Review of Probabilistic Acceptance Criteria and their Relation to Radiological Acceptance Criteria

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Abstract: Nuclear power plants are designed and operated to meet both deterministic radiological acceptance criteria and probabilistic acceptance criteria. The deterministic criteria are essentially used as a basis in design to develop functional requirements on capacities and environmental qualification. The safety analysis report presents deterministic analyses in principle showing that fuel is not overheated if at least one safety train - making up the fundamental safety functions reactivity control, cooling and confinement - is successful. The probabilistic analyses study the overall risk and contributors to this risk in identifying possible areas for further improvement, especially if a probabilistic criterion is not met. Even though basically having the same origin and efforts to harmonize, both deterministic and probabilistic criteria vary between countries. The renewed interest for nuclear power has emphasized the importance of harmonization. The recent published new regulations in Sweden lack radiological public acceptance criteria for new reactors. An SSM study has the aim to propose for new reactors, radiological acceptance criteria for the defined event classes with their associated frequency bands, taking harmonization issues into account. The study has compiled information on both deterministic radiological acceptance criteria and probabilistic criteria from about ten different countries. This paper will provide a background, and an overview of the radiological acceptance criteria and the probabilistic criteria used in these countries. Similarities and differences in the numbers as well as the metrics definitions and methods are identified. Examples of method issues are use of conservative versus realistic codes and assumptions and the time window used to calculate doses. Further, this paper will present some observations on the benchmarking of the respective deterministic and probabilistic criteria, and also some observations related to comparing of the deterministic vs probabilistic criteria – e.g., is the core damage frequency limit in line with the radiological acceptance criteria? A discussion of the treatment of some specific issues related to potentially new and smaller reactors compared to existing plants include siting in more densely populated areas and multi-core reactors.

Keywords: PRA, Radiological acceptance criteria, Probabilistic Acceptance Criteria, Event Classes.

1. INTRODUCTION

Nuclear power plants are designed and operated to meet both deterministic radiological acceptance criteria and probabilistic acceptance criteria. The deterministic criteria are essentially used as a basis in design to develop functional requirements on capacities and environmental qualification. The safety analysis report presents deterministic analyses with the purpose to show that fuel is not overheated if at least one safety train (100%) - making up the fundamental safety functions reactivity control, cooling and confinement - is successful, has the needed functional performance to take care of individual event scenarios. The probabilistic analyses study the overall sum of the risk (the total frequency to end up in an unwanted consequence) and the contributors to this risk in identifying possible areas for further improvement, especially if a probabilistic criterion is not met. Even though basically having the same origin and efforts to harmonize, both deterministic and probabilistic criteria vary between countries. The renewed interest for nuclear power has emphasized the importance of harmonization.

The recent published new regulations in Sweden do not include radiological public acceptance criteria for new reactors. An SSM study has the aim to propose for new reactors, radiological acceptance criteria for the defined event classes with their associated frequency bands, taking harmonization into account. The study has compiled information on both deterministic radiological acceptance criteria and probabilistic criteria from different countries to support a proposal for radiological acceptance criteria for potentially new nuclear power reactors in Sweden. The study (SSM2023-3711) [1] is still ongoing and final recommendations will be made late 2024. However, the section on acceptance criteria review is in principle completed.

The countries included in the review are Sweden, Finland, France, The Netherlands, Poland, Slovakia, Slovenia, Spain, the UK, the Czech Republic, Hungary, India, Japan, Coree, Canada and the US.

Information was collected through a survey and studies of the different countries' legislation and other documents as available on the respective regulator web sites. For probabilistic criteria, also the OECD/NEA report on use and development of PSA in member states [2] was studied.

The early Swedish analyses of environmental consequences used US references on assumptions and acceptance criteria, e.g.:

- RG 1.3 Assumptions used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors (june 1974) [3].
- RG 1.4 Assumptions used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurised Water Reactors (june 1974) [4].

