Safety Goals and Risk Metrics in Probabilistic Safety Assessment: An International Overview from the OECD NEA CSNI WGRISK

Joshua Gordon[a], Vinh N. Dang[b], and Marina Roewekamp[c]

[a] Office for Nuclear Regulation, London, United Kingdom, *joshua.gordon@onr.gov.uk*

[b] Paul Scherrer Institute (PSI), Villigen, Switzerland, *vinh.dang@psi.ch*

[c] Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Cologne, Germany, *Marina.Roewekamp@grs.de*

Abstract: The OECD Nuclear Energy Agency (NEA) Working Group on Risk Assessment (WGRISK) periodically publishes a status report on the use and development of Probabilistic Safety Assessment (PSA[1]) in its member countries. The most recent update, approved by the Committee on the Safety of Nuclear Installations (CSNI) and expected to be published in 2026, provides a snapshot of international PSA practices drawing on contributions from more than twenty regulatory authorities and technical safety organisations.

This paper presents an overview of the report with a particular focus on the treatment of safety goals, numerical safety criteria and risk metrics adopted in different national frameworks. The report demonstrates that PSA plays an integral role in safety assessment, licensing and regulatory decision-making across a wide range of countries. While some regulators specify explicit numerical criteria for core and/or fuel damage frequency, large or large early radioactive release frequency, or individual and societal risk, others employ qualitative goals supported by numerical targets used as decision aids rather than regulatory limits.

The paper highlights key similarities and differences in the definition and application of risk metrics for PSA Levels 1, 2 and 3. Recent developments in emerging challenges associated with extending safety goals beyond single unit assessments, including risk aggregation for multi-unit and multi-source sites, treatment of spent fuel pools and the increasing use of consequence-based metrics are identified and discussed.

The findings illustrate a general trend towards broader PSA scope, increased use of international guidance, and greater reliance on risk informed decision making, while also identifying areas where different countries adopt different approaches.

1. INTRODUCTION

Probabilistic Safety Assessment (PSA) is widely recognized as an essential component of nuclear safety analysis. It provides a structured quantitative approach to evaluating the risks of potential accidents in nuclear installations and complements traditional deterministic safety assessment methods. It is most commonly applied to complex nuclear facilities such as nuclear power plants.

The OECD Nuclear Energy Agency (NEA) Working Group on Risk Assessment (WGRISK) has played a central role in coordinating international efforts to advance PSA methodology and its use in improving nuclear safety. Operating under the NEA's Committee on the Safety of Nuclear Installations (CSNI), WGRISK brings together experts from regulatory authorities, technical support organisations, research institutes, and industry across NEA member countries and participating non-member countries and economies to share knowledge and develop best practices in risk assessment for nuclear facilities.

One of WGRISK's key activities is the periodic publication of a status report on the "Use and Development of Probabilistic Safety Assessment" in member countries. These reports, updated roughly

[1] Note that PSA and PRA (*P*robabilistic *R*isk *A*ssessment) are used in this context interchangeably.

every five years, provide a snapshot of PSA programs and practices worldwide. The reports cover PSA frameworks, methodologies, applications, results, safety goals, risk metrics, and new research and developments. The latest update [1] is based on contributions from more than twenty national organisations who attend and contribute to WGRISK. Detailed surveys were conducted, with the results analyzed and reviewed by a core team from WGRISK, summarized within the report. The report has been approved by the CSNI and is expected to be published in 2026.

The actual status report reflects the evolving landscape of PSA, with an emphasis on new developments and the growing applications of PSA results for safety management and risk-informed decision-making. This paper presents an overview of the status of PSA in participating NEA member and non-member countries and economies, with a particular focus on safety goals and risk metrics.

Some national regulatory frameworks specify explicit numerical criteria (quantitative safety goals) for key risk metrics such as core and/or fuel damage frequency (CDF/FDF), large or large early release frequency (LERF/LRF), and risk to individuals or society. In some cases these are formal limits or design targets. Other frameworks rely primarily on qualitative safety goals but may use numerical risk targets as advisory guidelines or decision-support tools rather than as limits. The differences often reflect the historical development of nuclear regulatory philosophies and the ways in which PSA has been adopted and integrated.

