly homogeneous, built by the same constructor, operated by the same utility. So it is not necessary to have a very strict regulation, but rather a continuous technical dialogue between EDF and the Safety Authority.

Due to this technical contact, it made it possible to obtain a large number of PSA applications even without strict criteria.

Response for Hungary

Identification

Please identify your organisation:

Name: **Hungarian Atomic Energy Authority**

Address : H-1036 Budapest, Fenyes Adolf utca 4

Country : Hungary

Are you

- A Regulatory Body ☒
- A Supporting organisation to a regulatory body ☐
- A Utility ☐
- A Vendor ☐

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1. Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1 Society level (qualitative)

The Volume 3 of the Nuclear Safety Codes issued by the Hungarian Governmental Decree No. 89/2005 in paragraph 2.002 prescribes:

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.

1.2 Intermediate level (qualitative or quantitative)

The Volume 3 of the Nuclear Safety Codes issued by the Hungarian Governmental Decree No. 89/2005 in paragraph 2.003 prescribes:

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 and the exposure has to be reduced to a reasonably possible extent during severe operational accidents.

1.3 Technical level (quantitative)

The Volume 3 of the Nuclear Safety Codes issued by the Hungarian Governmental Decree No. 89/2005 in paragraph 2.004 prescribes:

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.

. What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

There are no Probabilistic Risk Criteria neither formulated in the Nuclear Safety Codes, nor in use in the regulatory practice in Hungary. The only Probabilistic Safety Goal is stated in the Volume 3 of the Nuclear Safety Codes in paragraph 3.072:

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 E-5/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 5×10^{-4}*/year average value.*

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

Yes, the above Probabilistic Safety Goal shall be similarly applied both for existing and new plants.

<u>For Criterion X:</u>

The following examples will consider the CDF criterion

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

First of all, as it is mentioned, the referred statement above is not a Probabilistic Risk Criterion, but rather a Probabilistic Safety Goal. The Core damage Frequency, while not directly related to public safety, has been set to ensure compliance with defence-in-depth, avoiding design of an unsafe reactor with an extremely robust containment.

X.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

Core damage occurs if Criterion C5 applied within the safety assessment is not met.

Criterion C5: In the postulated accident conditions the primary reactor coolant system shall be maintained in a safe state so that short term and long term coolability of fuel can be maintained. It has to be proved that the following requirements shall be met:

- *the calculated maximum fuel rod cladding temperature does not exceed 1200 oC*
- *calculated changes in core geometry are such that the core remains amenable to cooling.*

X.5 What is the supporting documentation for this criterion?

Safety reassessment of the Paks Nuclear Power Plant
Final Report. Prepared in the Framework of the AGNES project. May 1995. Budapest, Hungary

X.6 To what is the criterion applicable?

The referred above Probabilistic Safety Goal is applicable to the reactor unit of the NPP.

X.7 How is the criterion expressed?

The referred above Probabilistic Safety Goal is expressed as a single value.

X.8 How is this criterion considered?

The referred above Probabilistic Safety Goal is considered as an orientation value. Meeting or not the criterion would result in prioritizing regulatory requirements for safety improvements.

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

The PSA used to assess the reactor Core Damage Frequency stated in the above Probabilistic Safety Goal shall be full scope (Internal and external events, full power and shutdown operating modes).

10 How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

The PSA report shall include uncertainty analysis. However, the computed CDF will be compared to the criterion as a best estimate.

11. When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

The Hungarian Regulatory Body performs the evaluation of plant performance against the Probabilistic Safety Goal at each Periodic Safety Review (every 10 years). The plant performance against the Probabilistic Safety Goal is also evaluated in cases of major modifications affecting the level of safety of the reactor facility as whole.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

A major Safety Upgrading Project was initiated and implemented for the Paks NPP during the period between 1995 and 2002. During this period specific Safety Upgrading Measures have been implemented thanks to which the general safety level of the reactor units of the Paks NPP has come closer step by step to the Probabilistic Safety Goal stated in the Volume 3 of the Nuclear Safety Codes in paragraph 3.072.

The PSA is updated regularly after each annual refuelling outages during which plant modifications have been implemented.

12. In case of band-defined goals, how is handled the case where the results are inside the band?

This question does not relate to the Hungarian practice.

13. Have you defined other subsidiary criteria for PSA applications?

No, other subsidiary criteria and goals have not been defined.

14. What is your experience with Probabilistic Risk Criteria?

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

For instance:

Introduction of the Probabilistic Safety Goal has been generally well received by the utility. The utility's efforts were and are much in harmony with the Probabilistic Safety Goal, which is accepted as a real target for improving the operation of the plant.

14.2 On consequences of implementation
Have Probabilistic Risk Criteria lead to system upgrades?

Yes, for details see response to question 11.2.

14.3 On communication with the public

The implementation of the Safety Upgrading Measures and thus the meeting of the Probabilistic Safety Goal has been communicated to the public. The public broadly accepts and supports the use of nuclear energy in general, what has been shown by a series of surveys and public hearings.

14.4 On interpretation

Considerable further internal discussions are expected for defining new Probabilistic Safety Goal for the Large Release Frequency, what is foreseen during the ongoing revision of the Nuclear Safety Codes.

15. Your organisation has not defined Probabilistic Risk Criteria:

During the ongoing revision of the Nuclear Safety Codes new Probabilistic Safety Goal is expected aiming on Large Release Frequency. We expect that the LRF as a new Probabilistic Safety Goal will lead to further safety enhancement at the utility specifically in the area of severe accident management.

Response for Japan

Identification

Please identify your organisation:

Name : **Japan Nuclear Energy Safety Organization**

Address : 4-3-20 Toranomon, Minato-ku, Tokyo 105-0001

Name : **Japan Atomic Energy Agency**

Address : 2-4 Shirakata-shirane, Tokai-mura, Ibaraki 319-1195

Country : *Japan*

Are you

- A Regulatory Body □
- A Supporting organisation to a regulatory body ■
- A Utility □
- A Vendor □

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1. Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

The Japanese Nuclear Safety Commission (NSC) stated in the White Paper on Nuclear Safety published in March 1999, that NSC promoted the discussion on the establishment of safety goals from a comprehensive point of view, taking into consideration international trends and outcomes from various PSA studies. Then, NSC established the "Special Committee on Nuclear Safety Goals" in September 2000 to deliberate on the concepts of safety goals.

In parallel with this activity, NSC recognised the important role of risk information in ensuring nuclear safety as addressed in the Basic Policy of NSC on "Introduction of Risk Informed Nuclear Safety Regulation" issued in 2003. NSC takes an initiative by coordinating among relevant organizations to encourage and promote utilization of risk information into the nuclear safety regulatory processes from viewpoint of enhancement of rationality, consistency and transparency as well as appropriate allocation of regulatory resources in the nuclear safety regulation. Safety goals and performance goals are recognized as candidates as quantitative and practical indicators for utilisation of risk information into the nuclear safety regulation.

In December 2003, the special committee proposed qualitative (the following 1.1) and quantitative (the following 1.2) safety goals to be applied consistently to all types of nuclear activities and future efforts to investigate performance goals for each field of utilization of nuclear energy in the "Interim Report on the Discussion of Safety Goals (ref.1)".

After the issue of quantitative safety goals, the special committee discussed and derived the performance goals for nuclear power plants (the following 1.3) so as to meet compliance with the quantitative safety goals. The performance goals were issued in the "Report on Performance Goals for Light Water Power Reactors (ref.2)" in March 2006.

1) *NSC, "Interim Report on the Discussion of Safety Goals (in Japanese)", Special Committee on Nuclear Safety Goals of NSC, December 2003.*

2) *NSC, "Report on Performance Goals for Light Water Power Reactors, - on performance goals consistent with safety goals - (in Japanese)", Special Committee on Nuclear Safety Goals of NSC, March 2006.*

1.1 Society level (qualitative)

"The likelihood of occurrence of health detriment to the pubic due to emission of radiation or release of radioactive materials from activities for nuclear energy utilisation should be controlled to such a level that members of the public bear no significant additional risk to their daily life."

(NSC, "Interim Report on the Discussion of Safety Goals", Special Committee on Nuclear Safety Goals of NSC, December 2003.)

1.2 Intermediate level (qualitative or quantitative)

"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."

"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, "Interim Report on the Discussion of Safety Goals", Special Committee on Nuclear Safety Goals of NSC, December 2003.)

1.3 Technical level (quantitative)

Core Damage Frequency (CDF) : approximately E-4per reactor year

Containment Failure Frequency (CFF): approximately E-5 per reactor year

Both of the two goals are required to be met at the same time for all events including internal and external initiating events.

(NSC, "Report on Performance Goals for Light Water Power Reactors, - on performance goals consistent with safety goals -", Special Committee on Nuclear Safety Goals of NSC, March 2006.)

2 What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

a. *Individual risk of early fatality and individual risk of cancer fatality*
b. *Core damage frequency and containment failure frequency*

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

The performance goals given in B) above have been developed to achieve an consistent level of safety with the quantitative safety goals, considering the characteristics of NPPs. They are to be applied for both existing and future NPPs. NSC, however, is expecting that utilities make efforts to develop NPPs with higher level of safety.

<u>For Criterion</u>

A) Individual risk of early fatality and individual risk of cancer fatality:

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

The interim report "Interim Report on the Discussion of Safety Goals" defines the role of the safety goal as the quantitative definition of the level to which the risks of the utilisation of nuclear energy are required to be controlled under the regulatory activities by the government. The proposed quantitative goal should be objective and common to various nuclear facilities and activities that give rise to potential adverse health effects from radiation exposure to the public. Among various kinds of adverse effects from nuclear accidents, individual risk of fatality to members of the public, which satisfies both requirements, has been set as risk indicators for the quantitative safety goals.

In the interim report "Interim Report on the Discussion of Safety Goals", NSA states that establishing such safety goals will provide the following benefits:

- *To prevent the risks from becoming actualized, the government requires the licensees to take all reasonably practicable efforts for assuring the safety and monitors their efforts. The safety goals may be used by the government for giving their regulatory activities more transparent and predictable, and for making each regulatory action more effective and efficient. The safety goals may also be used in reviewing regulatory actions in different fields of utilizing nuclear energy from a common point of view and make these regulatory actions more rational and consistent.*
- *In recent years, there is increasing demand that public opinion be reflected in the decision making processes by the government on regulatory and other actions. The safety goals, which use the risks to the public as a measure of the objectives of regulatory activities, may be used for making more effective and efficient communication between the government and the public on the nuclear regulatory actions taken by the government, such as the development of guidelines and standards.*
- *The licensees may make their risk management activities more efficient and effective in achieving the requirement by the regulators by using the safety goals as a basis for planning and assessing of their activities.*

X.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

The proposed quantitative safety goals cover the average individual risk of early fatality to the most exposed members of the public who stay in the vicinity of the nuclear power plant and the average individual risk of cancer fatality to members of the public who stay within a specified distance from the nuclear power plant. It should be noted that the target of the proposed goals is different from the critical group. But the "Interim Report on the Discussion of Safety Goals" does not describe the definitions of "vicinity of the site boundary" and also "within a certain distance from a nuclear facility". In the derivation of the CFF value, NSC suggests that the "vicinity of the site boundary" be within one or two kilometres from the site boundary and also "within a certain distance from a nuclear facility" be within two to five kilometres from the site boundary.

X.5 What is the supporting documentation for this criterion?
Please list and give a short abstract of each key publicly available document supporting the above criterion.

If no publicly available document exists, could you explain briefly how the criterion is supported?

– *ref.1: NSC, "Interim Report on the Discussion of Safety Goals (in Japanese)", Special Committee on Nuclear Safety Goals of NSC, December 2003.*

X.6 To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

The Individual risk of early and cancer fatality is set at site level.

X.7 How is the criterion expressed?

- Single value
- Band (limit and target)

Single value:
"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."

"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."

X.8 How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

These criteria are considered as safety indicator. They should be generally applied as reference levels in regulatory activities to examine their rationality and consistency and also determine the depth and extension of the actions to be taken for ensuring the safety of nuclear facilities. Meeting or not the criterion would result in reviewing regulations to find inappropriate matters.

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

These quantitative goals are applied to nuclear accidents caused by both internal and external events except for intentional man-made events such as sabotage.

<u>For Criterion</u>

B) Core damage frequency and containment failure frequency:

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

In "Interim Report on the Discussion of Safety Goals" NSC stated that performance goal for each type of nuclear facility be examined and set as safety benchmark to demonstrate compliance with these quantitative health objectives for measuring plant performance.

NSC considers that the trial use of performance goals in various activities on safety regulations for nuclear power plants will result in the accumulation of experiences of risk assessment techniques, the effective assurance and improvement of safety for nuclear power plants.

X.4 What is the definition for this criterion (be precise)?

In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

Core damage frequency (CDF) is defined as a benchmark for the performance of safety functions for preventing severe accident.

X.5 What is the supporting documentation for this criterion?

Please list and give a short abstract of each key publicly available document supporting the above criterion.

If no publicly available document exists, could you explain briefly how the criterion is supported?

- *NSC, "Report on Performance Goals for Light Water Power Reactors, - on performance goals consistent with safety goals - (in Japanese)", Special Committee on Nuclear Safety Goals of NSC, March 2006.*

X.6 To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

The CDF and CFF are set at reactor level.