The dose criteria were from GDC Part 100 Reactor Site Criteria [5] – the limit at the site boundary during two hours immediately after a release should be less than 25 rem (250 mSv) whole body or for the thyroid a dose at 300 rem (3 Sv) from Iodine.

According to [5] 25 rem numerically is in line with the life time risk for a worker at a nuclear power plant. It is pointed out that these high doses are acceptable limits, in emergency situations, for doses to the public and that they shall be considered as reference values for evaluation of a site regarding hypothetical reactor accidents with expected very low frequency of occurrence and small risk for the public to be exposed to ionizing radiation.

After the Three Mile Island (TMI) accident in 1979, Sweden introduced in 1981 requirements on filtered vented containment with a design criterion that the maximum release should be 0,1 % of the core inventory of Cs-134 and Cs-137 - core inventory in a nuclear power plant with a thermal power (1800 MW^{th}) corresponding to Barsebäcks NPPs operating at that time.

Swedish radiological acceptance criteria for the effective dose to individuals for event class H2-H4B for existing NPPs SSMFS 2021:5 [6] are the same as decided by SSM in 2009 (SSM2008-1945) [7], see Table 1 below.

Table 1. Swedish Radiological Acceptance Criteria for Existing NPPs.

Event class	Frequency band [per reactor year]	Effective dose [mSv]	Cs-137 [TBq]
Normal (H1)	1	<0.025	-
Expected (H2)	10^{-2}<=H2	<1	<0,1
Not expected (H3)	10^{-4}<=H3<10^{-2}	<10	<1
Unlikely (H4A)	10^{-6}<=H4A<10^{-4}; External: 10^{-5}<=H4A<10^{-4}	<100	<10
Special (H4B)	<10^{-4} + CCF; External:10^{-6}<=H4B<10^{-5}	<100	<10
Very unlikely (H5)	<10^{-6}	-	<100
Residual risk	No specific limit given	-	-

The first large PRA – the Reactor Safety Study – WASH-1400 [8] had an objective to do a realistic estimation of risks with nuclear power and compare with other risks that the society and individuals are exposed to. A core damage frequency at 10^{-4} was then seen as acceptable and later also introduced in IAEA publications as an acceptable limit for existing reactors. This was also a safety goal or value used in result evaluations in Swedish studies for many years. Current Swedish regulations do not prescribe any probabilistic safety goals. The requirement is that the licensees shall define and motivate criteria for evaluation of results from the probabilistic analyses.

The study collected detailed information per country. The main report compiled a comprehensive table with comparison of deterministic criteria (dose and release where applicable) per frequency bands defined to cover one order of magnitude each. This information is the basis for observations per country and per frequency band.

Similarly, information on probabilistic criteria were collected and put together and observations made.

2. DETERMINISTIC ACCEPTANCE CRITERIA

2.1. Basis for Deterministic Criteria

The deterministic radiological acceptance criteria are the basis for NPP safety. These criteria are related to both the defence-in-depth and to event classes.

The underlying assumption is that more frequent events must have a smaller consequence than less frequent events, both radiological consequences and other consequences (e.g. repair catastrophic component failures) that are important to avoid for a licensee.

This is a basis in the definition of the five layers of the Defence-in-Depth (DiD) [9].

1) Prevent disturbances (and failures)
2) Control disturbances (and detection of failures)
3) Prevent fuel overheating
4) Mitigate releases of radioactivity (on site)
5) Mitigate consequences resulting from releases of radioactivity (off site).

An elaborated view of DiD is given in Figure 1. This figure indicates the relations between the DiD levels and PSA initiating events (IEs), level 1 PSA end state Core Damage (CD) and level 2 PSA end state release categories (RCs). It also indicates that the process systems are taking care of DiD level 1 and 2 and the safety systems and emergency systems are taking care of DiD level 3-5. It also indicates that all DiD levels have both prevention and mitigation tasks. Maybe most important is that the figure shows all interfaces where it is of importance with independence between the blocks (as far as is reasonably achievable).