In the following sections, insights from the WGRISK status report are summarized to compare safety goal frameworks and risk metrics across several countries and regions. Key similarities and differences in the definition and application of safety goals at PSA Levels 1, 2 and 3 are highlighted and examples to illustrate how regulatory bodies set or use numerical risk criteria are provided. Emerging cross-cutting issues identified in the report such as multi-unit and multi-source risk aggregation, the treatment of spent fuel pool risks, the increasing use of consequence-based metrics, and the broader trend toward expanding PSA scope and risk-informed decision-making are also discussed.

2. PSA APPLICATIONS AND SAFETY GOAL FRAMEWORKS

The role of PSA in national nuclear safety frameworks has grown steadily over the past few decades. PSA is meanwhile routinely used to inform safety decisions, regulatory evaluations and plant operational practices. Many regulatory regimes require operating nuclear power plants to maintain PSA models as living tools that are updated with plant changes or new data on a regular basis.

Because PSA is inherently quantitative, it provides a means to compare plant risk estimates against safety goals through use of risk metrics. The risk metrics commonly correspond to the various PSA levels:

- At PSA Level 1, the typical risk metric is Core Damage Frequency (CDF) (or in some cases Fuel Damage Frequency (FDF), which extends to fuel damage in spent fuel pools as well as the reactor core). CDF represents the probability per reactor-year of an accident that results in core damage.
- At PSA Level 2, risk metrics gauge the frequency of significant radioactive releases from containment. The most common are Large Release Frequency (LRF) or Large Early Release Frequency (LERF).
- At PSA Level 3, the focus shifts to the consequences of radiological releases and risk is quantified in terms of impacts on the public or environment. Common measures include individual risk (risk of death or serious health effects to a person in the vicinity of a plant) and societal risk (risk from accidents causing a high number of casualties or having a wide-spread impact).

Level 1 and 2 PSAs are common for most operating power reactors in the countries of the WGRISK participants, to support risk-informed decision-making and demonstrate safety margins. Level 3 PSA, which involves modelling of off-site consequences, is less common but is carried out in countries where it aligns with regulatory goals or specific projects to quantify risk to the public. Examples include the United Kingdom, the Netherlands, and the Republic of Korea. Interest in Level 3 PSA metrics and applications is increasing as a way to address site level risk and to consider advanced technologies.

3. INTERNATIONAL SAFETY GOALS AND NUMERICAL RISK CRITERIA

The WGRISK report finds that countries differ significantly in how they define safety goals and use risk metrics in a regulatory context. Broadly, two patterns emerge:

- **Explicit quantitative criteria:** Many countries set one or more numerical risk criteria that serve as formal safety goals or limits for plant risk. These criteria usually specify maximum allowable frequencies (or target frequencies) for events such as core damage or large release. They may be enshrined in regulatory requirements, guidance documents, or even law.
- **Qualitative goals with numeric guidelines**: Other regulatory frameworks emphasize qualitative safety goals (e.g. maintaining defense-in-depth, ensuring no significant off-site consequences, reducing risk so far as is reasonably practicable) and use numeric risk estimates and targets as supporting guidelines or decision-making criteria.

Many national frameworks combine elements of both approaches. Several countries differentiate between new and operating plants with the criteria applied for new designs being more stringent (lower allowable frequencies), reflecting modern safety expectations.

Table 1, Table 2 and Table 3 contain the results of the survey conducted to identify the safety goals and risk metrics used in NEA member states, organized into Level 1, Level 2 and Level 3 PSA. Some responses indicated safety goals or metrics associated with system or plant performance or related to incremental changes in risk; these are reflected in Table 4 and Table 5.