Since the proposed safety goals are set at site level, additional considerations for the effects of multiple units should be made for applying the performance goals to a reactor in a multi-unit site. One can ensure that the combined risks to the individuals around the site from multiple units do not exceed the proposed safety goals.

X.7 How is the criterion expressed?

- Single value
- Band (limit and target)

Single value:
Core Damage Frequency (CDF): approximately E-4per reactor year
Containment Failure Frequency (CFF): approximately E-5 per reactor year

Both of the two goals are required to be met at the same time for all events including internal and external initiating events.

X.8 How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

These criteria are considered as safety indicator.

They should be generally applied as reference levels in regulatory activities to examine their rationality and consistency and also determine the depth and extension of the actions to be taken for ensuring the safety of nuclear facilities. Meeting or not the criterion would result in reviewing regulations to find inappropriate matters

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

These performance goals are also applied to nuclear accidents caused by both internal and external events except for intentional man-made events such as sabotage. However, there are large differences in maturity of risk assessment techniques and uncertainty bands of PSA results for external events such as tsunami and floods for which the experience of PSA and the development of database are not sufficient. In real applications of the performance goals, PSA will not be necessarily performed for all the initiating events.

10 How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

The mean values that could be obtained by evaluating the magnitude of the uncertainty will be used as a general rule for the comparison of the quantitative safety goals or the performance goals with the PSA results.

The reason why the quantitative goals are expressed by the 'approximate' numbers is to take into account the variability of PSA results for individual facilities due to the uncertainties in PSA and the variability in reasonably practicable safety measures at individual facilities. Even in the case where risks evaluated for some facilities are slightly larger (a factor of two will be used in the trial usage period) than the quantitative goals, it does not automatically mean that the regulatory rules applied to those facilities are inadequate but can be adequate provided that reasonable safety measures are taken in those facilities.

11 When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

The above safety goals and performance goals are not regulatory requirements. Therefore, there is no requirement for updating PSA and comparing their results with the safety goals or performance goals. However, as utilities have voluntarily implemented PSAs in PSR, PSAs are updated at least every tenth year.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

The above safety goals and performance goals are not regulatory requirements. The interim report of ref.1. Namely it states that the value of safety goal (1/1,000,000 a year) will not strictly be applied when determining whether or not individual facilities violate the safety goals because of uncertainties in the data handled during the risk assessment. If the assessment results exceed this value, nuclear facilities will not be judged to violate the safety goals as long as they have highly reliable and effective safety measures planned and implemented and the risk assessment results are equal to or below 2/1,000,000 a year. The validity of this factor of two will be examined through future trial applications.

12 In case of band-defined goals, how is handled the case where the results are inside the band?

N/A

13. Have you defined other subsidiary criteria for PSA applications?

The work on the risk information application guideline was started in October 2006 by the standards committee of the Atomic Energy Society of Japan. This application guideline will describe basic processes, which include allowable risk criteria, for utilities to apply risk information to safety management and ensuring safety in nuclear power plant. This guideline keeps in mind the basic principle of "Fundamental Guidelines for Utilization of "Risk Information" in Safety Regulation for NPPs - Trial Use –" issued by NISA. The following subsidiary criteria are under discussion to define in the guideline.

- *Acceptance guidelines for changes of CDF and CFF as functions of base-line CDF and CFF, similar to those of RG1.174 in USA*
- *Allowed Incremental Conditional Core Damage Probability and Incremental Conditional Containment Failure Probability*
- *Importance measure criteria used to identify candidate safety significance*

A draft standard will be completed in 2008.

14. What is your experience with Probabilistic Risk Criteria?

Please provide a separate response for every experience you wish to share. These experience statements can address one or several of the following sub-criteria.

We do not expect a complete relation of your experience with probabilistic Risk Criteria, but a limited number of statements addressing the most important.

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

We have not yet applied the above safety goals and performance goals.

14.2 On consequences of implementation
Have Probabilistic Risk Criteria lead to system upgrades?

N/A because we have not yet implemented the safety goals

14.3 On communication with the public

N/A because we have not yet implemented the safety goals

"Interim Report on the Discussion of Safety Goals (ref.1)" places emphasis on the dialogs with the public in the following.

To further develop and deepen discussion on the safety goals, this interim report needs to be presented to the public and dialog promoted. This dialog should explain the importance of the efforts to keep the health risks caused by utilising nuclear energy as low as reasonably practicable. The concepts of the safety goals indicating the health risk levels and the probabilistic risk assessment on which the safety goals are based, which are still unfamiliar to the public, should also be explained to the public. Documents should be prepared describing the risk assessment basics and concepts and the risks involved in various activities in society in order to explain the safety goals. For these safety goals to be widely accepted by the public and valued by the persons concerned, their aims, contents, and application manner should be discussed through continuous dialog with the public. There should be dialog at each stage of implementing the safety goals, i.e., at the stage of presenting the draft safety goals to the public, the stage of applying them on a trial basis, and the stage of establishing and applying formal safety goals.

14.4 On interpretation

In the special committee on safety goals there have been considerable discussions about various issues such as "collective risk", "societal risk", "risk to individuals covered by safety goals", "comparison with the risk control goals in the air quality standards".

In the performance goals subcommittee under the special committee on safety goals there has been considerable discussions about "metrics of performances goal".

15 Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

N/A

Response for Korea

Identification

Please identify your organisation:

Name: **Korea Institute of Nuclear Safety(KINS)**

Address : P.O. Box 114 Yuseong Daejeon 305-600

Country : Republic of Korea

Are you

- A Regulatory Body □
- **A Supporting organisation to a regulatory body □**
- A Utility □

A Vendor □

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1 Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1 Society level (qualitative)

The qualitative criteria at society level have not been declared yet in Korea.

1.2 Intermediate level (qualitative or quantitative)

The Korean Nuclear Safety Commission which is the high-level advisory commission of nuclear safety issued the Policy on Severe Accident in 2001. It prescribes comprehensive measures against severe accident, including PSA implementation. The main objectives of the policy are to assure that the possibility of a severe accident occurrence is extremely low and its risk to the public is sufficiently reduced. One of the essential elements in the Policy is the establishment of probabilistic safety criteria. In the Policy the quantitative values (Safety Goal) were already declared as follows;

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.

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."

1.3 Technical level (quantitative)

The Policy also required the establishment of the performance goals (the probabilistic risk criteria) in order to practically introduce the above goals to NPPs in terms of risk.

Thus, the performance goals (Core Damage Frequency, Large Containment Release Frequency) should be established in near future, and so far the official Probabilistic Risk Criteria have not been existed in Korea.

However, Korean sole utility (Korea Hydro and Nuclear Power: KHNP) has its own criteria in designing the new reactor (APR-1400) which is now under the construction permit phase. The technical base of the design was the KURD (Korean Utility Requirement Document) which was developed for the design of APR-1400 in Korea. Standard SAR for APR-1400 describes that the purpose of PSA is to demonstrate that the cumulative frequency be less than 1.0E-06/ry for sequences resulting in greater than 1 Rem effective dose equivalent for 24hours at site boundary from any individual reactor, as required in KURD.

Korea Institute of Nuclear Safety (KINS), as a supporting organisation to a regulatory body, tentatively agrees with the IAEA risk criteria such as 'less than 1E-04/yr of CDF and less than 1E-05/ry of LERF for the operating plants. For future plants including APR-1400, one tenth of operating plant's level is agreeable.

2. What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

As mentioned in (1.3), Korea has not developed the Probabilistic Risk Criteria requested in this section. However, according to the Policy on severe accident the performance goals (the Probabilistic Risk Criteria) will be established in near future in terms of Core Damage Frequency and Large Early Release Frequency. It means Korea will adopt CDF and LERF as the technical level Probabilistic Risk Criteria.

(The values below are based on the tentative regulatory decision until the official issuance of the Criteria)

- *CDF for existing plants and life extension : less than 1E-04/yr*
- *CDF for new plants : less than 1E-05/yr*
- *LERF for existing plants and life extension, : less than 1E-05/yr*
- *LERF for new plants : less than 1E-06/yr*

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

The Large Early Release Frequency and Core Damage Frequency are the same for existing plants and life extension, but are different for new plants.

<u>For Criterion CDF:</u>

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

Please elaborate on the reasons

Korea has several reasons CDF must be one of the Probabilistic Risk Criterion.

- *CDF is a well known measure to identify the plant design vulnerability as well as to represent the operational safety, while not directly related to public safety.*
- *The criterion for CDF has been developed in other countries and international organization, which means so much international experiences with long history exist. Also, it makes possible the comparison between our own value and other countries' ones to judge the acceptance of the individual plant safety level.*
- *Korea is introducing the risk-informed approach in some areas like the regulatory inspection, In-service Inspection and Technical Specification change, and so on. Korean licensee started the operation of risk monitoring system in their own plants as well. From the view point of this situation, CDF-based criterion would play an essential role as a standard or basis in licensing and operation.*

X.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

(The definitions below are based on the tentative regulatory decision until the official issuance of the Criteria)

Core damage in PWR is defined as follows;

- *Maximum fuel cladding temperature > 2200 (1204), or*
- *Uncovery of top of the reactor core except the case caused by instant re-flooding.*

Core damage in PHWR is defined as 'Multiple fuel channel failure', might be occurred by following specific situations;

- *Early loss of core structural integrity*
- *Late loss of core structural integrity with high Primary Heat Transport system pressure*
- *Late channel disassembly with core structural integrity maintained.*

This definition used in PHWR was used in PSA by Korean licensee, and was admitted by KINS.

X.5 What is the supporting documentation for this criterion?

Please list and give a short abstract of each key publicly available document supporting the above criterion.

If no publicly available document exists, could you explain briefly how the criterion is supported?

- *Internal discussions comparing the proposed CDF with international practice.*

X.6 To what is the criterion applicable?

- *A reactor,*
- *A plant (multiple reactors sharing at least one safety system),*
- *A site (several reactors on the same location),*
- *A population of reactors (all the reactors in the country)?*

The CDF is set at reactor level.

Please explain why you chose this applicability.

It is very complicated situation that accidents could simultaneously occur at more than one plant. Korea has not considered such a complicated case while a simplified approach is preferred in the beginning of the establishment of the Criteria. Another reason is there is no enough information and experience for the other cases except a reactor level.

X.7 How is the criterion expressed?

- *Single value*

- *Band (limit and target)*

(These values below are based on the tentative regulatory decision until the official issuance of the Criteria)

Single value;

- *CDF for existing plants and life extension: less than 1E-04/yr*
- *CDF for new plants: less than 1E-05/yr*

X.8 How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

Korea has not decided the characteristic of the risk criterion since the official risk criteria has not fixed. But, our experience indicates that the criterion has been used with the consideration of "strict, but not legally-bound, limits".

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

Basically, the PSA used to assess the CDF shall be full scope (internal and external events, full power and shutdown operating modes). Among 20 operating plants, only recently constructed plants (4 units) performed low power shutdown PSA while 16 plants have PSAs only in the full power mode. Korean utility has the plan of the implementation of low power shutdown PSA. Until the completion of the full scope PSA in the future, qualitative aspect of low power shutdown risk shall be considered in the application of the risk criteria.

<u>For Criterion LERF:</u>

X.3 What is the reason why you developed this Probabilistic Risk Criterion?
Please elaborate on the reasons

Korea has several reasons LERF must be one of the Probabilistic Risk Criterion.

- *LERF is a well known measure to identify the countermeasure design and management against severe accident as well as to estimate the radiation release frequency which is directly related to public safety.*
- *The criterion for LERF has been developed in other countries and international organization, which means not a few international experiences with long history exist. Also, it makes possible the comparison between our own value and other countries' values to judge the acceptance of the individual plant safety level.*
- *Korea is introducing the risk-informed approach in some areas like the regulatory inspection, In-service Inspection and Technical Specification change, and so on. Korean licensee started the operation of risk monitoring system in their own plants as well. From the view point of this situation, LERF-based criterion would play an essential role as a standard or basis in licensing and operation.*

X.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

(The definitions below are based on the tentative regulatory decision until the official issuance of risk criteria)

Large Early Release is defined as follows;

- *The rapid, unmitigated large release of airborne fission products from the containment to the environment, resulting in the early death of more than 1 person or causing the severe social effect.*

X.5 What is the supporting documentation for this criterion?

Please list and give a short abstract of each key publicly available document supporting the above criterion.

If no publicly available document exists, could you explain briefly how the criterion is supported?

- *Internal discussions comparing the proposed LERF with international practice*

X.6 To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

The LERF is set at reactor level.

Please explain why you chose this applicability.

The same reason with CDF.

X.7 How is the criterion expressed?

- Single value
- Band (limit and target)

(These values below are based on the tentative regulatory decision until the official issuance of the Criteria)

Single value;

- LERF for existing plants and life extension: less than 1E-05/yr
- LERF for new plant : less than 1E-06/ry

X.8 – How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

Korea has not decided the characteristic of the risk criterion since the official risk criteria has not fixed. But, our experience indicates that the criterion has been used with the consideration of "strict, but not legally-bound, limits".

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

Basically, the PSA used to assess the LERF shall be full scope (internal and external events, full power and shutdown operating modes). All of operating plants have performed PSAs to show their specific LERF only in the full power mode. Recently constructed plants performed the low power shutdown PSAs, but these PSAs were modelled just in level-1 boundary, not in level-2. Korean utility has the plan of the implementation of low power shutdown PSA. Until the completion of the full scope PSA in the future, qualitative aspect of low power shutdown risk shall be considered in the application of the risk criteria.