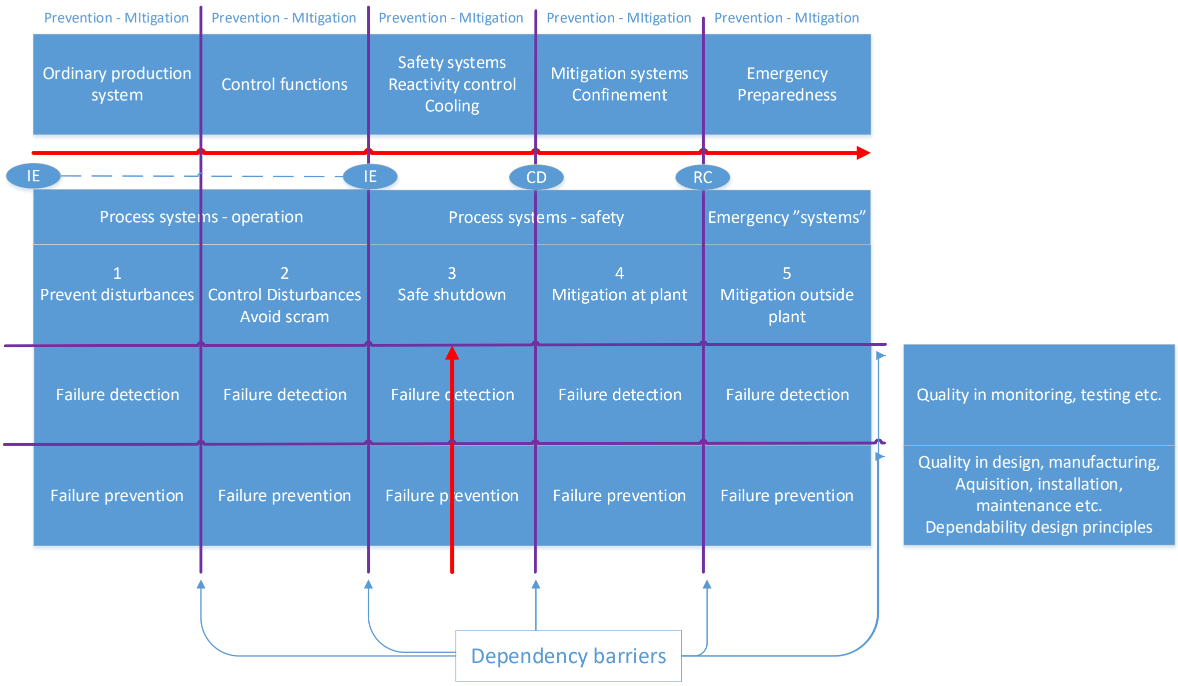

Figure 1. The Defence-in-Depth[1].

[1] IE=Initiating Event; CD=Core Damage; RC=Release category

An overview of event classes is given in Table 2 SSM2011-4329-3 [10]. The table shows how international event class definitions relates to the event classes developed for Swedish BWRs before the new regulations in 2021. The new Swedish regulations has divided event class H4 into H4A and H4B, there is no defined lower limit for H5 and there is thus no defined residual risk limit. In principle, H5 is large fuel damage where the acceptance criteria is defined (for existing NPPs) as the release of maximum 100 TBq CS-137 per event.

Table 2. Relation Between Swedish and International Event Class Definitions.