Numerical risk criteria are not defined in all WGRISK member countries. Countries without numerical criteria specified by the regulators are Belgium, Germany, Spain and Sweden. Some countries (e.g. the United Kingdom) have goal setting frameworks (e.g. to reduce risks so far as is reasonably practicable) with numerical risk targets established by the regulator to aid decision-making. Other countries use the numerical criteria as an indicative figure (e.g., Czech Republic, France, India). A few countries have specified safety criteria to be applied only for new builds (e.g., Canada and Slovenia). In some countries, criteria have been defined for existing plants as well as for new builds (e.g., Hungary, the Netherlands and Slovenia).

The use of a Level 1 PSA metric CDF is common, with 1×10^{-5} per reactor-year for core damage being a commonly adopted benchmark used in many jurisdictions, but with significant variation seen in the standard expected for new and existing designs.

Over time, attention has broadened to Level 2 PSA metrics (LRF/LERF) as well, recognizing that preventing large releases is a distinct safety objective beyond preventing core damage. The notion of “large” or “early” release is not always uniformly defined. Some countries define it by a specific radiological quantity (e.g. 100 TBq of Cs-137), while others define it qualitatively (such as “requiring long-term relocation of the public” or “requiring off-site emergency measures”).

Level 3 PSA metrics extend to the end consequences of accidents, such as the risk of early or latent fatalities in the exposed population or radiation dose-based criteria. These are less common, however have been the topic of significant discussion in recent years as a way to address multi-unit risk, new technologies and adopt a technology-neutral approach. For example, since the production of the WGRISK report, the U.S. NRC has adopted a technology-neutral licensing approach which uses Level 3 PSA metrics.

Table 1: Level 1 PSA numerical criteria, from [1]

Country / Economy	Organisation	Risk Metric	Frequency	Remarks
Canada	Regulator	CDF	1.0 E-05 /ry	Objective for new plants
	Licensee	CDF	1.0 E-04 /ry	Limit for existing plants
		CDF	1.0 E-05 /ry	Objective for existing plants
Peoples Republic of China	Regulator	CDF	1.0 E-04 /ry	Objective for existing plants
			1.0 E-05 /ry	Objective for new plants
Czech Republic	Licensee	CDF	1.0 E-04 /ry	Objective for existing plants, full scope PSA
		CDF	1.0 E-05 /ry	Objective for new plants, full scope PSA
Finland	Regulator	CDF	1.0 E-05 /ry	Objective for existing plants
		CDF		Limit for new plants
France	Regulator	CDF	1.0 E-05 /ry	Limit for new plants (all types of failures (human, technical) and hazards (excluding malicious acts))
		CDF	1.0 E-06 /ry	Objective for new plants (internal initiating events excluding hazards, respectively for power and shutdown states)
		FDF	Practical elimination	All reactors SFPs
		Qualitative	Qualitative	Long-term operation of existing plants: implement safety improvements to reduce the gap with the new plant safety objectives
Hungary	Regulator	CDF	1.0 E-04 /ry	Limit for existing plants
		CDF	1.0 E-05 /ry	Limit for new plants
India	Regulator	CDF	1 E-06 /ry	Limit (internal events, power and shutdown states)
		CDF	1 E-06 /ry	Limit (internal events and external hazards)
Italy	Regulator	CDF	1.0 E-05 /ry to 1.0 E-06 /ry	Objective
Japan	Regulator	CDF	1.0 E-04 /ry	Objective
Republic of Korea	Regulator	CDF	1.0 E-04 /ry	Objective for existing plants
		CDF	1.0 E-05 /ry	Objective for new plants
Mexico	Regulator	CDF	1.0 E-05 /ry	Limit for existing plants
Netherlands	Regulator	CDF	1.0 E-06 /ry	Limit for existing plants
		CDF	1.0 E-06 /ry	Limit for new plants
Poland	Regulator	CDF	1.0 E-05 /ry	Objective for new plants
Romania	Regulator	CDF	1.0 E-04 /ry	Objective for existing plants
		CDF	1.0 E-05 /ry	Objective for new plants
Slovakia	Regulator	CDF	1.0 E-04 /ry	Objective for existing plants
Slovenia	Regulator	CDF	1.0 E-04 /ry	Objective for existing plants
		CDF	1.0 E-05/ry	Objective for new plants
Sweden	Licensee	CDF	1.0 E-05 /ry - 1.0 E-04 /ry	Objective: Licensees are required to define and motivate criteria for result evaluation. These criteria vary between licensees and can be seen as objectives.
Switzerland	Law	CDF	1.0 E-04 /ry	Limit for existing plants
		CDF	1.0 E-05 /ry	Objective for existing plants
	Regulator	FDF	1.0 E-05 /ry	Objective for existing plants
Chinese Taipei	Licensee	CDF	1.0 E-04 /ry	Limit for existing plants
United States of America	Regulator	CDF	1.0 E-04 /ry	Objective