10 How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

The PSA report shall include uncertainty analysis. However, the computed CDF will be compared to the criterion as a best estimate. Comparing the PSA results with the criterion, the quality and uncertainty of PSA results would be considered and several sensitivity analyses are required.

11 When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

The risk criteria shall be applied to compare the PSA results of new plant during licensing review as well as the periodically updated PSA results of operating NPPs with the criteria

Specifically, risk values reflecting the design changes, improvement in the systems, and/or procedural changes shall be thoroughly reviewed during Periodic Safety Review with 10 years interval and life extension.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

- *Korea has no specific rule in the case of the beyond criteria. But following procedure is expected to cover the unsatisfied cases.*
- *If a Probabilistic Risk Criterion is exceeded, analysis of the reasons for this violation may be requested to identify the cause. Regulatory actions will be set according to the cause of violation, ranging from strict regulatory actions to correct the cause in a given time to acceptance of the situation with reasonable alternatives or/and background.*
- *The regulatory action may depend on the phase of the plants, i.e. during construction, during commercial operation, during Periodic Safety Review with 10 years interval, and life extension. Life extension could apply more strict requirements than other phases, considering the public acceptance of life extension.*

12. In case of band-defined goals, how is handled the case where the results are inside the band?

Korea does not consider the band-defined goals.

13. Have you defined other subsidiary criteria for PSA applications?

Korea has defined subsidiary criteria for PSA application. They are the acceptance criteria for the plant specific changes to the licensing basis such as Technical Specification modification (AOT, STI), Risk-informed In-service Inspection (RI-ISI), and so on. The subsidiary criteria consist of band-defined criteria for evaluating the increased CDF or LERF affected by the changes. The criteria of increased risk value proportionally change according to the plant specific base CDF and LERF. (similiar to the acceptance criteria in the USNRC Regulatory Guide 1.174)

14. What is your experience with Probabilistic Risk Criteria?

Please provide a separate response for every experience you wish to share. These experience statements can address one or several of the following sub-criteria.

We do not expect a complete relation of your experience with probabilistic Risk Criteria, but a limited number of statements addressing the most important.

14.1. On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

Korea has 20 NPPs in operation and 6 NPPs under construction, all of the plants have their specific PSA models. Among 20 operating plants, some have initially performed their PSAs while some updated since the issuance of the Policy on severe accident in 2001, which required performance of PSAs for all operating NPPs to confirm the plant risk level low enough to prevent and mitigate a severe accident.

The results of final PSA for all operating NPPs show that the risk level of the most plants is less than 1E-4/ry for CDF and 1E-5/ry for LERF except a couple of plant. Since 2001, Korean utility has reduced the risk level through the design modification and operation procedural changes when the risk level had not met the interim risk criteria.

However, this result does not always include the full scope of PSA. Only four plants include full scopes in PSA, i.e. internal and external events, and full power and low power shutdown mode. Accordingly, the acceptance of risk level for the remaining 16 plants has to be qualitatively judged through the consideration of alternatives, technical difficulties, and/or the results of cost/benefit analysis suggested by the utility.

Korean utility has tried to reduce the risk levels of their plants as much as possible to meet the Probabilistic Risk Criteria even they are interim. It is very meaningful the Probabilistic Risk Criteria can provide both of the regulator and the utility with the visible target for their missions. Additional benefits are that both of them can acknowledge what they need to do or track down to satisfy the criteria. For example,

- *Establishment of the rationale for the Criteria.*
- *Consensus of the quantitative values of the Criteria, i.e., CDF and LERF among the regulator, the utility and the public.*
- *Well-defined Criteria and effective approach to set up the Criteria considering various plants types, i.e., PWR and PHWR in Korea*
- *Manipulation of PSA scopes like external event and low power shutdown*

14.2 On consequences of implementation
Have Probabilistic Risk Criteria lead to system upgrades?

Yes, Korea has the experience that the Probabilistic Risk Criteria have lead to system upgrades. See (14.1)

14.3 On communication with the public

Before the issuance of the official Probabilistic Risk Criteria they should keep the official step to be announced to the public through several channels like Nuclear Safety Information Centre operated by KINS. On the way of drafting the Criteria, the opinions from the utility, the research institute, engineering company, and college will be reviewed and reflected if necessary.

14.4 On interpretation

As mentioned in (14.1), considerable internal discussion for defining what would be Criteria is positively needed.

Examples of other areas of interest might be beneficial:

15. Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

Korea has not officially established the Probabilistic Risk Criteria. KINS will prepare the draft of the Criteria (CDF, LERF) in the near future after assessing the plant specific risk level and applied methodologies. International cooperation on this subject is very helpful, and the collaborated results will be a valuable material to set up the Korean specific Criteria.

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J4-TM-29260
Technical Meeting on
DEVELOPMENT OF A SAFETY GUIDE ON LEVEL 1 PROBABILISTIC SAFETY ASSESSMENT (PSA) AND APPLICATIONS FOR NUCLEAR REACTORS
IAEA, Vienna, 20 – 24 November 2006

Questionnaire on National Safety Goals and PSA Regulations

Item of Inquiry	YOUR Reply	Comments
Country:	Romania	
Is PSA required for license?	Elaboration of PSA was required as licensing condition. The regulation issued does not specifically address PSA's role in licensing.	
Is PSA required as part of PSR?	The review of PSA as required as part of the PSR	
Do you have numerical risk limits established in the national regulatory documents? If yes, please give the title(s) of the documents:	No -	
Are the values formulated as safety TARGETS/GOALS or CRITERIA (i.e. to strictly comply with?):	-	
What is the nature of the estimate (i.e. point estimate, mean, median, other)?	-	
What is the scope of PSA for which safety goals/criteria are specified?	-	
Please give the numerical values for:	-	
CDF:	-	
LERF:	-	
Level-2 (if any):	-	
Values related to risk for population (i.e. outcome of a Level-3 PSA) – please specify what value is meant:	-	
Other (please specify what are the values):	-	
Do you have requirements for uncertainty bounds for the safety goals/criteria? If yes, what are they?	-	

Do you have safety goals/criteria for a multiple unit site? If yes please specify this:	For siting: For reactors that share common systems, such that an accident at the reactor could affect the safety of any of the other reactors, the exclusion area and how density of population area should be chosen on the assumption that all the "interconnected" reactors release the postulated quantities of fission products simultaneously. This requirement can be relaxed in relation with the extent to which the reactors are "interconnected" taking also into account the probability of simultaneously occurring accidents and the probabilistic that a person from the public as exposed to the effects of radioactivity from simultaneous releases. The basis for assuming a lower release has to be justified. (for exclusion zone boundary: 25 Rem whole body (2 hours after accident) or 100 Rem thyroid adult due to iodine. For low density of population zone boundary: 100rem thyroid (adult) due to iodine collective dose calculated per any radial sector of 22,5° shall be less than 10^6 man Rem Note: the above mentioned requirement originates in a NRC requirement.	

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DEVELOPMENT OF A SAFETY GUIDE ON LEVEL 1 PROBABILISTIC SAFETY ASSESSMENT (PSA) AND APPLICATIONS FOR NUCLEAR REACTORS
IAEA, Vienna, 20 – 24 November 2006

Questionnaire on National Safety Goals and PSA Regulations

Item of Inquiry	YOUR Reply	Comments
Country:	**Russia**	
Is PSA required for license?	Yes	
Is PSA required as part of PSR?	Yes	
Do you have numerical risk limits established in the national regulatory documents?	Yes	
If yes, please give the title(s) of the documents:	1. Main safety rules OPB 88/97 2. Requirements to siting	
Are the values formulated as safety TARGETS/GOALS or CRITERIA (i.e. to strictly comply with?):	Safety goals	
What is the nature of the estimate (i.e. point estimate, mean, median, other)?	Not defined	
What is the scope of PSA for which safety goals/criteria are specified?	Full scope Level 1 PSA Level 2 PSA at power for all initiating events	Document RB-032-04
Please give the numerical values for:	**-**	
CDF:	***1E-5 per reactor year***	
LERF:	**-**	
Level-2 (if any):	***Limit accident release frequency 1E-7 per reactor year***	
Values related to risk for population (i.e. outcome of a Level-3 PSA) – please specify what value is meant:	-	
Other (please specify what are the values):	Screening criteria: 1) 1E-7 $^{1}/_{a}$ initiating event 2) 1E-6 $^{1}/_{a}$ – external hazards	
Do you have requirements for uncertainty bounds for the safety goals/criteria? If yes, what are they?	No	
Do you have safety goals/criteria for a multiple unit site? If yes please specify this:	No	

Response for Slovakia

Identification

Name: **Nuclear Regulatory Authority of the Slovak Republic**

Address : Bajkalska 27, Bratislava

Country : Slovak Republic

Are you

- A Regulatory Body ☒
- A Supporting organisation to a regulatory body ☐
- A Utility ☐
- A Vendor ☐

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1. Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1 Society level (qualitative)

Protect the public and the environment from unreasonable risk

1.2 Intermediate level (qualitative or quantitative)

Not available.

1.3 Technical level (quantitative)

Core Damage Frequency should not be exceeded 1,0E-4/year.

Large Early Release Frequency should not be exceeded 1,0E-5/year.

These criteria are published in the regulatory guideline Requirements for PSA performance BNS I.4.2/2006.

2. What technical level Probabilistic Risk Criteria have you developed?

   Please list these criteria

   – *Large early release frequency*
   – *Core damage frequency*

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

The regulatory guideline Requirements for PSA performance BNS I.4.2/2006 defines criteria for existing plants. Criteria for the new plant could be more stringent than once. Existing criteria are defined in accordance with IAEA recommendations (The Role of Probabilistic Safety Assessment and Probabilistic Safety Criteria in Nuclear Power Plant Safety, Safety Series No.106, International Atomic Energy Agency, Vienna, May 1992).

3. What is the reason why you developed this Probabilistic Risk Criterion?

UJD defines probabilistic safety criteria (numerical targets) for reviewing and increasing the nuclear safety. These criteria has been developed and implemented based on international practises and experiences.

4. What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

Large early release is defined:

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.

Core damage is defined:

For VVER 440 the core considered to be in damage state once the top of the active fuel assemblies is uncovered. The criterion for the core damage is conservatively assumed to be core uncovery. Large LOCA is considered as the exception due to the temporary core uncover during the blowdown phase.

These criteria are applied to average risk for the one unit per year.

5. What is the supporting documentation for this criterion?

Please list and give a short abstract of each key publicly available document supporting the above criterion.

The regulatory guideline Requirements for PSA performance, UJD SR, BNS I.4.2/2006, Bratislava, Slovakia, 2006 characterises regulatory body approach to the PSA and its applications in practice. It provides the purpose of PSA and defines probabilistic safety targets to be fulfilled to reach adequate level of safety. It summarises requirements and conditions, which are laid down for the preparation and review of PSA. Duties of the utility are described on the scope, contents, review, quality assurance and upgrading of PSA. A possible application of PSA is provided for its use at the regulatory body and utility.

6. To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

Criteria are set at reactor level.

7. How is the criterion expressed?

- Single value
- Band (limit and target)

Single value:

Core Damage Frequency should not be exceeded 1,0 E-4/year.

Large Early Release Frequency should not be exceeded 1,0E-5/year.

These values taking into account all regimes of operation.

8. How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

Criteria are considered as an orientation value and have status of recommendation.

9. What is the scope of the analysis used for measuring plant performance against the criterion?

PSA level 1 represents an assessment of the risk of nuclear fuel damage at the nuclear facility for all operating modes and includes important initiating events and risks, internal fires and floods, extreme meteorological conditions and earthquake.

10. How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

Sensitivity and uncertainty analysis is required. There are not any criteria defined for these analyses. Criteria are defined on the level of CDF for level 1 PSA and LERF for level 2 PSA.

11. When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

The PSA used to compare plant compliance with the Probabilistic Safety Criteria. PSA shall be performed as part of a periodic assessment of the safety of the nuclear installation (every 10 years) and always if there has been a significant change in the design of the nuclear installation, or there has been a significant change in the operating regulations or a new significant risk has been identified.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

If a Probabilistic Risk Criterion is exceeded, identification of the reasons for this exceedance is performed and improvement measure is done.

12. In case of band-defined goals, how is handled the case where the results are inside the band?

N/A

13. Have you defined other subsidiary criteria for PSA applications?

The cumulative effect of changes (sum of all increments) should not be exceeded 2,5 E-6 for CDF during 10 years of operation or during the interval of periodic safety review and single change should not be exceeded 1,0 E-6/y. The increment of the risk is acceptable only if the mean value of CDF is less than 1,0 E-4/y.

The cumulative effect of changes (sum of all increments) should not be exceeded 2,0 E-7 for LERF during 10 years of operation or during the interval of periodic safety review and single change should not be exceeded 1,0E-7/y. The increment of the risk is acceptable only if the mean value of LERF is less than 1,0 E-5/y.

14. What is your experience with Probabilistic Risk Criteria?

Please provide a separate response for every experience you wish to share. These experience statements can address one or several of the following sub-criteria.

We do not expect a complete relation of your experience with probabilistic Risk Criteria, but a limited number of statements addressing the most important.

14.1, 14.2, 14.3, 14.4

Probabilistic safety criteria are defined in accordance with IAEA recommendations and accepted by operator and support organizations.

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

Safety upgrading of the plants was performed. The benefit of the safety upgrading was evaluated using PSA. Safety measures were implemented if the criteria were not met.