<table>
<tr><th>Frequency band per reactor year</th><th>Event class Swedish practice (before new regulations)</th><th>US RG 1.206 (based on 10 CFR)</th><th>US ANSI/ ANS 51.1/52.1</th><th>US ANS N18.2</th><th>France</th><th>Finland</th></tr>
<tr><td>$F \geq 1$</td><td>Normal operation (H1)</td><td>Normal</td><td>PC-1</td><td>Condition I</td><td>Normal operation (Condition I)</td><td>Normal operation</td></tr>
<tr><td>$10^{-1} \leq F < 1$</td><td rowspan="2">Expected events (H2)</td><td rowspan="2">Anticipated Operational Occurrences (AOO)</td><td>PC-2</td><td>Condition II</td><td rowspan="2">Incidents of moderated frequency (Condition II)</td><td rowspan="2">Anticipated operational transients</td></tr>
<tr><td>$10^{-2} \leq F < 10^{-1}$</td><td>PC-3</td><td>Condition III</td></tr>
<tr><td>$10^{-3} \leq F < 10^{-2}$</td><td rowspan="2">Not expected events (H3)</td><td rowspan="6">Postulated Accidents</td><td rowspan="2">PC-4</td><td rowspan="6">Condition IV</td><td rowspan="2">Design basis accidents (Condition III)</td><td>Class 1 postulated accidents</td></tr>
<tr><td>$10^{-4} \leq F < 10^{-3}$</td><td rowspan="2">Class 2 postulated accidents</td></tr>
<tr><td>$10^{-5} \leq F < 10^{-4}$</td><td rowspan="2">Unlikely events (H4)</td><td rowspan="4">PC-5</td><td rowspan="4">Hypothetical accidents (Condition IV)</td></tr>
<tr><td>$10^{-6} \leq F < 10^{-5}$</td><td rowspan="3">Severe accidents</td></tr>
<tr><td>$10^{-7} \leq F < 10^{-6}$</td><td>Very unlikely events (H5)</td></tr>
<tr><td>$F < 10^{-7}$</td><td>Residual risk</td></tr>
</table>

2.2 Radiological Acceptance Criteria for Different Event Classes / Frequency Bands

Event classes are defined in terms of frequency bands and related criteria to make sure that the plant in principle stays in operation (DiD 1-2) with very limited radiological consequences (that are really expected from events with a frequency of occurrence down to approximately 10^{-2} per year). Radiological acceptance criteria used in this frequency band are usually between 0.1-1 mSv per event with 0.1 mSv in most countries being the effective whole body dose limit for normal operation and for a whole site. This frequency band is also many times referred to as normal operation. Events with lower frequency are usually referred to as Anticipated Operational Occurrences (AOO). Many countries have defined dose criteria also for these events down to a frequency of about 10^{-4} per year. Events / scenarios with a frequency below 10^{-4} per year are many times referred to as accident conditions and in case of fuel damage the term used is Beyond Design Basis Accidents (BDBA).

The deterministic criteria are usually defined in a way to be the basis for the design (performance and environmental qualification) so that the plants Structures, Systems and Components (SSCs) adequately can respond to the challenges from the design basis events. They are therefore used per event / condition.

The investigation has compiled a rather detailed table with existing radiological criteria according to the regulations in the countries listed above. This table is too large to include in this paper, however, an overview of the dose criteria is given in Figure 2. A summary showing the low and high values for different frequency bands is provided in Table 3.

Table 3. Overview of Low and High Deterministic Radiological Acceptance Criteria.

Frequency band	Comment	Dose low	Dose high	Release (TBq Cs-137)
10^{-1}-1	Site criteria	0.1 mSv	1 mSv	-
10^{-2}-10^{-1}		0.1 mSv	1 mSv	< 0.1
10^{-3}-10^{-2}		1 mSv	10 mSv	< 1
10^{-4}-10^{-3}		1 mSv (new) 5 mSv	10 mSv	< 1
10^{-5}-10^{-4}		5 mSv	250 mSv	< 10
10^{-6}-10^{-5}		5 mSv	500 mSv	< 10
$5*10^{-7}$-10^{-6}		-	< 250 mSv	< 100 (core damage)
<$5*10^{-7}$	Only Finland			> 100 (LRF/LERF)

