DRAFT
Regulatory Document

RD-152
Guidance on the Use of Deterministic and Probabilistic Criteria in Decision-making for Class I Nuclear Facilities

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CNSC REGULATORY DOCUMENTS

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PREFACE

A safety assessment is a systematic process to verify that applicable safety requirements are met in all phases of the life cycle of a Class I nuclear facility. Safety analysis is a key component of a safety assessment. Safety analysis incorporates both probabilistic and deterministic approaches, which complement each other.

Deterministic safety analysis is the principal means of demonstrating that the dose acceptance criteria and safety goals are met with a high degree of confidence for all accidents within the design basis.

Probabilistic safety analysis is the principal means of demonstrating that the safety goals are met for potential accidents both within and beyond the design basis. It identifies vulnerabilities not necessarily accessible through deterministic safety analysis alone.

With the development of probabilistic analysis techniques, Class I nuclear facilities licensees and applicants have introduced probabilistic arguments in support of applications for a licence to operate as well as submissions aimed at obtaining approval for facility modifications, for closure of action items, or for temporary licence exemptions.

This document informs Class I nuclear facilities licensees and applicants of the criteria CNSC uses to assess submissions that use both deterministic and probabilistic arguments.

Nothing contained in this document is to be construed as relieving any applicant from the requirements of any pertinent regulations. It is the applicant’s responsibility to identify and comply with any applicable legislation or licence conditions.
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Guidance on the Use of Deterministic and Probabilistic Criteria in Decision-making for Class I Nuclear Facilities

1.0 INTRODUCTION

With the development of probabilistic analysis techniques, Class I nuclear facilities licensees and applicants have introduced probabilistic arguments in support of applications for a licence as well as submissions aimed at obtaining approval for facility modifications, for closure of action items, or for temporary licence exemptions.

Probabilistic analysis can complement the deterministic analysis in any of the following areas:

1. Supporting applications for licensing new or existing Class I nuclear facilities;
2. Supporting submissions for modifications of the facility configuration, maintenance or operating procedures;
3. Framing the decisions to be made following reportable events as required by the Canadian Nuclear Safety Commission (CNSC) regulatory standard S-99 Reporting Requirements for Operating Nuclear Power Plants; and
4. Supporting submissions proposing revisions to deterministic criteria.

The integration of probabilistic and deterministic safety analyses and the degree of use of probabilistic criteria is considered on a case-by-case basis.

1.1 Purpose

This document informs Class I nuclear facilities licensees and applicants of the criteria CNSC uses to assess submissions that use both deterministic and probabilistic arguments.

1.2 Scope

This document provides guidance regarding CNSC staff’s approach to decision-making when deterministic and probabilistic arguments are presented. This document may also be of interest to other licensed facilities.

1.3 Relevant Regulations

Safety analysis reports form the basis for licensees and applicants’ submissions for changes or modifications to equipment or procedures.

The following regulations are relevant to this document:

Paragraph 3(1)(i), of the General Nuclear Safety and Control Regulations provides that an application for a licence shall contain, in addition to other information,
“…a description and the results of any test, analysis or calculation performed to substantiate the information included in the application”;

Paragraph 5(f) of the Class I Nuclear Facilities Regulations provides that an application for a licence to construct a Class I nuclear facility shall contain, in addition to other information, information on “a preliminary safety analysis report demonstrating the adequacy of the design of the nuclear facility”;

Paragraph 5(i) of the Class I Nuclear Facilities Regulations provides that an application for a licence to construct a Class I nuclear facility shall contain, in addition to other information, information on “the effects on the environment and the health and safety of persons that may result from the construction, operation and decommissioning of the nuclear facility…”;

Paragraph 6(c) of the Class I Nuclear Facilities Regulations provides that an application for a licence to operate a Class I nuclear facility shall contain, in addition to other information, information on “a final safety analysis report demonstrating the adequacy of the design of the nuclear facility”.

2.0 SAFETY ANALYSIS

2.1 General Information

Safety analysis involves deterministic and probabilistic analyses in support of the siting, design, commissioning, operation, or decommissioning of a Class I nuclear facility.

In accordance with subsection 24(4) of the Nuclear Safety Control Act, the applicant is required to demonstrate that they are qualified to carry on the activity authorized by the licence and that in carrying on that activity that they will make adequate provision for the protection of the environment, health and safety of persons.

For new nuclear power plants the technical safety objective of RD-337, Design of New Nuclear Power Plants provides expectations related to the requirement for licensees and applicants to make adequate provision for the protection of the environment and health and safety of persons. The technical safety objective is the basis for the following criteria and goals:

1. Dose acceptance criteria for events within the design basis; and

2. Safety goals for beyond design basis accidents.

Similar criteria and goals may be established for other types of Class I nuclear facilities.