14.2 On consequences of implementation
Have Probabilistic Risk Criteria lead to system upgrades?

Within the safety upgrading of the plant the safety systems were modified to achieve the Probabilistic Risk Criteria.

14.3 On communication with the public

It was declared for the public that high level of safety is achieved because the Probabilistic Risk Criteria are met.

14.4 On interpretation

The achievement of Probabilistic Risk Criteria means high level of plant safety.

15 Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

N/A

Response for Switzerland

Identification

Please identify your organisation:

Name: **Gerhard Schoen**

Address: Hauptabteilung für die Sicherheit der Kernanlagen, HSK, CH-5232 Villigen-HSK, Switzerland

Country: Switzerland

Are you

- A Regulatory Body **X**
- A Supporting organisation to a regulatory body
- A Utility
- A Vendor

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1. Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1 Society level (qualitative)

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.

1.2 Intermediate level (qualitative or quantitative)

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 10^{-6}/yr.

1.3 Technical level (quantitative)

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 E-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.

2. What technical level Probabilistic Risk Criteria have you developed?

2.1 Please list these criteria

Core Damage Frequency (CDF)

2.2 Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

As mentioned in the response to Q1.3, for getting the construction permit of a new nuclear power plant, the applicant is required to demonstrate that the CDF is below E-5 per year. This risk criterion is also expected to be fulfilled by the existing plants, to the extent that is reasonably achievable (orientation value for existing plants).

Please answer the following questions for each of the listed criteria, limiting the responses to criteria at or above system level.

<u>For Criterion X:</u>

The following examples will consider the CDF criterion

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

Please elaborate on the reasons

The Swiss Federal Nuclear Safety Inspectorate (HSK) follows an integrated regulatory safety oversight approach where PSA is one element in the decision making. Comprehensive and full scope PSAs for Swiss NPPs are required, which are subject to extensive reviews to ensure similar high level of quality and some degree of harmonisation. It is therefore concluded that there is reasonably sufficient experience on CDF quantification and thereby to develop a risk criterion. Furthermore, the probabilistic criterion complements the deterministic rules.

X.4 What is the definition for this criterion (be precise)?

In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

CDF is defined as the expected number of events per calendar year that occur during power operation resulting in uncovery and heat-up of the reactor core to the point at which prolonged oxidation is anticipated and involving enough of the core to cause a significant release.

X.5 What is the supporting documentation for this criterion?

Please list and give a short abstract of each key publicly available document supporting the above criterion.

If no publicly available document exists, could you explain briefly how the criterion is supported?

The definition of the core damage is defined in a draft HSK regulatory guideline: HSK-A05 "Scope and Quality of a PSA" (currently out for public comment).

X.6 To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

Please explain why you chose this applicability.

The CDF criterion is set at a nuclear power reactor level.

X.7 How is the criterion expressed?

- Single value
- Band (limit and target)

Single value (mean value).

X.8 How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

For new nuclear power plants (NPP), the CDF criterion is considered as a legally strict value. For the construction permit of a new nuclear power plant, the applicant shall demonstrate that the core damage frequency is below E-5 per year.

This risk criterion is also expected to be fulfilled by the existing plants, to the extent that is reasonably achievable. This means that if an existing plant does not fulfil this requirement, the plant is expected to carry out a systematic investigation and evaluation of risk- and cost-effective measures to reduce the risk. Thus, for existing plants this criterion represents an orientation value.

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

The CDF shall be based on a risk quantification of all relevant internal and external events for full power operating mode of the reactor. (The risk of severe fuel damage in low power and shutdown modes of the reactor is covered by a separate risk measure called Fuel Damage Frequency (FDF) for which the risk criterion is in discussion.

10. How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

Since the mean value of the CDF is used for the criterion, the range of the uncertainty for the CDF is considered by definition of the criterion itself. (e.g.: a large uncertainty of the CDF increases the mean value.)

11. When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

The risk level of existing plants will be re-evaluated within the living PSA process. A detailed reassessment of the risk level (complete update of PSA) is required as integral part of Periodic Safety Review of the plant (usually once every ten years).

For a new NPP the risk level has to be evaluated by the applicant for getting the construction permit. This analysis shall be updated for the operational permit.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

If the criterion is exceeded:

- *for the existing NPPs, the licensees are expected to provide evaluation of risk and cost-effective measures to reduce the risk,*
- *for new NPPs the construction permit / operational permit will not be awarded.*

12. In case of band-defined goals, how is handled the case where the results are inside the band?

This question is not applicable to the considered risk criterion.

13. Have you defined other subsidiary criteria for PSA applications?

Further risk criteria are under discussion.

14. What is your experience with Probabilistic Risk Criteria?

Please provide a separate response for every experience you wish to share. These experience statements can address one or several of the following sub-criteria.

We do not expect a complete relation of your experience with probabilistic Risk Criteria, but a limited number of statements addressing the most important.

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

Every new or modified HSK regulatory guideline or ordinance is published for public comment. The utilities are involved in this process.

14.2 On consequences of implementation
Have Probabilistic Risk Criteria lead to system upgrades?

The assessment of the CDF initiated a number of backfits. Especially seismic resistance of structures and components as well as the emergency operation procedures have been improved.

14.3 On communication with the public

The risk of a nuclear power plant is in general something that the public is interested in. However, it is difficult to communicate in simple terms different risk measures (CDF or LERF) as well as different risk criteria for existing and new NPPs.

14.4 On interpretation

Examples of other areas of interest might be beneficial:

The introduction of risk criteria increased the need to harmonize the quality and scope of a PSA in order to make the PSA-results more comparable.

15. Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

This question is not applicable.

Response for Sweden (OKG)

Identification

Please identify your organisation:

Name: **OKG Aktiebolag**

Address: SE-572 83 Oskarshamn

Country: Sweden

Are you

- A Regulatory Body □
- A Supporting organisation to a regulatory body □
- A Utility ×
- A Vendor □

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1 Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1 Society level (qualitative)

1.2 Intermediate level (qualitative or quantitative)

1.3 Technical level (quantitative)

- *Core damage frequency shall be less than E-5 per year for severe core damage.*
- *Release shall be considerably lower than E-5 per year for release involving more than 0,1% (1800 MWt) of the core inventory excluding noble gases.*
- *An additional criterion for the Oskarshamn plants states that if the core damage frequency is within 10% from E-5 per year, then no initiating event family shall contribute more than E-6 per year; this criterion is usually not applicable.*

2. What technical level Probabilistic Risk Criteria have you developed?

Large early release frequency
Core damage frequency
Barrier strength

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

The criteria are the same for existing plants and life extension of those plants.

For Criterion 1, large early release frequency:

1.3 What is the reason why you developed this Probabilistic Risk Criterion?

1.4. What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

Release shall be considerably lower than E-5 per year for release involving more than 0,1% (1800 MWt) of the core inventory excluding noble gases.

The release criterion has been adapted to the power ratings of the Oskarshamn plants, i.e., it is set to 0,13 % for Oskarshamn 1, 0,07 % for Oskarshamn 2, and 0,055% for Oskarshamn 3.

1.5 What is the supporting documentation for this criterion?

For large releases, the safety goals were based on the government decisions regarding severe accident management measures [IndDep_1183/81] and [IndDep_2717/85].

The requirements locally applied at the Oskarshamn NPP were originally derived from the Sydkraft safety policy, and the E.ON safety policy still defines the basic levels. In addition more detailed local criteria for interpretation and judgement of PSA results have been developed [OKG_1996-00385]. They are referred to in the SAR for Oskarshamn but are expected soon to be deleted from the SAR.

Documents:

- *[IndDep_1183/81]:*
  Industridepartementet; Villkor för fortsatt tillstånd enligt 2 § atomenergilagen (1956:306) att driva atomreaktor; Industridepartementet 1183/81 1981-10-15; Industridepartementet; 1981

- *[IndDep_2717/85]:*
  Industridepartementet; Villkor för fortsatt tillstånd enligt 5 § lagen (1984:3) om kärnteknisk verksamhet för att driva kärnkraftreaktorerna Oskarshamn I, II och III; Industridepartementet 2717/85 1986-02-27 (dossier 8523); Industridepartementet; 1986

- *[OKG_1996-00385]:*
  Gunnarsson, Kerstin; Mål och strategi för modernisering avseende reaktorsäkerhet av OKG:s kärnkraftblock; OKG/96-00385; OKG AB; 1997

1.6 To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

The criterion is applicable at reactor level.

Please explain why you chose this applicability.

Government demand

1.7 How is the criterion expressed?

- Single value
- Band (limit and target)

Single value:

The release criterion has been adapted to the power ratings of the Oskarshamn plants, i.e., it is set to 0,13 % for Oskarshamn 1, 0,07 % for Oskarshamn 2, and 0,055% for Oskarshamn 3.

1.8 How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

The safety goals are used as a limit value for singling out situations that need to be further studied, i.e., as a trigger for starting analysis and evaluation of whether an identified plant condition is a safety problem. As such it is used as a safety indicator.

1.9 What is the scope of the analysis used for measuring plant performance against the criterion?

The PSA used to assess the criteria shall be full scope, i.e. include all initiating events and all operating modes.

<u>For Criterion 2, Core Damage Frequency:</u>

2.3 What is the reason why you developed this Probabilistic Risk Criterion?

Requirement from Sydkraft management board

2.4 What is the definition for this criterion (be precise)?

Core damage is defined as local fuel temperature above 1204 ºC, i.e., the limit defined in section 1b of 10 CFR 50.46, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.

The criterion is expressed as a yearly average:

- *Core damage frequency shall be less than E-5 per year.*

2.5 What is the supporting documentation for this criterion?

The requirements locally applied at the Oskarshamn NPP were originally derived from the Sydkraft safety policy, and the E.ON safety policy still defines the basic levels. In addition more detailed local criteria for interpretation and judgement of PSA results have been developed [OKG_1996-00385]. They are referred to in the SAR for Oskarshamn but are expected soon to be deleted from the SAR.

2.6 To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

The criterion is applicable at reactor level.

Please explain why you chose this applicability.

Decision from Sydkraft management board

2.7 How is the criterion expressed?

Single value
Band (limit and target)

The core damage frequency criterion is expressed as a single value; core damage frequency shall be less than E-5 per year.

2.8 How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

The safety goals are used as a limit value for singling out situations that need to be further studied, i.e., as a trigger for starting analysis and evaluation of whether an identified plant condition is a safety problem. As such it is used as a safety indicator.

2.9 What is the scope of the analysis used for measuring plant performance against the criterion?

The PSA used to assess the criteria shall be full scope, i.e. include all initiating events and all operating modes.

10 How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

11. When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

The policy states that the frequencies shall be used as a basis for assessing the severity of safety problems.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

Identified deviations are judged with respect to cause, e.g., if the cause is due to an incomplete or conservative PSA model or if it is due to a weakness in the plant. The basic procedure is that identified weaknesses are eliminated. However, if there is a major impact to the result from complex modelling issues, e.g., CCF, decisions on changes will be delayed until a more detailed description is available.

The internal guidelines for judgement of PSA results regarding impact on reactor safety includes the graded approach described in the IAEA document Safety Evaluation of Operating Nuclear Power Plants Built to Earlier Standards [IAEA_CB-5]. This document was never issued by the IAEA, but appeared in a somewhat reduced version in the safety report series as [IAEA_SRS_12]. The action levels specified for core damage frequency are:

- *PSA results >10^{-3} per year – immediate shutdown*
- *10^{-3} per year > PSA results > E-4per year – correction at next planned yearly shutdown*
- *E-4per year > PSA results > E-5 per year – long-term planning of actions*

12. In case of band-defined goals, how is handled the case where the results are inside the band?

No band defined goals are defined.

13. Have you defined other subsidiary criteria for PSA applications?

An additional criterion for the Oskarshamn plants states that if the core damage frequency is within 10% from E-5 per year, then no initiating event family shall contribute more than E-6 per year; this criterion is usually not applicable.

Additional probabilistic criteria have been defined, with a focus on assessment of the remaining system barrier after an initiating event. PSA results are presented based on the cause of the core damage (failure of shut-down systems, emergency core cooling or residual heat removal) and on a split-up of initiating events according to the event category they belong to (H2, H3, H4). This has worked well, especially for events with large uncertainties in initiator frequencies. Criteria have been defined according to an internal document [OKG_2005-14190]. For area events a procedure has been defined for assessing the acceptability of the system barrier against core damage [OKG_2006-09475].

Documents:

- *[OKG_2005-14190]*
- *Lindahl, Pär; Giltighet av probabilistiska säkerhetsmål; OKG/2005-14190; OKG AB; 2005*
- *[OKG_2006-09475]*
- *Lindahl, Pär; Metodbeskrivning för probabilistisk analys av rumshändelser med avseende på risken för härdskada/bränsleskada; OKG Anvisning 2006-09475; OKG AB; 14*

14. What is your experience with Probabilistic Risk Criteria?

In spite of problems in connection with discussion of high PSA results both internally and externally in media, the use of probabilistic safety goals has triggered a number of important safety improvements in the Oskarshamn plants (and previously at Barsebäck).

PSA has generally provided an aspect on safety that has been valuable for the total activities at the plants, but this has largely been achieved independently of the safety goals.

A general concern with probabilistic safety goals is the risk of the goals being seen as absolute limits, as this might indirectly have an impact on the quality and relevance of the PSA models.