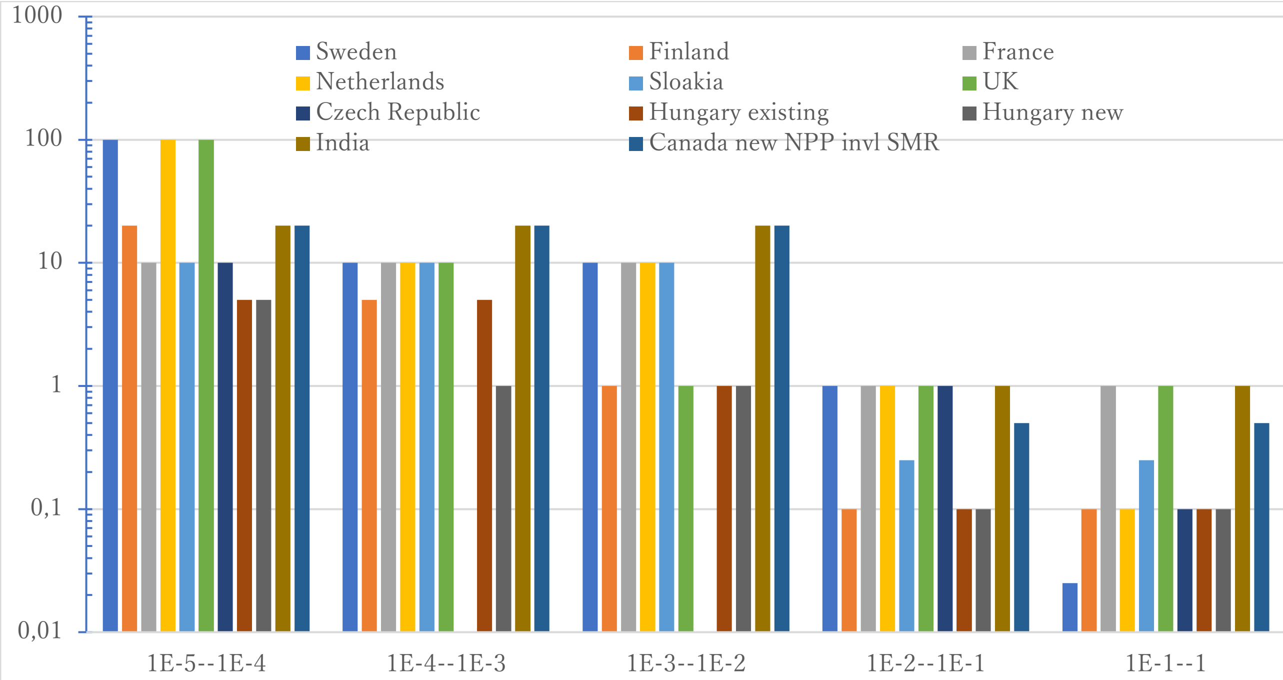

Figure 2. Overview of Dose Criteria (mSv) in Different Countries per Frequency Band.

2.3 Observation from Compilation of Deterministic acceptance criteria

Terminology and notation differ between countries. Most countries use AOO-Anticipated Operating Occurrences, DBA – Design Basis Accidents, also called ”accident conditions”, and events outside design, BDBA – Beyond Design Basis Accidents. Some countries, especially in the EU, use WENRA nomenclature with a number of DBC - Design Basis Conditions and extended conditions - DEC – Design Extension Conditions. However, there seem to be some variation in the use of DEC nomenclature. In most countries, DEC-A is a case with limited fuel damage and DEC-B is a core melt. Some countries also have DEC-B (and DEC-C) as limited core damage where the different DEC classes reflect the cause: DEC-A - CCF; DEC-B – complex sequences identified with PSA; and DEC-C – rare external threats.

Sweden is the only country with a release acceptance criterion (per event) in all frequency bands (for existing NPPs).

Several countries have the same acceptance criteria for events with a frequency from 1 per year to 10^{-2} per year. This criterion is in several cases also applied for the whole site, compared to the criteria for lower frequency events that are per reactor /facility.

Some countries have a release acceptance criterion for severe accidents, and they are given as risk criteria (PSA), i.e. it is the total frequency of all scenarios that shall be considered and the release is maximum 0,1% of the core or 100 TBq Cs-137. This is in line with the Filtered venting design criteria from the 80-ties in Sweden. However, the Swedish release criteria in event class H5 is per event/scenario.

Most of the countries have the same criteria for both existing NPPs and new NPPs. Some countries, though, have stricter criteria for new plants.