Table 2: Level 2 PSA numerical criteria, from [1]

Country / Economy	Organisation	Risk Metric	Frequency	Remarks
Canada[2]	Regulator	Small release frequency(SRF) > 10^{15} Bq I-131 (or requires temporary evacuation of the local population)	< 1.0 E-05 /ry	Objective for new plants
		LRF 100 TBq Cs-137 (or requires temporary evacuation of the local population)	1.0 E-06 /ry	Objective for new plants
	Licensee	SRF	1.0 E-05 /ry	Limit for existing plants
		LRF	1.0 E-06 /ry	Objective for existing plants
Peoples Republic of China	Regulator	LRF	1.0 E-05 /ry	Objective for existing plants
			1.0 E-06 /ry	Objective for new plants
Czech Republic	Licensee	LERF > 1 % Cs-137 (up to 10 h from fuel damage)	1.0 E-05/ry	Objective for existing plants, full scope PSA, including risk from SFP
			Practically eliminated	Objective for new plants, full scope PSA, including risk from SFP
Finland	Regulator	100 TBq Cs-137 (large release)	5.0 E-07 /ry	Objective for existing plants (required since 20.12.1996)
				Limit for new plants
		Accident sequences, in which the containment function fails or is lost in the early phase of a severe accident	Small contribution to CDF	Objective for older plants (required since 15.11.2013).
				Limit for new plants
				Early releases requiring measures to protect the population shall be practically eliminated (typically smaller than large release) Early is approx. 4 – 5 h from the initiating event
France	Regulator	Unacceptable consequences (not defined)	Negligible[3]	Objective for new plants
Hungary	Regulator	LRF or LERF; meaning large or early defined qualitatively, not by radiation level or quantity of released fission products or radioactivity	1.0 E-05 /ry	Limit for existing plants
			1.0 E-06 /ry	Limit for new plants

[2] Note:
Canada defines small release frequency and large release frequency as follow:
- **Small release frequency:** The sum of frequencies of all event sequences that can lead to any release to the environment that requires temporary evacuation of the local population or a release to the environment of more than 10^{15} Bq I-131 shall be less than E-05 /ry.
- **Large release frequency**: The sum of frequencies of all event sequences that can lead to any release to the environment that requires long-term relocation of the population or a release to the environment of more than 10^{14} Bq Cs-137 shall be less than E-06 /ry.

[3] The aim is that the sequences that lead to a large early release should be "*practically eliminated*".