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

See question number 14.

14.2 On consequences of implementation
Have Probabilistic Risk Criteria lead to system upgrades?

A number of major plant changes have been triggered by PSA results or involved PSA in the process. Important examples in both the Barsebäck plants and in Oskarshamn 1 and 2 are the improvement of cable separation in order to improve robustness with respect to area events, especially internal fires. In the FENIX project for Oskarshamn 1 (large-scale renovation 1993–95) the probabilistic criteria for plants built to earlier standards as defined in [IAEA_CB-5] were crucial for the decision by SKI to allow restart of the plant.

14.3 On communication with the public

14.4 On interpretation

15. Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

Response for Sweden (Ringhals)

Identification

Please identify your organisation:

Name: **Ringhals AB**

Address:

Country: Sweden

Are you

- A Regulatory Body □
- A Supporting organisation to a regulatory body □
- A Utility ×
- A Vendor □

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1 Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1 Society level (qualitative)

Information on nuclear safety is available on the public website http://www.vattenfall.se/ *:*

Säkerhetsarbetet på ett kärnkraftverk är uppbyggt i tre steg:

1. *Förebygga fel*
2. *Motverka att fel leder till haveri genom olika övervaknings- och säkerhetssystem*
3. *Lindra konsekvenser om fel uppstår, dvs förhindra radioaktiva utsläpp etc.*

1.2 Intermediate level (qualitative or quantitative)

Interpreted in "Övergripande mål och förhållningssätt för reaktorsäkerhet", id 1839723. Gives qualitative and quantitative goals. This document is a site-wide procedure for fulfilling the company goals for reactor safety

Deeper understanding for the qualitative and quantitative goals can be had from the attachment to this document.

1.3 Technical level (quantitative)

- *Core damage frequency shall be less than E-5 per year.*
- *Release shall be less than 1 E-7 for release involving more than 0,1% of the core inventory of substances causing ground contamination*

2. What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

1) *Large release*
2) *Core damage frequency*

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

The criterions are the same for existing plants as well as life extensions for existing plants.

For Criterion 1, Large Release:

1.3 What is the reason why you developed this Probabilistic Risk Criterion?

According to the SKI decision (beslut) 1988-12-19 "Utsläppsbegränsande åtgärder vid kärnkraftverken...", which is the formal acceptance of the activities and modifications performed to fulfil IndDep 2717/85 (government decision 17)

1.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

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 < 1 E-7 per year).

The criterion is expressed as a yearly average.

1.5 What is the supporting documentation for this criterion?

The criterion is based on the government requirement regarding large releases, interpreted as 1 E-7 per year [IndDep_2717/85].

Note: This frequency is not spelled out in any of the government decisions, neither in [SKI_SSI_1985]. All of these only refer to the concept of an "extremely unlikely event".

Documents:

- *[IndDep_2717/85]*
  Industridepartementet; Villkor för fortsatt tillstånd enligt 5 § lagen (1984:3) om kärnteknisk verksamhet för att driva kärnkraftreaktorerna Oskarshamn I, II och III; Industridepartementet 2717/85 1986-02-27 (dossier 8523); Industridepartementet; 1986

1.6 To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

The criterion is applicable at reactor level.

The license is per reactor.

1.7 How is the criterion expressed?

- Single value
- Band (limit and target)

The criterion is defined as a single value.

1.8 How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

The use of probabilistic analyses is generally discussed, and it is stated that the focus shall not be on absolute numerical results. Instead priority shall be on long-term safety improvements, identification of weaknesses and prioritisation of safety improvements. The criterion is there for considered to be a target value or a safety indicator. ("målvärde").

The value for releases is a legally-bound limit

1.9 What is the scope of the analysis used for measuring plant performance against the criterion?

The PSA used to assess the criteria shall be full scope, i.e. include all initiating events and all operating modes.

In some cases qualitative analysis or risk analysis are used in order to determine risks for specific tasks, on a go no go bases. (for instance handling of heavy loads during operation)

For Criterion 2, Core Damage Frequency:

2.3 What is the reason why you developed this Probabilistic Risk Criterion?

At the beginning, the risk criterion was not emphasised the earliest PSA for one of the reactors showed CDF 1E-7. After the publication of the draft INSAG-8, a value that was one order of magnitude better than existing reactors was adopted, i.e. E-5 to INSAG's E-4.

2.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

Core damage is defined as local fuel temperature above 1204 ºC, i.e., the limit defined in section 1b of 10 CFR 50.46, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.

The criterion is expressed as a yearly average:

- *Core damage frequency shall be less than E-5 per year.*

2.5 What is the supporting documentation for this criterion?

Ringhals document "Övergripande mål och förhållningssätt för reaktorsäkerhet", id 1839723. Gives qualitative and quantitative goals. This document is a site-wide procedure for fulfilling the company goals for reactor safety

Deeper understanding for the qualitative and quantitative goals can be had from the attachment to this document.

These criteria are in line with the levels defined in the draft IAEA guide CB-5 [IAEA_CB-5], but one order of magnitude stricter.

Documents:

- *[IAEA_CB-5]*
  IAEA; Safety Evaluation of Operating Nuclear Power Plants Built to Earlier Standards – A Common Basis for Judgement (draft version of IAEA Safety Reports Series No. 12); IAEA CB 5 (draft version of IAEA Safety Reports Series No. 12); IAEA; 1996

2.6 To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

The criterion is applicable at reactor level.

The license is per reactor

2.7 How is the criterion expressed?

- Single value
- Band (limit and target)

The core damage frequency criterion is expressed as a single value; core damage frequency shall be less than E-5 per year.

2.8 How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

The use of probabilistic analyses is generally discussed, and it is stated that the focus shall not be on absolute numerical results. Instead priority shall be on long-term safety improvements, identification of weaknesses and prioritisation of safety improvements. The criterion is there for considered to be a target value or a safety indicator ("målvärde").

2.9 What is the scope of the analysis used for measuring plant performance against the criterion?

The PSA used to assess the criteria shall be full scope, i.e. include all initiating events and all operating modes.

End of questions on each Probabilistic Risk Criterion

10. How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

Sensitivity analyses are part of the result presentation, but does not play any major role in measuring plant performance against the criterion.

If risk follow-up is considered, uncertainties are not considered – but in LPSA applications uncertainties are eliminated as far as possible.

11. When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

1. *Major plant modifications*
2. *Risk follow-up (done on a yearly bases)*

Also other risk-informed applications, such as in-service inspections, waivers and Tech Spec, use this.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

For interpretation af a probabilistic 'base' analysis, the following applies:

The procedure for judging deviations from safety goals is case specific, since the procedure depends on the size of the deviation. One important part of the procedure is to evaluate the conditions for the analysis.

To distinguish between what corrective action is necessary three levels with different required actions are defined;

- *If the CDF is > E-4 per year, immediate corrective actions for identified deviations are required. If this is not possible, the plant shall be immediately shut down.*
- *If the CDF is between E-4 per year and E-5 per year, the plant may remain in operation for a limited period of time. Temporary corrective actions are allowed while permanent safety enhancing measures are developed, designed and implemented.*
- *If the CDF is < E-5 per year, long term planning of safety enhancing measures is allowed, to be implemented in connection with plant modernisations.*

These criteria are in line with the levels defined in the draft IAEA guide CB-5 [IAEA_CB-5], but one order of magnitude stricter.

If a risk-follow up is used for a certain event (i.e. inoperability of certain valves), usually deterministic criteria overrides any probabilistic evaluations.

12. In case of band-defined goals, how is handled the case where the results are inside the band?

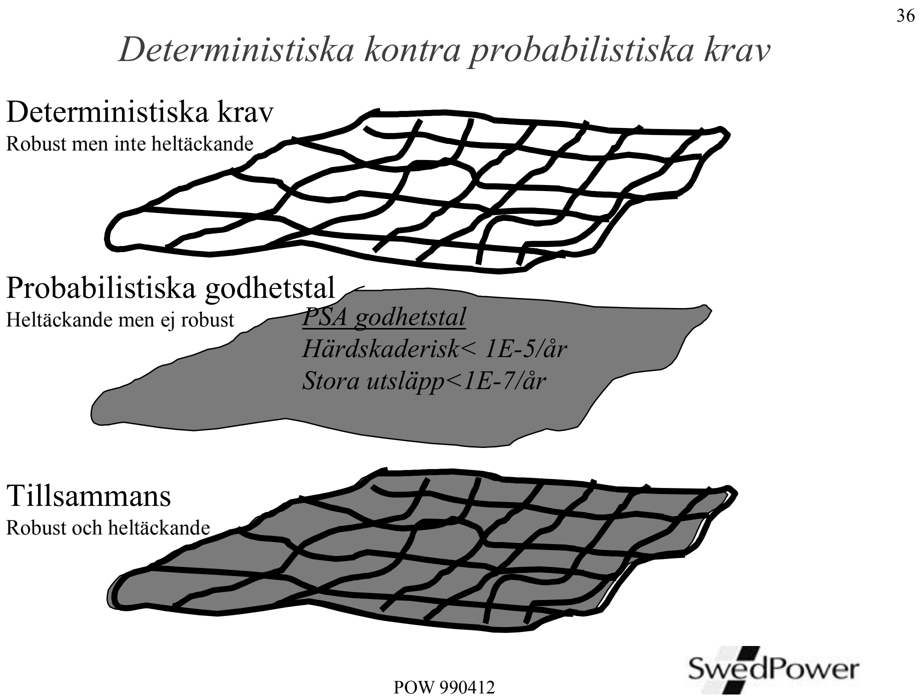
No band defined goals are used as safety goals.

13. Have you defined other subsidiary criteria for PSA applications?

Yes – CDF should have a balanced risk profile with no dominating risks. Releases are defined in different categories (LERF etc).

14. What is your experience with Probabilistic Risk Criteria?

At Ringhals, the first probabilistic studies were used for evaluation of dependencies in the plant and plant line-up/ mode of operation. The use and usefulness of PSA was limited due to model restrictions. Further updates, with improved models with extended mapping of dependencies, have given added knowledge of the weak spots in the plant.



The current view is that PSA in the right context and accompanied by other relevant information (deterministic analyses, human reliability analyses, and operating experiences) gives a valuable contribution to safety analysis, and PSA has become an integrated part of the total safety analysis concept.

This relates to PSA as such rather than to safety goals, but safety goals have also to some extent contributed to an increased awareness of the usefulness of PSA. At an earlier stage, they are also believed to have had a slightly repellent effect, mainly because of a fear that exceedances might lead to unreasonable requirements on implementation of safety improvements, and that such actions might then be based on crude assumptions and prerequisites. An important background to this concern is the fact that previous updates and extensions of PSAs have resulted in large variations of results both regarding the total CDF or release frequency and the distribution between different groups of initiating events. These concerns still exist to some extent. For this reason, the view is that safety goals are mainly to be used as indicators showing that changes made point in the right direction, and that they can be useful as guidelines in the safety work.

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

Answer included in question number 14.

14.2 On consequences of implementation
Have Probabilistic Risk Criteria lead to system upgrades?

At the sites, safety goals have not had a decisive impact. The focus has been on the identification and importance ranking of dominating contributors. PSA has also been an efficient tool for identification of functional dependencies. However, one perceived positive effect of the safety goals is that they have increased the focus on the correctness of the PSA models. Another experience is that the quality requirements on PSA increase in risk-informed applications. In discussion with the SKI, safety goals have never had any importance.

14.3 On communication with the public

In our opinion, probabilistic analyses and results cannot be communicated with the public in any plausible way. One of the reasons for this is probably that risk perception by the public is not by frequency, but by number of people involved. I.e. the consequence is dominant.

14.4 On interpretation

There have been discussions on whether it is meaningful to use two safety goals or not. It is perceived as easier to communicate only core damage frequency both internally and externally, since this safety goal is closer to the technique. It can be difficult to relate to release as a safety goal due to the extremely low probability.

15. Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

Response for Sweden (SKI)

Identification

Please identify your organisation:

Name: **Statens Kärnkraftsinspektion, SKI**

Address: Visit: Klarabersviaduktion 90, SE-106 58 Stockholm

Country: Sweden

Are you

- A Regulatory Body ×
- A Supporting organisation to a regulatory body
- A Utility
- A Vendor

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1. Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

The Swedish regulatory tradition is mainly non-prescriptive, meaning that often high-level requirements are given, while the exact ways to fulfil the requirements is left to the licensees to decide. An important aim of SKI inspections is to maintain confidence in the fulfilment of requirements. Except for the explicit safety goal for release, the SKI has not defined any safety goals.

1.1. Society level (qualitative)

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.

1.2 Intermediate level (qualitative or quantitative)

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.*
- *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.*
- *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.*

1.3 Technical level (quantitative)

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 < 1 E-7 per year).

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.

2. What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

1) Large release

Except for the implicit numerical safety goal on large release, the SKI has not defined any safety goals. However, the newly issued regulation concerning safety in nuclear facilities [SKIFS 2004:1] requires the licensees to have clearly defined goals for their activities.

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

The same requirements on maximum acceptable release of radioactive substances apply to all NPPs, regardless of location. The justification for this requirement is that the same level of individual risk shall be achieved at all sites, regardless of population density, property values or numbers of reactors at one site.

For Criterion 1, Large Release:

1.3 What is the reason why you developed this Probabilistic Risk Criterion?

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. The criterion for large release was defined after discussions regarding severe accidents (for justification se question 1.5 below).