UK seem to have the most comprehensive set of criteria including dose criteria representing all scenarios. The UK use both limit and objective criteria.

Compared to Swedish criteria for existing NPPs, all countries in the comparison have lower dose criteria for event classes covering normal operation and anticipated operational occurrences.

The Swedish filtered venting containment criteria seem to have been adopted by several countries, though as a PSA requirement.

Events with very low expected frequency of occurrence have acceptance criteria with a very large variation, the US with 250 mSv, Hungary 5 mSv for new nuclear and Sweden for existing nuclear 100 mSv. However, in both Sweden and Hungary it is not expected that any large fuel damage occurs at these frequency levels.

For events with frequencies between 10^{-6} and 10^{-5}, only a few countries have defined dose limits, e.g., Sweden and the Netherlands with 100 mSv, India with 20 mSv and the US with 250 mSv. Finland instead has a PSA level 2 release criterion – The frequency of all events with max 100 TBq Cs-137 release shall be lower than 10^{-5} per reactor year. For existing Swedish NPPs, the release criterion is 100 TBq, though this is per event.

Only a few countries have a release criterion for very rare events. In Finland, all sequences with a release above 100 TBq shall have a total frequency below $5*10^{-7}$ per reactor year and this can be seen as the cutoff value. Such criterion is not given in the existing Swedish regulation.

Several countries do not explicitly state a cutoff value, but interpretation of the PSA criteria indicate a frequency cutoff criterion at 10^{-6} for existing and 10^{-7} for new builds.

The main observations are:
- Different terminology
- Differences in frequency bands.
- Differences in values
- Most countries have dose criteria for events and scenarios expected to be more frequent.
- Some countries do not have either dose or release criteria for rare events/scenarios.
- Criteria for events within design base are per event while for events beyond design, i.e., large fuel damage, the criteria transition to risk criteria.
- Some countries have a release criterion for rare events expressed in PSA terms, i.e., it is the total frequency of scenarios with a release below a defined number that shall be compared with the criteria.

The dose criteria are based on knowledge about the risk (consequences and frequency of the consequence) of exposure to ionizing radiation. Some countries also use risk criteria, especially for less frequent events. Note that the approaches for calculating doses also may vary between countries, and may be one reason for the differences in the criteria.

3. PROBABILISTIC CRITERIA

3.1 Types and Basis for Probabilistic Criteria

There are in principle four types of criteria:
- PSA level 1 criteria: Core or Fuel Damage Frequency (CDF/FDF)
- PSA level 2 criteria: Frequency of Large or Large Early Release Frequency (LRF/LERF) or Conditional Containment Damage Probability (CCDP).
- PSA level 3 criteria: Frequency of dose exceedance levels, early or late fatalities.
- System level criteria: Expected success failure probability or failure probability for different types of safety systems

Most countries requires that probabilistic safety analyses are performed as a complement to the deterministic safety analyses. The requirement is usually to perform a PSA level 1 and PSA level 2 and thus regulatory criteria are defined for PSA level 1 and 2, i.e., the core damage and release frequencies shall be below the defined criteria.

In many cases it is a strict limit but many countries also see the criteria as orientation values to evaluate results. It is also rather frequent with both limits and objective criteria (ALARA).

PSA level 1 and 2 criteria can be seen as surrogates for risk criteria, e.g., Quantitative Health Objectives (QHOs).

NUREG-1860, Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing (2007) [11] describes the high-level criteria as the additional fatality risk resulting from a nuclear power plant shall be below 0,1% of the frequency of fatalities due to all other accident causes.