Country / Economy	Organisation	Risk Metric	Frequency	Remarks
India	Regulator	LERF	1.0 E-07 /ry	Limit (for internal events at full power)
Japan	Regulator	Containment failure	1.0 E-05 /ry	Objective
		100 TBq Cs-137	1.0 E-06 /ry	Objective
Republic of Korea	Regulator	LERF	1.0 E-05 /ry	Objective for existing plants
		LERF	1.0 E-06 /ry	Objective for new plants
		100 TBq Cs-137	1.0 E-06 /ry	Objective for all plants
Mexico	Regulator	LERF	1.0 E-06 /ry	Objective for existing plants
Netherlands	Regulator	LERF	Practically eliminated	Limit for new plants
Poland	Regulator	LRF, LERF	1.0 E-06 /ry	Objective for new plants
Romania	Regulator	LRF	1.0 E-05 /ry	Objective for existing plants
		LRF	1.0 E-06 /ry	Objective for new plants
		LRF 1000 TBq I-131	1.0 E-05 /ry	Objective for all plants
		LRF 100 TBq Cs-137	1.0 E-06 /ry	Objective for all plants
Slovakia	Regulator	LERF	1.0 E-05 /ry	Objective for existing plants
Slovenia	Regulator	Not defined	5.0 E-06 /ry	Limit for existing plants
		Not defined	1.0 E-06 /ry	Limit for the new builds
Sweden	Licensee	> 0.1 % of core inventory	1.0 E-07 /ry	Objective: This is a criterion or safety goal established by the licensees, for L(E)RF from Level 2 PSA.
Switzerland	Regulator	LERF	1.0 E-06 /ry	Objective for existing plants
Chinese Taipei	Licensee	Not defined	1.0 E-05 /ry	Limit for existing plants
United States of America	Regulator	LERF	1.0 E-05 /ry	Objective

Table 3: Level 3 PSA numerical criteria, from [1]

<table>
<tr><th>Country / Economy</th><th>Organisation</th><th>Risk Metric</th><th colspan="3">Frequency</th><th>Remarks</th></tr>
<tr><td rowspan="8">Netherlands</td><td rowspan="8">Regulator</td><td>Individual risk</td><td colspan="3">1 E-06 /yr</td><td>Risk of death off-site, also considering stochastic effects</td></tr>
<tr><td>Group risk, risk of death of n persons</td><td colspan="3">1 E-03/n2 /yr, n > 10</td><td>Risk of death off-site, only considering deterministic effects</td></tr>
<tr><td rowspan="6">Dose (off-site)</td><td>Frequency [1/yr]</td><td colspan="2">Dose [mSv]</td><td rowspan="6"></td></tr>
<tr><td></td><td>>16 yrs</td><td><16 yrs</td></tr>
<tr><td>F ≥ 1 E-01</td><td>0.1</td><td>0.04</td></tr>
<tr><td>1 E-01 > F ≥ 1 E-02</td><td>1</td><td>0.4</td></tr>
<tr><td>1 E-02 > F ≥ 1 E-04</td><td>10</td><td>4</td></tr>
<tr><td>F < 1 E-04</td><td>100</td><td>40</td></tr>
<tr><td>Republic of Korea</td><td>Regulator</td><td>Early fatality
Latent fatality</td><td colspan="3">5.0 E-07 /yr
1.0 E-06 /yr</td><td>Calculated for each individual unit</td></tr>
</table>

<table>
<tr><th>Country / Economy</th><th>Organisation</th><th>Risk Metric</th><th colspan="3">Frequency</th><th>Remarks</th></tr>
<tr><td rowspan="8">United Kingdom[4]</td><td rowspan="8">Regulator</td><td>Risk of death (persons off-site)</td><td colspan="3">BSL = 1 E-04 /yr
BSO = 1 E-06 /yr</td><td>Calculated for the site</td></tr>
<tr><td rowspan="6">Dose (off-site) from accidents</td><td>Dose [Sv]</td><td>BSL [1/yr]</td><td>BSO [1/yr]</td><td rowspan="6">Calculated for each individual facility</td></tr>
<tr><td>0.1 1</td><td>1</td><td>1 E-02</td></tr>
<tr><td>1 - 10</td><td>1 E-01</td><td>1 E-03</td></tr>
<tr><td>10 - 100</td><td>1 E-02</td><td>1 E-04</td></tr>
<tr><td>100 -1000</td><td>1 E-03</td><td>1 E-04</td></tr>
<tr><td>> 1000</td><td>1 E-04</td><td>1 E-06</td></tr>
<tr><td>Risk of 100 fatalities (societal risk)</td><td colspan="3">BSL = 1 E-05 /yr
BSO = 1 E-07 /yr</td><td>Calculated for the site</td></tr>
<tr><td>United States of America</td><td>Regulator</td><td>Offsite individual fatality risk</td><td colspan="3">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 1 % (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 percent) of the sum of cancer fatality risks resulting from all other causes.</td><td>These QHOs are generally quantified as follows:
- Prompt fatality risk to an individual within one mile of the exclusion area boundary should not exceed 5 E-07 /yr
- Cancer fatality risk to an individual within ten miles of the site should not exceed 2 E-06 /yr
QHOs are typically used as a technical basis for surrogate measures such as CDF and LERF used in risk-informed decision-making (RIDM) activities.</td></tr>
</table>