1.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

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 < 1 E-7 per year).

1.5 What is the supporting documentation for this criterion?

As part of the background description and justification for the selected release criterion, The Swedish Radiation Protection Authority (SSI) presented a comparison between the fatality risk from exposure to radon in habitations to other risks in society.

The 0,1 % criterion is also justified by the argument that the requirement on the filtering capacity of the filtered venting systems to be installed should not exceed the level of diffuse leakage that is to be expected.

The quantification of the frequency requirement, i.e., converting "extremely low probabilities" into a frequency of occurrence, was done by relating to the concept of residual risk. In [SKI_SSI_1985], reactor vessel rupture is given as an example of a residual risk[11]. Based on the quantification of this event in WASH-1400, this was interpreted by both the SKI and the licensees to correspond to an event with a frequency of about 1 E-7 per year. However, this frequency is not spelled out in any of the government decisions, neither in [SKI_SSI_1985].

Documents:

- [SKI_SSI_1985]: SKI / SSI; Utsläppsbegränsande åtgärder vid svåra härdhaverier; SKI ref 7.1.24 1082/85; SKI / SSI; 1985

1.6 To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

The criterion is set at reactor level. The justification for this requirement is that the same level of individual risk shall be achieved at all sites, regardless of population density, property values or numbers of reactors at one site.

[11] In SKIFS 2004:2, the definition is given as "Extremely improbable events (residual risks). Events that are so improbable that they do not need to be taken into account as initiating events in connection with safety analysis."

1.7 How is the criterion expressed?

- Single value
- Band (limit and target)

The criterion is expressed by a single value.

1.8 How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

The safety goals are considered to be target values (orientation values). Exceedance of safety goals is allowed, but should be accompanied by an evaluation stating the reason for the exceedance and if needed a plan for correction.

1.9 What is the scope of the analysis used for measuring plant performance against the criterion?

In the view of the SKI, safety goal defined on the level of core damage or large release should reasonably cover the complete spectrum of risks as calculated in a full-scope PSA, i.e., all categories of initiating events and all plant operating modes.

10. How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

11. When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

The licensees are required to have a safety policy, and if the policy includes probabilistic safety goals the licensee is, in principle, expected to fulfil these goals.

At situations when the safety goals of a licensee are exceeded, SKI requires an explanation of that situation. The requested statement has to be preliminary and independently reviewed. To exceed the safety goals may lead to a plant modification.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

The SKI does not desire a situation where actions need automatically to be taken because of the violation of a safety goal, i.e., it is not judged to be feasible to treat safety goals as absolute acceptance limits. Fluctuations in PSA results over time are unavoidable (updates, extension) and sharp acceptance criteria might in fact be counter-productive. Therefore it is judged reasonable to see safety goals as target values, and to treat exceedances as triggers for further analysis or planning of safety enhancing actions.

While the fulfilment of the safety goals defined by the licensees is basically mandatory, the actual procedure is more flexible. Exceedance of safety goals is allowed, but should be accompanied by an evaluation stating the reason for the exceedance and if needed a plan for correction.

Exceedance of safety goals is normally not a problem in safety related activities, and can often be justified (uncertainty, conservative approach, etc.). However, an exceedance can be complicated to communicate to the public, and may also be a problem due to the general requirement that licensees are expected to fulfil their safety policy. Deviations which are not handled may cause doubts regarding the self inspection of the licensee.

12. In case of band-defined goals, how is handled the case where the results are inside the band?

No band-defined goals are defined.

13. Have you defined other subsidiary criteria for PSA applications?

14. What is your experience with Probabilistic Risk Criteria?

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

PSA results are at present (2007) not used very actively at SKI, but more PSA applications are expected in the near future.

SKI is not itself performing any PSA modelling or calculations.

To date, some issues have been supported by PSA, and SKI has sometimes had PSA evaluation as a condition for acceptance. However, this is still done on a rather small scale. The degree to which PSA related information is considered in SKI decisions depends on the perceived degree of importance of the information.

14.2 On consequences of implementation
Have Probabilistic Risk Criteria lead to system upgrades?

PSA results and fulfilment of safety goals has been important in some applications and influenced the decision taken by the SKI, e.g.,

- *in the FENIX project for restart of Oskarshamn 1.*
- *in the AFSIE project at Oskarshamn 2*
- *in the Barsebäck 2 event, 1992*
- *in several modernisation projects through the years*

14.3 On communication with the public

Exceedances of safety goals can be complicated to communicate to the public. It is therefore important not to exceed the goals or in case of exceedances to present justifications (uncertainty, conservative approach, etc.).

An actual such a tricky situation, was after the Forsmark 1 event in 2006.

14.4 On interpretation

A general view is that the evaluation of results needs to be more efficient, which is even more important in view of the fact that safety goals are not absolute. There is a need to break down the top level safety goals to make them useful for more detailed applications.

15. Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

This question is already answered, see e.g., question nr 1.

At present (2007), there are no plans to express any more exact probabilistic criteria, other than already are expressed in the regulations of SKI.

Response for Chinese Taipei

Please identify your organisation:

Name: **Institute of Nuclear Energy Research**

Address : P.O. Box 3-3, Lungtan, Taoyuan, Chinese Taipei 325

Country : Chinese Taipei

Are you

- A Regulatory Body ☐
- A Supporting organisation to a regulatory body ☑
- A Utility ☐
- A Vendor ☐

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

Task 2006-2 addresses Probabilistic Risk Criteria. This scope is wider than is generally understood as "Safety Goals" and includes lower level criteria, such as regulatory targets for the plant systems or requirements on Fussel-Vessely importance for equipment.

However, for maintaining this questionnaire to a manageable level, detailed Probabilistic Risk Criteria are excluded from its scope.

They is addressed by the "Performance Indicators" project (NEA/CNRA/R(2006)-1.

1. Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

There is no probabilistic risk criterion in Chinese Taipei.

15. Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

For the time being, there is no any project or plan to set probabilistic risk criteria for nuclear power plant in Chinese Taipei.

Response for UK

Identification

Please identify your organisation:

Name: **Nuclear Installations Inspectorate (Health and Safety Executive)**

Address : Redgrave Court, Merton Road, Bootle L20 7HS

Country : United Kingdom

Are you

- A Regulatory Body YES
- A Supporting organisation to a regulatory body □
- A Utility □
- A Vendor □

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1. Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1 Society level (qualitative)

The HSE's Safety Assessment Principles, SAPs (http://www.hse.gov.uk/nuclear/saps/saps2006.pdf) provide UK nuclear inspectors with a framework for making consistent regulatory judgements on nuclear safety cases. The SAPs also provide nuclear site duty-holders with information on the regulatory principles against which their safety provisions will be judged.

HSE's SAPs (2006 Edition) include the following fundamental principles (paragraph 42):

- *FP.3 Protection must be optimized to provide the highest level of safety that is reasonably practicable.*
- *FP.5 Limitation on risks to individuals: "Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm"*
- *FP.6 Prevention of accidents: "All reasonably practicable steps must be taken to prevent and mitigate nuclear or radiation accidents"*
- *FP.8 Protection of present and future generations: "People, present and future, must be protected against radiation risks"*

The 2006 SAPs are consistent with "Reducing risks protecting people: HSE's decision making process (R2P2)" (http://www.hse.gov.uk/risk/theory/r2p2.pdf), which provides an overall framework for decision making based on the demonstration by the duty-holders that the risk is as low as reasonably practicable (ALARP), as required by UK Health & Safety Law.

Technical Assessment Guides (TAGs) support the SAPs by providing more detailed guidance to NII inspectors for the assessment of safety submissions. Of particular relevance here is "NSD guidance on the demonstration of ALARP", T/AST/005, Issue 3, www.hse.gov.uk/foi/internalops/nsd/tech_asst_guides/tast005.pdf

1.2 Intermediate level (qualitative or quantitative)

1.3 Technical level (quantitative)

Detailed numerical targets are established in the UK for NII Inspectors to use when judging whether the duty holder is controlling radiological hazards adequately and reducing risks ALARP. These are described in "Safety Assessment Principles for Nuclear Facilities", 2006 Edition, HSE, www.hse.gov.uk/nuclear/saps/saps2006.pdf (paragraphs 594 to 628).

These targets are explained in a public document entitled "Numerical targets and legal limits in Safety Assessment Principles for Nuclear Facilities, An explanatory note", December 2006, HSE, www.hse.gov.uk/nuclear/saps/explanation.pdf

Of particular relevance here are:

Target 5: Individual risk of death from on-site accidents – any person on the site

Target 6: Frequency dose targets for any single accident – any person on the site

Target 7: Individual risk to people off the site from accidents

Target 8: Frequency dose targets for accidents on an individual facility – any person off the site

Target 9: Total risk of 100 or more fatalities

The above targets are discussed further below (2.1).

2. What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

Introductory note: *The structure of the targets included in the SAPs is based on the TOR framework (http://www.hse.gov.uk/nuclear/tolerability.pdf) which has been extended in the more recent R2P2 (http://www.hse.gov.uk/risk/theory/r2p2.pdf), In assessing the safety of nuclear facilities, inspectors should examine the safety case to judge the extent to which targets are achieved. The Basic Safety Level (BSL) and the Basic Safety Objective (BSO) are used in translating the TOR (R2P2) framework into targets. The BSO marks the start of the broadly acceptable level in R2P2. 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.*

1. Individual risk of death from on-site accidents – any person on site (Target 5 of SAPs 2006-para. 606)

The targets for the individual risk of death to a person on the site, from on-site accidents that result in exposure to ionising radiation, are:

BSL: 1 x E-4pa
BSO: 1 x E-6 pa

2. Frequency dose targets for any single accident–any person on the site (Target 6 of SAPs 2006-para. 606)

The targets for the predicted frequency of any single accident in the facility, which could give doses to a person on the site, are: **Effective dose, mSv**	**Predicted frequency per annum**	
	BSL	**BSO**
2 – 20	1×10^{-1}	1×10^{-3}
20 – 200	1×10^{-2}	1×10^{-4}
200 – 2000	1×10^{-3}	1 x E-5
> 2000	1 x E-4	1×10^{-6}

3. Individual risk to people off the site from accidents (Target 7 of SAPs 2006 – para. 615)

The targets for the individual risk of death to a person off the site, from on-site accidents that result in exposure to ionising radiation, are:

BSL: 1 x E-4pa

BSO: 1 x E-6 pa

4. Frequency dose targets for accidents on an individual facility – any person off the site (Target 8 of SAPs 2006 – para. 617

The targets for the total predicted frequencies of accidents on an individual facility, which could give doses to a person off the site, are: **Effective dose, mSv**	**Total predicted frequency per annum**	
	BSL	**BSO**
0.1 – 1	1	1×10^{-2}
1 – 10	1×10^{-1}	1×10^{-3}
10 – 100	1×10^{-2}	1×10^{-4}
100 – 1000	1×10^{-3}	1 x E-5
> 1000	1 x E-4	1×10^{-6}

5. Societal risk – total risk of 100 or more fatalities (Target 9 of SAPs 2006 – para. 623)

The targets for the total risk of 100 or more fatalities, either immediate or eventual, from on-site accidents that result in exposure to ionising radiation, are:

BSL: 1 x E-5 pa

BSO: 1 x 1 E-7 pa

Details for the above targets are presented in SAPs 2006 paragraphs 585 to 628.

Are these criteria the same for existing plants, life extension of existing plants, new builds, and new designs?

The above targets apply to existing plants, life extension of existing plants and new builds and new designs, and use of the ALARP principle will determine the level that must be achieved on a case by case basis.

HSE policy is that new facilities or activities should at least meet the BSLs, with the expectation that application of ALARP will result in lower risks, again on a case by case basis. BSLs provide benchmarks for existing facilities.

For Criterion X:

The following examples will consider the CDF criterion

X.3 What is the reason why you developed this Probabilistic Risk Criterion?

The rationale behind the Targets is described in "Numerical targets and legal limits in Safety Assessment Principles for Nuclear Facilities, An explanatory note", December 2006, HSE, www.hse.gov.uk/nuclear/saps/explanation.pdf

Basically, the BSLs and BSOs translate the Tolerability of Risk (TOR) framework (http://www.hse.gov.uk/nuclear/tolerability.pdf) and guide decision making by inspectors. HSE policy is that the BSLs indicate risks which new facilities should meet and they provide benchmarks for existing facilities. The BSOs have been set at a level where HSE considers it not to be a good use of its resources or taxpayers' money, nor consistent with a proportionate regulatory approach, to pursue further improvements in safety. In contrast, licensees have an overriding duty to consider whether they have reduced risks to as low as reasonably practicable (ALARP) on a case by case basis irrespective of whether the BSOs are met. As such, it will in general be inappropriate for licensees to use the BSOs as design targets, or as surrogates to denote when ALARP levels of dose or risk have been achieved.

X.4 What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

Definitions are given in SAPs 2006, paragraphs 585 to 628. The Risk Targets of the SAPs are given as frequencies based on annual averages. In doing this, there is an implicit assumption that risk is fairly uniform over a year. Since, in reality, there are exceptions to this, some of which can be significant, the 2006 SAPs include guidance for dealing with time at risks situations (paragraphs 629 – 638). In this regard, it would not be generally acceptable (as a sole safety justification) to compare, against the Targets of the SAPs, large risks over a short periods averaged over a year

X.5 What is the supporting documentation for this criterion?

Please list and give a short abstract of each key publicly available document supporting the above criterion.