According to [11], for currently operating LWRs, surrogate objectives related to core damage prevention and accident mitigation, (i.e., core damage frequency (CDF) and large early release frequency (LERF) or conditional containment failure probability (CCFP), were developed and used as surrogates for the quantitative health objectives (QHOs) expressed in the Safety Goal Policy Statement (i.e., $2*10^{-6}$ individual risk for latent fatalities and $5*10^{-7}$ individual risk for early fatalities). These LWR specific surrogate risk objectives have been used as the basis for various risk-informed activities for currently operating plants. The numerical values used for these surrogates (10^{-4} for CDF, 10^{-5} for LERF (Large Early Release Frequency) , and 0.1 for CCFP (Conditional Containment Failure Probability)) were based upon the characteristics and risk analysis associated with currently operating light-water reactor plants (e.g., plant size, performance, source term, emergency preparedness) and their site characteristics (i.e., meteorology and population distribution). In effect, for current LWRs the 10^{-4} CDF serves as a surrogate for the latent fatality QHO as well as a measure of accident prevention, and the 10^{-5} LERF or 0.1 CCFP serves as a surrogate for the early fatality QHO for currently operating reactors.

Many countries are using this number that also is referred to in IAEA INSAG-12 [9] where it is stated that CDF shall be below 10^{-4} per reactor year and the frequency for early or large release shall be below 10^{-5} per reactor year. It is expected that new light water reactors or other reactor types Gen IV are a factor of 10 safer.

Core or fuel damage is usually considered to occur if the fuel or cladding temperature is above a predefined level that is used in the PSA deterministic supporting analyses that develops the success criteria for safety functions and safety systems used along identified scenarios. It is not the purpose for this paper to discuss in detail, but these definitions also vary between different PSAs and may be subject to separate studies.

A few countries have defined risk criteria requiring a PSA level 3.

A few countries also define reliability criteria, i.e., criteria for the safety system success. Those countries that define system level criteria use 10^{-3} as the maximum system failure probability. Canada and the US also have special requirements for follow-up of these system criteria. In the US it is the Mitigating systems Performance Index (MSPI) that is part of the Reactor Oversight Program safety indicators. Canada has a specific requirement on Reporting on system dependability [12]. Sweden has no such specific requirements.

3.2 Overview of Probabilistic Criteria

Table 4. PSA Level 1 and 2 Criteria.

	PSA level 1		PSA level 2	
Country	Existing NPP	New NPP	Existing NPP	New NPP
Finland	10^{-5}	10^{-5}	$5*10^{-7}$	$5*10^{-7}$
France	10^{-4}	10^{-5}	-	-
Netherland	10^{-6}	10^{-6}	-	-
Poland	-	10^{-5}	-	10^{-6}
Slovakia	10^{-4}	10^{-5}	10^{-5}	10^{-6}
Slovenia	10^{-5} (10^{-6})[2]	10^{-5} (10^{-6})	10^{-6} (10^{-7})	10^{-6} (10^{-7})
Spain	10^{-4}	10^{-5}	10^{-6}	10^{-7}
UK	10^{-4} (10^{-5})	10^{-4} (10^{-5})	-	-
Cech Republic	10^{-4}	10^{-5}	10^{-5}	10^{-6}
Hungary	10^{-4}	10^{-5}	10^{-5} (10^{-6})	10^{-6}
India	10^{-5}	10^{-5}	10^{-7}	10^{-7}
Japan	10^{-4}	10^{-4}	10^{-6} (10^{-5})	10^{-6} (10^{-5})
Coree	10^{-4}	10^{-5}	10^{-5}	10^{-6}
Canada	10^{-5}	10^{-5}	-	-
The US	10^{-4}	-		

Sweden has no PSA criteria defined by the regulator. However, it is required that the licensees define own criteria for result evaluation.

PSA level 1 criteria are in most cases 10^{-4}/per year with a few examples with 10^{-5} per year, at least as an objective. The Netherlands has a PSA level 1 criteria at 10^{-6} as limit. Several countries defines both a limit and objective value. The majority of criteria shall be met when all contributors – operating states and events – are included.

For new NPPs, in most cases the criteria / objective is a factor of 10 lower, however, Finland, the Netherlands, India, Japan, and Canada has the same requirements for existing and new NPPs: 10^{-4} (Japan) and 10^{-6} (the Netherlands).

Some countries have separate criteria for external events and in particular seismic events.

India also has a requirement that the CDF contribution from internal events during power operation and low power and shutdown states shall be less than 10^{-6}.