Table 4: Safety systems reliability criteria, from [1]

Country / Economy	Safety Systems Reliability Criteria
Canada	Safety systems and their support systems shall be designed to ensure that the probability of failure per demand from all causes is lower than 1.0 E-03. Probability of failure to shut down on demand < 1 E-05 Large Release Frequency (LRF) due to failure of shutdown < 1 E-07 /ry
India	Failure probability of the safety system < 1.0 E-03 (SDS-1, SDS-2, ECCS, Containment Isolation)
Romania	Safety systems shall be designed to ensure that the probability of failure per demand from all causes is lower than 1.0 E-03.

[4] In the United Kingdom the numerical targets are expressed in terms of the Basic Safety Level (BSL) and Basic Safety Objective (BSO). The BSLs and BSOs are used to inform regulatory decisions. New facilities should meet at least the BSLs. Regulatory resources will generally not be used to seek further improvements below the BSOs.

Table 5: Incremental numerical risk increase criteria, from [1]

Country / Economy	Numerical Criteria
Czech Republic	Change allowed if: - ΔCDF (ΔFDF) < 5 E-06 /ry - ΔLERF < 1 E-06 /ry Change not allowed if: - Overall FDF > 1 E-04 /ry - Overall LERF > 1 E-05 /ry FDF = CDF + frequency of fuel damage in the SFP - Note: Numerical criteria for incremental risk increase have not been set specifically for new plants.
Hungary	No risk increase is allowed due to plant changes
Republic of Korea	Changes are not allowed if: - ΔCDF > 1 E-05 /ry - ΔLERF > 1 E-06 /ry Changes are allowed if: - ΔCDF < baseline CDF (for baseline CDF < 1 E06 /ry) - ΔCDF < 1 E-06/ry (for baseline CDF < 1 E-04 /ry) - ΔLERF < baseline LERF (for baseline LERF < 1 E-07 /ry) - ΔLERF < 1 E-07 /ry (for baseline LERF < 1 E-05/ry) Area with detailed assessment: Between allowable area and not allowable area
Mexico	Same criteria as in U.S. NRC Regulatory Guide 1.174
Netherlands	Instantaneous TCDF shall never exceed the value of 1 E-04 /ry
Romania	Similar criteria as in U.S. NRC Regulatory Guide 1.174. Change not allowed if: - Overall CDF > 1 E-04/ry - Overall LERF > 1 E-05/ry
Slovakia	Changes are considered non-risk significant if: - ΔCDF < 1 E-04 /ry - ΔLERF > 1 E-05 /ry and their cumulative effects do not cause the safety goals to be exceeded. Changes are considered unacceptable if: - ΔCDF > 1 E-04 /ry - ΔLERF > 1 E-05 /ry
Slovenia	Limits for risk increase: - ΔCDF < E-07 /ry - ΔLERF < E-08 /ry
Spain	Criteria provided in GS 1.14 “Basic criteria for carrying out PSA Applications”, similar to U.S. NRC Regulatory Guide 1.174
Switzerland	Changes are allowed if: - ΔCDF < 1 E-07 /ry - ΔFDF < 1 E-07 /ry - ΔLERF < 1 E-08 /ry and: - Average CDF < 1 E-05 /ry - Additional requirements for the extension of test intervals apply
Chinese Taipei	Same criteria as in U.S. NRC Regulatory Guide 1.174
United States of America	Criteria provided in U.S. NRC Regulatory Guide 1.174