If no publicly available document exists, could you explain briefly how the criterion is supported?

The following publicly available references explain the background and reasons for all the Targets.

"Safety Assessment Principles for Nuclear Facilities", 2006 Edition, HSE, www.hse.gov.uk/nuclear/saps/saps2006.pdf

"Numerical targets and legal limits in Safety Assessment Principles for Nuclear Facilities, An explanatory note", December 2006, HSE, www.hse.gov.uk/nuclear/saps/explanation.pdf

"Tolerability of risk from nuclear power stations"

www.hse.gov.uk/nuclear/tolerability.pdf

X.6 To what is the criterion applicable?

- A reactor,
- A plant (multiple reactors sharing at least one safety system),
- A site (several reactors on the same location),
- A population of reactors (all the reactors in the country)?

Target 5 (Individual risk of death from on-site accidents – any person on the site) applies to all the facilities on a site

Target 6: Frequency dose targets for any single accident – any person on the site) applies to single facilities

Target 7 (Individual risk to people off the site from accidents) applies to all the facilities on a site

Target 8 (Frequency dose targets for accidents on an individual facility – any person off the site) applies to single facilities

Target 9 (Total risk of 100 or more fatalities) applies to all the facilities on a site

X.7 How is the criterion expressed?

- Single value
- Band (limit and target)

The numerical targets of the SAPs are expressed as bands with a Basic Safety Level (BSL) and a Basic Safety Objective (BSO). These are explained in 2.1 above. Further information can be found in SAPs 2006 paragraphs 568 to 573.

X.8 How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

The targets shown above are not legal limits but are targets that inspectors use when judging whether a duty holder is controlling radiological hazards and reducing risks ALARP. They are essentially guides to inspectors to indicate where there is need for consideration of additional safety measures.

X.9 What is the scope of the analysis used for measuring plant performance against the criterion?

SAPs 2006 (SAP FA10, paragraph 528) state that a suitable and sufficient PSA should be performed. Paragraph 528 clarifies that the scope and depth of PSA may vary depending on the magnitude of the radiological hazard and risks, the novelty of the design, the complexity of the facility, and the nature of the decision that the safety case is supporting. For example, for some facilities qualitative arguments, application of good practice and DBA may be sufficient to demonstrate that the risk is ALARP. However, for a complex facility such as a power reactor or a reprocessing facility, a comprehensive PSA should be developed. Therefore, the PSA for NPPs should include internal and external events, full power and shutdown operating modes. The results of this PSA should be compared against the numerical targets of the SAPs.

In fact, paragraph 12 of report "Numerical targets and legal limits in Safety Assessment Principles for Nuclear Facilities, An explanatory note" (www.hse.gov.uk/nuclear/saps/explanation.pdf) indicates that the BSLs and BSOs in Targets 5 to 8 have been set at a level judged appropriate for a full-scope PSA (i.e. one in which all qualifying faults at the site/facility are included). If a reduced-scope PSA is to be assessed then these BSLs and BSOs will need to be adjusted accordingly. Similarly, inspectors may need to apply other adjustments to these Targets to take account of aspects of the licensee's methodology that differ from what was assumed when these Targets were set (cf paragraphs 576 and 608).

NII's Technical Assessment Guide (TAG) on PSA (http://www.hse.gov.uk/foi/internalops/nsd/tech_asst_guides/tast030.pdf) provides guidance to NII inspectors for assessing PSAs for nuclear facilities in general. This TAG is currently being revised and the new version will soon be made available in HSE's website. The most important modification to this TAG is the addition of an Appendix on "Assessment expectations for review of PSAs for modern nuclear power plants". This appendix covers PSA Level 1, 2 and 3, internal initiating events, internal and external hazards, and full power low power and shutdown.

End of questions on each Probabilistic Risk Criterion

10. How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

SAPs 2006 state the following:

- *Assumptions made regarding the behaviour of the facility or its operators should be justified, and the sensitivity to those assumptions should be assessed (paragraph 538 of Section on Probabilistic Safety Analysis).*
- *Due regard should be given to the uncertainties in input probability and frequency values used, and their impact on the results (paragraph 539 of Section on Probabilistic Safety Analysis).*
- *When comparing duty holder estimates with targets, inspectors should take account of the assumptions and limitations of the analysis used (paragraph 574 of Section on "Applying the targets and legal limits").*
- *The uncertainties in the duty holder's safety analyses, and claims of accuracy and precision in numerical estimates, should be assessed, e.g. through sensitivity analysis, as appropriate (paragraph 575 of Section on "Applying the targets and legal limits").*

NII's Technical Assessment Guide (TAG) on PSA (

http://www.hse.gov.uk/foi/internalops/nsd/tech_asst_guides/tast030.pdf)

provides guidance to NII inspectors for assessing PSAs for nuclear facilities in general.

Section 3.5 of this draft TAG indicates the following:

- *In all aspects of the analysis where assumptions have been made about how the plant and the operating staff behave, these and their justification should be clearly described. The sensitivity of the results of the PSA to changes in assumptions should be evaluated and clearly documented.*
- *Uncertainty on input probability and frequency values should be estimated and propagated through the models to generate uncertainty distributions on the resulting frequencies or probabilities of undesired events. The means of these distributions should be compared against the numerical targets in the SAPs.*
- *Based on the importance, sensitivity and uncertainty evaluations, the duty-holder should gain an understanding of which parametric and modelling uncertainties contribute most to the overall uncertainty in the probabilities or frequencies of undesired events and should, subject to reasonable practicability, take steps to reduce such uncertainties.*
- *Ultimately, the results of the uncertainty and sensitivity evaluations should provide confidence that the overall conclusions obtained from the PSA are still valid.*

11. When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

This is covered in the SAPs, see FA14 and paragraphs 541 and 542

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

Identify engineering or operational features responsible for missing the target and pursue improvement to the plant and the PSA

12 In case of band-defined goals, how is handled the case where the results are inside the band?

Require demonstration that there are no further reasonably practicable improvements (see T/AST/005 for details) that could reduce the risks

13. Have you defined other subsidiary criteria for PSA applications?

No but any such criteria used by licensees will need to be justified by them

14. What is your experience with Probabilistic Risk Criteria?

Please provide a separate response for every experience you wish to share. These experience statements can address one or several of the following sub-criteria.

We do not expect a complete relation of your experience with probabilistic Risk Criteria, but a limited number of statements addressing the most important.

Has evolved over a period of years. The criteria are seen by NII as helping to identify potential weaknesses and focus regulatory efforts rather than absolutes (see responses to 11.2 & 12 above).

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

14.2 On consequences of implementation
Have Probabilistic Risk Criteria lead to system upgrades?

Yes, but they focus attention on potential weaknesses of the engineering or operational features and the emphasis is on improving those rather than simply getting lower numbers. We do expect the PSA to be used as part of the justification for such improvements and the numerical results adjusted accordingly.

14.3 On communication with the public

Both the SAPs, the PSA TAG and the explanatory note on numerical targets have been published.

14.4 On interpretation

Examples of other areas of interest might be beneficial:

15. Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

Response for USA

Identification

Please identify your organisation:

Name: **U.S. Nuclear Regulatory Commission** (http://www.nrc.gov/)

Address: Two White Flint North, 11545 Rockville Pike, Rockville, MD 20852

Mailing Address: U.S. Nuclear Regulatory Commission, Washington, DC 20555

Country: United States of America

Are you

- A Regulatory Body x
- A Supporting organisation to a regulatory body □
- A Utility □
- A Vendor □

While the following questionnaire presents some alternatives for the responses that could be answered by ticking, please comment and/or provide background on the rationales for your responses.

1. Have your organisation (Regulatory Body, Utility) developed Probabilistic Risk Criteria? Please describe and give reference to publicly available document.

1.1 Society level (qualitative)

The mission of the U.S. Nuclear Regulatory Commission (NRC) is to "license and regulate the Nation's civilian use of by-product, source, and special nuclear materials to ensure adequate protection of public health and safety, promote the common defence and security, and protect the environment." http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1614/v4/index.html#mission

In 1986, the NRC issued a policy statement entitled "Safety Goals for the Operation of Nuclear Power Plants." (http://www.nrc.gov/reading-rm/doc-collections/commission/policy/51fr30028.pdf) The safety goals focus on the risks to the public from nuclear power plant operation. The goals broadly define an acceptable level of radiological risk, and consist of two qualitative safety goals which are supported by two quantitative objectives.

The qualitative safety goals are as follow:

- *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.*
- *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.*

1.2 Intermediate level (qualitative or quantitative)

In the Safety Goal Policy Statement, the NRC established two quantitative objectives that were to be used to determine achievement of the qualitative safety goals. These quantitative objectives are as follows:

- *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.*

1.3 Technical level (quantitative)

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-04 and the LERF measure is 1E-05. 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. (See Appendix D to NUREG-1860, http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1860/v2/sr1860v2.pdf)

2. What technical level Probabilistic Risk Criteria have you developed?

Please list these criteria

- *Core damage frequency*
- *Large early release frequency*

Are these criteria the same for existing plants, life extension of existing plants, new builds, new designs?

For operating plants (existing plants) including license renewal for operating plants, these criteria are the same. For new or advanced nuclear power plants, the NRC expects a higher level of severe accident safety performance consistent with the NRC's Severe Accident Policy Statement. (http://www.nrc.gov/reading-rm/doc-collections/commission/policy/50fr32138.pdf)

	CDF	LERF	Conditional Containment Failure Probability
Operating Plants & License Renewal	<1E-04	<1E-05	n/a
New Plants	<1E-04	<1E-06	<0.1

Please answer the following questions for each of the listed criteria, limiting the responses to criteria at or above system level.

3. What is the reason why you developed this Probabilistic Risk Criterion?

Please elaborate on the reasons

The probabilistic risk criteria were developed to provide a measure to assess the safety of nuclear power plants and to evaluate adherence to the Commission's Safety Goal Policy.

4. What is the definition for this criterion (be precise)?
In particular, indicate whether the criterion applies to average (for instance yearly) or instantaneous risk.

Core Damage Frequency is defined as the likelihood that, given the way a reactor is designed and operated, an accident could cause the fuel in the reactor to be damaged.

Large Early Release Frequency is defined as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects. Such accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and a loss of containment isolation.

5. What is the supporting documentation for this criterion?

Please list and give a short abstract of each key publicly available document supporting the above criterion

Over the past 20 years, the U.S. Nuclear Regulatory Commission (NRC) has continuously increased the use of risk analysis practices in our approach to regulation. NRC's guidance relating to risk analysis practices is provided in various types of documents. These types of documents include Policy Statements, Regulatory Guides, Standard Review Plans, and NUREGs. In addition, the NRC works closely with various national consensus organizations in the development of standards. When found acceptable, the NRC endorses these consensus standards for use in meeting NRC requirements. The following summarizes these types of documents and provides multiple examples of each.

1) *Policy Statements - The Commission issues policy statements to provide its consolidated views and expectations with regard to how the NRC will approach or treat a particular issue.*

   a. *Safety Goals for the Operations of Nuclear Power Plants, August 1986*

   i. *This policy statement focuses on the risks to the public from nuclear power plant operation. Its objective is to establish goals that broadly define an acceptable level of radiological risk. The Commission established two qualitative safety goals which are supported by two quantitative objectives. The two supporting objectives are based on the principle that nuclear risks should not be a significant addition to other societal risks.*

   ii. *http://www.nrc.gov/reading-rm/doc-collections/commission/policy/51fr30028.pdf*

   b. *Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities, August 1995*

   i. *This policy statement presents the policy that the NRC follows in the use of probabilistic risk assessment (PRA) methods in nuclear regulatory matters. The Commission established this policy statement so that the many potential applications of PRA can be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency. In addition, the Commission believes that the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach.*

   ii. *http://www.nrc.gov/reading-rm/doc-collections/commission/policy/60fr42622.pdf*

c. *Commission White Paper on Risk-Informed and Performance-Based Regulation, March 1999*

   i. *This white paper was issued to provide a collective view of what is meant by the terms "risk-informed, "risk-based" and "performance-based" regulation. It is intended to be used as guidance to help ensure consistent interpretation of these terms and implementation of the Commission's expectations with respect to their use in regulatory activities.*

   ii. *http://www.nrc.gov/reading-rm/doc-collections/commission/srm/1998/1998-144srm.html*