RG 1.174 ”*Risk-informed Changes to the Licensing Basis*” [13] provides guidance for NRC result evaluation, e.g. to:

- Eliminate proposed changes when the risk already is acceptably low.
- As basis for decisions on actions by studying expected CDF decrease.

Table 5. NRC guidance for result evaluation [13].

Expected CDF-decrease [$year^{-1}$]	Action
> 10^{-4}	High priority to continue analysis
10^{-4} – 10^{-5}	Decision needs to be taken by responsible ”Division Director”
< 10^{-5}	Further analysis is ended unless ”Office director” decides to continue based on strong engineering or qualitative arguments.

4. DETERMINISTIC VS PROBABILISTIC

As noted above, most countries have deterministic radiological acceptance criteria for the more frequent / expected events with a return frequency above 10^{-3} per year. Some countries also have dose criteria for less

[2] Objective within parenthesis

frequent events. A number of countries do not specify radiological criteria for unlikely events and in several cases, for very unlikely events, severe accidents, when large fuel damage is expected and with potential release, the criteria used are in terms of total frequency of a release below a defined limit, i.e., PSA type criteria.

Overall, there seem to be a balance between the deterministic criteria and the probabilistic criteria. In principle, all criteria are more or less probabilistic, but in different ways. The frequency classes used for the dose and release criteria also have flavors of probabilistic properties. The main difference is the interpretation that, for developing the design and make sure that functional performance and environmental qualifications are such that the SSCs can do their job, it is important to identify the most challenging events/scenarios in each event class. Part of that work is to make sure that SSC dependability supports the reasoning for placing events/scenarios in the defined event classes. It is thus important that the appropriate dependability is in the design by the use of design principles such as single failure criteria, functional and spatial separation and diversity. The latter cannot be achieved without the use of probabilistic methods.

5. CONCLUSIONS

A review of radiological acceptance criteria for assessing results from deterministic safety analyses and safety goals (risk criteria) for assessing results from probabilistic safety analysis is performed.

Regarding deterministic criteria some results from the comparison between Sweden and other countries is listed in Table 5. Note that the approaches for calculating doses also may vary between countries, and may be one reason for the differences in the criteria.

Table 5. Comparison of deterministic criteria.

Sweden	Other countries
• Dose criteria for existing NPPs are higher than dose criteria in the countries in this investigation • Special frequency bands for external events • All criteria are per event / condition • No lower limit for residual risk • No risk criteria	• Lower dose criteria, both existing and new • Same for existing and new NPPs • No separate treatment of external events • Some countries has a defined lower limit (residual risk limit) • Transition to probabilistic criteria for low frequency events corresponding to sever accident situations

Further, most countries have the same criteria for existing and new NPPs.

Sweden does not have explicit PSA criteria, however, the 100 TBq Cs-137 criteria originating from the FILTRA design criteria for Barsebäck NPP (described in the introduction) can implicit be interpreted as a PSA criteria, and that was also the case for the licensees developing their first own criteria, 10^{-5}/ry for core damage and 10^{-7}/ry for large release (above 100 TBq). The current Swedish release criteria for 100 TBq CS-137 (in the regulation [6]) is that the frequency shall be less than 10^{-6} per event. This is a factor of 10 lower than the Finnish frequency criteria 10^{-5}, but then the Finnish criteria is a PSA criteria. It is of some interest to note that several countries seem to have adopted the FILTRA criterion, but as a PSA criterion.

Sweden has no limit for so called residual risk whereas the Finnish PSA level 2 criteria is that a release above 100 TBq shall have a frequency below $5*10^{-7}$/ry.

The study is still ongoing but will propose lower dose criteria for new NPPs compared to the existing fleet. This is reasonable given that all other countries already have lower dose criteria and is a step towards some harmonization of Swedish requirements in relation to other countries.

Other ideas (ongoing work) are to further study if some more changes can be of value, e.g., introducing a frequency limit for residual risk.

Acknowledgements

The author acknowledges the co-operation with SSM colleagues during the study.

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