4. EMERGING ISSUES AND TRENDS

The WGRISK report identifies several emerging and cross-cutting challenges related to safety goals and risk metrics, reflecting the evolution of nuclear power and PSA practice in recent years. A major theme is the expansion of PSA scope beyond a single reactor unit at full power. Traditionally, safety goals and risk metrics have been framed per reactor unit, largely because PSA has been performed separately for each individual reactor. However, the Fukushima Dai-ichi accidents in 2011 highlighted the importance of considering site-level risks and accidents that might involve multiple reactors or other co-located sources (like spent fuel pools) simultaneously. This has raised interest in multi-unit PSA and raised questions about how to define safety goals for a whole site, such as through Level 3 PSA metrics.

The overall scope of PSAs and their use in decision-making continue to broaden, which influences how safety goals are applied. PSAs now more frequently address multiple plant operational modes (including low-power and shutdown conditions), a full spectrum of internal and external hazards (from internal fires and floods to earthquakes, high winds and external flooding), and combinations of hazards, reflecting learning from events like Fukushima. PSA techniques are also being applied to emerging reactor types such as small modular reactors (SMRs), which present new considerations for risk metrics (e.g., multi-module risk and the interplay of multiple small reactors on one site).

With more comprehensive PSAs, risk-informed decision-making (RIDM) in regulatory and operational contexts is growing. PSA results are increasingly used to support decisions on plant modifications, optimize maintenance and inspection intervals, prioritize safety backfits, and evaluate design alternatives. The expanding influence of PSA strengthens the relevance of safety goals: decision-makers often refer to the numerical risk metrics (CDF, LERF, etc.) to judge the significance of changes.

5. CONCLUSIONS

The most recent version of the WGRISK "Use and Development of Probabilistic Safety Assessment" report confirms that safety goals and risk metrics are central to the interpretation and use of PSA results, but their application remains diverse. A core set of metrics, core damage frequency (CDF) at Level 1, large or large early release frequency (LRF/LERF) at Level 2, and individual or societal risk measures at Level 3, are widely used as benchmarks for assessing nuclear safety. However, the role of these metrics varies from formal regulatory requirements in some countries to advisory targets supporting qualitative safety goals in others.

Despite this diversity, there is strong convergence in underlying practice with many countries applying numerical criteria broadly consistent with international guidance, typically in the order of E-05 per year for CDF and E-06 per year for large releases.

The WGRISK status report also highlights ongoing evolution in both safety goals and risk metrics, including increasing attention to Level 3 consequence-based measures, the need for site-level and multi-unit risk aggregation, and the extension of metrics beyond reactor cores to include spent fuel pools and other sources of risk. These developments indicate a shift towards more comprehensive and integrated representations of nuclear risk.

Acknowledgements

The members of the Nuclear Energy Agency (NEA), the Committee on the Safety of Nuclear Installations (CSNI) and the Working Group on Risk Assessment (WGRISK) wish to acknowledge the significant contributions of those individuals who had a key role in the preparation of the "Use and Development of Probabilistic Safety Assessment" report. The invaluable contributions from the WGRISK members and their associated organisations and individuals who provided input to this task are also acknowledged and highly appreciated. Additional thanks are expressed to Yuji Kumagai, Technical Secretary of the WGRISK, in preparing the report.

References

[1] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI). Working Group on Risk Assessment (WGRISK). "*Use and Development of Probabilistic Safety Assessment: An Overview of the Status as of 2024*", NEA/CSNI/R(2025)15, www.oecd-nea.org/nsd/docs/indexcsni.html, OECD NEA, Paris (in publication 2026).