2) *Regulatory Guides – The NRC develops and issues regulatory guides to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the NRC staff in its review of applications for permits and licenses. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.*

   a. *Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002*

      i. *This regulatory guide provides guidance on the use of PRA findings and risk insights in support of licensee requests for changes to a plant's licensing basis, such as requests for license amendments and technical specification changes for nuclear power plants.*

      ii. *http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-174/01-174.pdf*

   b. *Regulatory Guide 1.175, "An Approach for Plant-Specific, Risk-Informed Decision-Making: In-service Testing," August 1998*

      i. *This regulatory guide addresses in-service testing of pumps and valves and provides a framework for incorporating risk insights into the regulation of nuclear power plants.*

      ii. *http://adamswebsearch2.nrc.gov/idmws/ViewDocByAccession.asp?AccessionNumber=ML003740149*

   c. *Regulatory Guide 1.176 "An Approach for Plant-Specific, Risk-Informed Decision-Making: Graded Quality Assurance," August 1998*

      i. *This regulatory guide describes an acceptable approach for identifying the safety significance of systems, structures, and components in nuclear power plants, and assigning quality assurance controls accordingly to ensure that quality assurance requirements are being graded commensurate with safety.*

      ii. *http://adamswebsearch2.nrc.gov/idmws/ViewDocByAccession.asp?AccessionNumber=ML003740172*

   d. *Regulatory Guide 1.177 "An Approach for Plant-Specific, Risk-Informed Decision-Making: Technical Specifications," August 1998*

      i. *This regulatory guide describes acceptable methods for assessing the nature and impact of proposed technical specification changes by considering engineering issues and applying risk insights. Applicants submitting risk information are expected to address each of the principles of risk-informed regulation discussed in this regulatory guide.*

      ii. *http://adamswebsearch2.nrc.gov/idmws/ViewDocByAccession.asp?AccessionNumber=ML003740176*

e. *Regulatory Guide 1.178 "An Approach for Plant-Specific, Risk-Informed Decision-Making for In-service Inspection of Piping," August 1998*

   i. *This regulatory guide focuses on the use of PRA in support of a risk-informed in-service inspection program. The guide provides guidance on acceptable approaches to meeting the existing ASME Section XI requirements for the scope and frequency of inspection of ISI programs.*

   ii. *http://adamswebsearch2.nrc.gov/idmws/ViewDocByAccession.asp?AccessionNumber=ML003740181*

f. *Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," January 2007*

   i. *This regulatory guide describes an approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors. This guidance is intended to be consistent with the NRC's PRA Policy Statement and subsequent, more detailed, guidance in Regulatory Guide 1.174. It is also intended to reflect and endorse guidance provided by standards-setting and nuclear industry organisations.*

   ii. *http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-200/01-200r1.pdf*

g. *Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006*

   i. *This regulatory guide provides guidance for complying with the NRC's requirements in §50.69, by using the process described in Revision 0 of NEI 00-04 to determine the safety significance of structures, systems, and components and place them into the appropriate risk informed safety class. The safety significance of SSCs is determined using an integrated decision-making process, which incorporates both risk and traditional engineering insights.*

   ii. *http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-201/01-201r1.pdf*

h. *Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," May 2006*

   i. *This regulatory guide provides guidance for use in complying with the NRC requirements for risk-informed, performance-based fire protection programs that meet the requirements of Title 10, Section 50.48(c), of the Code of Federal Regulations (10 CFR 50.48(c)) and the referenced 2001 Edition of the National Fire Protection Association (NFPA) standard, NFPA 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants."*

   ii. *http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-205/01-205.pdf*

i. *Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants," June 2007*

   i. *This regulatory guide provides guidance regarding the information to be submitted in a combined operating license application for a nuclear power plant. In particular, Section C.I.19, "Probabilistic Risk Assessment and Severe Accident Evaluation," addresses the information that an applicant should submit to enable the NRC staff to determine the acceptability of the risks to public health and safety associated with operation of a proposed new plant. The acceptability of the risks to public health and safety is determined from the interpretation of the results and insights of the applicant's (1) plant-specific PRA1 and (2) severe accident evaluations.*

ii. *http://adamswebsearch2.nrc.gov/idmws/doccontent.dll?library=PU_ADAMS^PBNTAD01&ID=071710180*

3) *Standard Review Plans - The Standard Review Plan (SRP) provides guidance to NRC staff in performing safety reviews of applications for permits or operating licenses. The principal purpose of the SRP is to assure the quality and uniformity of NRC safety reviews. It is also the intent of the SRPs to make information about regulatory matters widely available, and to improve communication between the NRC, interested members of the public, and the nuclear industry, thereby increasing understanding of the NRC's review process.*

a. *NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," June, 2007*

i. *This SRP chapter pertains to the NRC review of the design-specific PRA for a design certification and plant-specific PRA for a combined license application.*

ii. *http://adamswebsearch2.nrc.gov/idmws/doccontent.dll?library=PU_ADAMS^PBNTAD01&ID=071710059*

b. *NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," June 2007*

i. *This SRP chapter was primarily developed to address the technical adequacy of a baseline PRA used by a licensee to support license amendments for an operating reactor, and it is also applicable to assessing the technical adequacy of a PRA used to support a design certification or combined license application.*

ii. *http://adamswebsearch2.nrc.gov/idmws/doccontent.dll?library=PU_ADAMS^PBNTAD01&ID=071710155*

4) *NUREGs – The NRC issues other technical reports (such as NUREGs) that provide guidance for addressing regulatory or technical issues. Examples of these reports that contain guidance on conducting risk analysis are provided below.*

a. *NUREG/CR-6823, "Handbook of Parameter Estimation for Probabilistic Risk Assessment," September 2003*

i. *The data analysis portion of a nuclear power plant PRA provides estimates of the parameters used to determine the frequencies and probabilities of the various events modelled in a PRA. This report provides guidance on sources of information and methods for estimating the parameters used in PRA models and for quantifying the uncertainties in the estimates. This includes determination of both plant-specific and generic estimates for initiating event frequencies, component failure rates and unavailability's, and equipment non-recovery probabilities.*

ii. *http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6823/*

b. *NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," April 2005*

   i. *This report documents "good practices" for performing human reliability analyses (HRAs) and assessing the quality of those analyses. Good practices are those processes and individual analytical tasks and judgments that would be expected in an HRA (considering current knowledge and state-of-the-art), in order for the HRA results to sufficiently represent the anticipated operator performance as a basis for risk-informed decisions. The NRC prepared this report as part of the agency's activities to address quality issues related to probabilistic risk assessment (PRA).*

   ii. *http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1792/*

c. *NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," September 2005*

   i. *This report documents state-of-the-art methods, tools, and data for the conduct of a fire PRA for a commercial nuclear power plant application. This report is intended to serve the needs of a fire risk analysis team by providing a structured framework for conduct of the overall analysis, as well as specific recommended practices to address each key aspect of the analysis.*

   ii. *http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/*

d. *NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing," December 2007*

   i. *This report documents the results of a feasibility study for developing a risk-informed and performance-based regulatory structure for the licensing of future nuclear power plants (NPPs). As such, this report documents a "Framework" that provides an approach, scope and criteria that could be used to develop a set of requirements that could serve as an alternative to the existing requirements for licensing future nuclear power plants.*

   ii. *http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1860/*

e. *NUREG/CR-2300, "A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," January 1983*

   i. *This report is intended to provide an overview of the risk-assessment field as it existed in the early 1980's and to identify acceptable techniques for the systematic assessment of the risk from nuclear power plants. The main objective of the PRA Procedures Guide is to provide general assistance in the performance of probabilistic risk assessments for nuclear power plants.*

   ii. *http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr2300/*

5) *Voluntary Consensus Standards – The NRC's policy is to increase the involvement of stakeholders in our regulatory development process. This is consistent with the provisions of the National Technology Transfer and Advancement Act of 1995 and Office of Management and Budget (OMB) Circular A-119, "Federal Participation in the Development and Use of Voluntary Consensus Standards and in Conformity Assessment Activities." NRC staff participates in the development of consensus standards in support of the NRC's mission, and encourage industry to develop codes, standards, and guides that can be endorsed by the NRC and carried out by the industry. To that end, the NRC staff is actively engaged with the American Nuclear Society (ANS) and American Society of Mechanical Engineers (ASME) to develop consensus standards for the development and use of PRAs.*

   a. *ASME RA-S-2002, ASME RA-Sa-2003, ASME RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications,*

      i. *This Standard sets forth requirements for PRAs used to support risk-informed decisions for commercial nuclear power plants, and prescribes a method for applying these requirements for specific applications. This standard covers internal events in support of a Level 1 and limited Level 2 PRA.*

      ii. *http://catalog.asme.org/Codes/PrintBook/RAS_2002_Probabilistic_Risk.cfm*

   b. *ANSI/ANS-58.23-2007, "Fire PRA Methodology"*

      i. *This standard sets forth requirements for fire PRAs used to support risk-informed decisions for commercial nuclear power plants, and prescribes general requirements for fire PRA practice intended to suit a wide range of applications. This standard covers fires occurring within the plant.*

      ii. *http://www.ans.org/store/vi-240270*

   c. *ANSI/ANS-58.21-2007, "External-Events PRA Methodology"*

      i. *This standard sets forth requirements for analyzing accident sequences initiated by external events that might occur while a nuclear power plant is at nominal full power. It is further limited to requirements for (a) a Level 1 analysis of the core damage frequency (CDF) and (b) a limited Level 2 analysis sufficient to evaluate the large early release frequency (LERF). The scope of a seismic margin assessment (SMA) covered by this standard is limited to analyzing nuclear power plant seismic capacities according to either the so-called Electric Power Research Institute (EPRI) method ("EPRI Method") or the so-called U.S. Nuclear Regulatory Commission (NRC) method ("NRC Method").*

      ii. *http://www.ans.org/store/vi-240265*

6. To what is the criterion applicable?

   – A reactor,
   – A plant (multiple reactors sharing at least one safety system),
   – A site (several reactors on the same location),
   – A population of reactors (all the reactors in the country)?

Please explain why you chose this applicability.

The criteria are applied for each reactor, so that the safety of each individual reactor could be evaluated.

7. How is the criterion expressed?

- Single value
- Band (limit and target)

The criteria are expressed as a single value. For decision-making purposes, there are several bands of varying CDF (LERF) and ΔCDF (ΔLERF) values.

8. How is this criterion considered?

- Legally-bound limits
- Strict, but not legally-bound, limits
- Safety indicator
- Orientation values

The criteria are safety indicators and denote a boundary that, if surpassed, will often lead to increased regulatory oversight. It is only used as one piece of information in the regulatory process (risk-informed not risk-based.)

9. What is the scope of the analysis used for measuring plant performance against the criterion?

The analysis must be of sufficient scope, depth, and quality to support the decision to be made. Generally, internal and external events are considered.

10. How is uncertainty considered when measuring plant performance against the criterion? Do sensitivity analyses play any role in measuring plant performance against the criterion?

Uncertainty and sensitivity analyses are considered in risk-informed decision-making. In making an integrated decision, the decision maker must consider multiple elements such as risk information, compliance with requirements, and defence in depth. When the risk analysis element is considered, the uncertainties in the results of that analysis need to be addressed to understand the robustness of the conclusions of the analysis. Depending of their significance, the impacts of the uncertainties could influence the decision under consideration. Under the best circumstances, the impact of the uncertainties can be directly quantified, and the analyst can determine whether the planned defence-in-depth measures, the safety margins, and other factors adequately address the uncertainties and then make his or her decision accordingly. Uncertainties should be addressed quantitatively where possible, if this is not possible, an alternative approach is needed such as a bounding analysis, demonstrating the uncertain portion is irrelevant, or adopting conditions on implementation. These conditions could be adopting performance monitoring requirements, limiting the scope of application of plant changes, or establishing compensatory measures. (http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1855/)

11. When and how do Probabilistic Risk Criteria apply?

11.1 When do you require evaluation of plant performance against the Probabilistic Risk Criteria?

Plants are evaluated against the probabilistic risk criteria in various ways and for different purposes. Examples include: (1) the assessment of inspection findings for their significance can occur on a daily basis, (2) the combined assessment of performance indicators and inspection findings occurs on a quarterly basis, (3) the assessment of the need for NRC response to an event occurs immediately following the event, (4) changes to the plant licensing basis are done in response to requests from licensees.

11.2 What type of action do you engage if the Probabilistic Risk Criteria is exceeded?

For plant operating performance, a range of actions can occur including increased regulatory oversight and inspections, public meetings, demands for information, and issuance of orders up to and including the shutdown of a plant.

12. In case of band-defined goals, how is handled the case where the results are inside the band?

Dependent upon where the band is set and where the plant performance is with regard to the band, NRC's response could range from no action to increased regulatory oversight to a cessation of activities.

13. Have you defined other subsidiary criteria for PSA applications?

List other risk criteria (note that the question refers specifically to PSA applications. So which criteria are relevant to PSA applications?)

There are system level measures and importance measures that are generally used by most PRA practioners.

14. What is your experience with Probabilistic Risk Criteria?

Please provide a separate response for every experience you wish to share. These experience statements can address one or several of the following sub-criteria.

We do not expect a complete relation of your experience with probabilistic Risk Criteria, but a limited number of statements addressing the most important.

14.1 On implementation

- How did first implementation of the Probabilistic Criteria work?
- Have you identified practical benefits of setting Probabilistic Risk Criteria?

The NRC has been working with PRAs for more than 25 years. As such, implementation has been increasing over the years. The benefits of PRA are most widely acknowledge in providing better decision-making, and focusing resources on those issues of most importance to safety.

14.2 On consequences of implementation
Have Probabilistic Risk Criteria lead to system upgrades?

Part of NRC's regulatory analysis program including an assessment of the performance of plants, and assessment of safety issues. PRA is used to identify system upgrades and support changes to the regulatory structure.

14.3 On communication with the public

Most guidelines and documentation supporting the use and development of risk assessments are publicly available. In addition, the 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," January 2004 contains a comparative analysis of NRC's risk communication needs and state-of-the-art risk communication practices. The document provides the Risk Communication Guidelines and how the NRC can best incorporate risk communication principles throughout the agency. (http://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0308/br0308.pdf)

14.4 On interpretation

Examples of other areas of interest might be beneficial:

No response.

15. Your organisation has not defined Probabilistic Risk Criteria:
If deliberate, please explain the reasons
Does it expect setting Probabilistic Risk Criteria in the future?

No response.