Safety analyses are performed to confirm that these fundamental criteria and goals are met, to demonstrate effectiveness of measures for preventing accidents, and mitigating radiological consequences of accidents if they do occur.
2.2 Deterministic Safety Analysis

Deterministic safety analysis focuses on evaluating the consequences of various events to confirm that the dose acceptance criteria are met. This includes demonstrating the efficiency of defence in depth and the integrity of protective barriers. In the case of nuclear power plants, RD-310, Safety Analysis for Nuclear Power Plants, sets out the requirements related to safety analysis, including the selection of events to be analyzed, acceptance criteria, safety analysis methods, and safety analysis documentation and review.

The objectives of deterministic analysis are given in RD-310 as:

1. Confirm that the design of the facility meets design and safety requirements;
2. Derive or confirm operational limits and conditions that are consistent with the design and safety requirements for the facility;
3. Assist in establishing and validating accident management procedures and guidelines; and
4. Assist in demonstrating that safety goals, which may be established to limit the risks posed by the facility, are met.

The licensee is responsible for identifying and classifying a set of events that covers all credible accident sequences for the facility. Identification of these events uses a variety of methods, including probabilistic analysis.

These events are analysed to demonstrate that the appropriate regulatory limits are met. For new reactor designs, the high level expectations (the dose acceptance criteria) are given in RD-337, Design of New Nuclear Power Plants, and are reproduced in Appendix A.

To simplify the analysis, the design or operating organization selects more restrictive limits, often referred to as derived acceptance criteria or decoupling criteria. Meeting these criteria is a sufficient condition to demonstrate meeting the dose acceptance criteria. For example, showing that there is no damage to any of the barriers to fission product release is sufficient to show that there will be no offsite dose, so the dose acceptance criteria are met. Failure to meet the derived acceptance criteria does not necessarily mean that dose acceptance criteria will be exceeded, but additional justification is necessary.

As described in RD-310, Safety Analysis for Nuclear Power Plants, deterministic analysis of Anticipated Operational Occurrences (AOOs) and Design Basis Accidents (DBAs) is performed in a conservative manner. For Beyond Design Basis Accidents (BDBAs), less conservative assumptions may be used.

2.3 Probabilistic Safety Analysis

Probabilistic safety analysis focuses on evaluating the risk arising from various events to confirm that safety goals are met. In the case of nuclear power plants, the requirements
for probabilistic safety analysis are provided in regulatory standard S-294, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants*.

The objectives of the probabilistic safety analysis are to:

1. Evaluate the frequencies of severe accidents to the core (for nuclear reactors) and of severe releases, and compare them to the safety goals;
2. Evaluate the effect of facility or operational modifications on the frequency of accidents;
3. Evaluate the effectiveness of defence in depth; and
4. Evaluate the impact of post-accident management on the frequency of releases.

Probabilistic analyses are performed using “best estimate” data and assumptions, and consider all existing plant systems to provide a realistic risk prediction.

### 2.4 Safety Goals for Nuclear Power Plants

RD-337, *Design of New Nuclear Power Plants*, provides direction concerning the establishment of safety goals for the design of a nuclear power plant. Two qualitative safety goals have been established:

1. Individual members of the public are 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
2. Societal risks to life and health from nuclear power plant operation are 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.

Quantitative safety goals for existing nuclear power plants are set to achieve the intent of these qualitative goals. Quantitative safety goals are categorized into three frequency types:

1. **Core Damage Frequency (CDF)**: The sum of frequencies of all event sequences that can lead to significant core degradation.
2. **Small Release Frequency (SRF)**: The sum of frequencies of all event sequences that can lead to such a release that may require temporary evacuation of the local population.
3. **Large Release Frequency (LRF)**: The sum of frequencies of all event sequences that can lead to such a release that may require long term relocation of the local population.

The above safety goals include the contribution of facility-originated events (such as equipment failure, operator errors, internal fire, and internal floods) and external events (such as earthquakes, weather-originated events, and fire), but exclude malevolent acts.
The numerical values for these goals are found in Appendix B.1 and B.2.

3.0 SUBMISSION REVIEW

3.1 General Information

The depth and scope of the review of a submission is in proportion to its impact on safety. When a submission is supported by both deterministic and probabilistic arguments, the review is conducted as a multidisciplinary project.

The goal of the review is to check that the following general principles are met:

1. The status of the facility after implementation of the decision meets the relevant regulations and the current licence requirements;
2. Defence in depth is maintained; and
3. Sufficient safety margins are maintained.

The approach used for evaluating the safety impact of the submission is to ensure that all aspects of the submission have been addressed. Specialists of both deterministic and probabilistic safety analysis, facility operation, systems, maintenance, management, training, radiation protection, human factors, organizational factors, and software reliability are consulted as appropriate.

The review of the submission takes into account probabilistic and deterministic considerations, current regulatory requirements, and licence conditions. Information and insights from the probabilistic and deterministic analyses are considered, together with quantitative sensitivity studies, operational experience, historical facility performance, and engineering judgment.

The scope and quality of the analyses conducted to justify the submission are assessed for the appropriateness for the nature and scope of the proposal(s) contained therein and whether it is based on the as-built and as-operated facility. The assumptions and elements of the models used are assessed for whether they are correct and adequate for resolving the issue. Furthermore, it must be clear that there is a commitment to perform the diverse activities (monitoring, surveillance, operating and maintenance procedures, etc.) that are credited in the submission.

3.2 Use of Safety Goals

The probabilistic arguments supporting the licensee’s submissions include an evaluation of the absolute or relative change in risk metrics. The complexity and depth of this evaluation depends on the magnitude of the potential risk impact.

The results from the PSA, after modelling the proposal(s), are evaluated against the safety goals. The margin between the presented results (for example in terms of core
damage or releases) and the safety goal limits is used to weigh the probabilistic and deterministic arguments.

The result of the evaluation of the PSA against the safety goals will identify that the facility is in one of the following categories:

3. Frequency more than one decade\(^1\) lower than the safety goal;
4. Frequency between the goal and one decade lower than the goal; or
5. Frequency greater than the goal.

3.2.1 Frequency more than one decade lower than the safety goal

If the review of the submissions reveals that the facility meets the safety goal with a margin of more than one decade, measures to further reduce the identified safety risk are not expected and all normal licensing conditions apply.

3.2.2 Frequency between the goal and one decade lower than the goal

If the review of the submission reveals that the facility is within one decade of the safety goal, the licensee is expected to identify measures to reduce the identified safety risk.

The principle of As Low as Reasonably Practicable (ALARP) is used to determine the practical course of action. For a risk to be ALARP, the cost to further reduce the identified risk must be grossly disproportionate to the improvement in safety. Cost benefit assessment can be used to evaluate and select appropriate options to reduce the risk. The effort applied and cost incurred to reduce the risk should be in proportion to the deviation from the safety goal.

3.2.3 Frequency greater than the goal

If the review of the submission reveals that the facility does not meet the safety goal, compensatory measures to reduce the identified safety risk will be required in order to comply with relevant regulations and licence conditions.

3.3 Supporting Documentation

Data, methods, assessment criteria, and analyses used to support any proposed modification should be available for review, along with all appropriate documentation, including justification of the assumptions. The source of data and assumptions should also be available for review.

CNSC staff may choose to independently reproduce the results of the analysis, and modify the models, data, and assumptions presented in the submission. Sufficient information should be available to support this activity.

\(^1\) A difference of a factor of ten. For example, the difference between \(10^{-6}\) and \(10^{-5}\) is one decade.
During the review of a submission, deficiencies in the analysis and the models, or indications of facility weaknesses (for instance, features weakening defence in depth) may be detected. The licensee should provide any requested clarification or perform any necessary corrective action.

Such clarification or corrective action may include:

1. For the analysis and models:
   a) More analysis on an issue or a specific part of the analysis;
   b) Correction of defective parts of the analysis; and

2. For the facility:
   a) Modification of the facility configuration to correct weaknesses in its defence in depth or safety margins; and
   b) Modification of the operating policies and procedures to improve the safety margins or reduce uncertainty.
ADDITIONAL INFORMATION

Relevant International Standards

Key principles and elements used in developing this document are consistent with international standards:

1. International Atomic Energy Agency (IAEA), Safety of Nuclear Power Plants: Design, Safety Requirements NS-R-1, Vienna, Austria, 2000;
2. IAEA, Safety of Nuclear Power Plants: Operation, NS-R-2, 2000;
4. IAEA Safety Series No. 50-P-4, Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 1), 1992;
5. IAEA Safety Series No. 50-P-8, Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 2), Accident Progression, Containment Analysis and Estimation of Accident Source Terms, 1995; and

Relevant Canadian Nuclear Safety Commission Regulatory Documents

These regulatory documents provide guidance for safety analysis and safety goals:

1. P-242 Considering Cost-benefit Information (2000);
2. S-294 Probabilistic Safety Assessment (PSA) for Nuclear Power Plants (2005);
3. P-299 Regulatory Fundamentals (2005);
4. RD-310 Safety Analysis for Nuclear Power Plants (2008); and
APPENDIX A
Dose Acceptance Criteria for Nuclear Power Plants

The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event.

This dose is less than or equal to the dose acceptance criteria of:

6. 0.5 millisievert for any Anticipated Operational Occurrence (AOO); or
7. 20 millisievert for any Design Basis Accident (DBA).

For plants first licensed before 2008, these criteria are considered to be targets. Source: RD-337, Design of New Nuclear Power Plants.
APPENDIX B
Quantitative Safety Goals for Nuclear Power Plants

B.1 Existing Nuclear Power Plants
(First licensed before 2008)

<table>
<thead>
<tr>
<th>Frequency Type</th>
<th>Sum of frequencies of all event sequences that can lead to...</th>
<th>Safety Goal</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core Damage Frequency (CDF)</td>
<td>Significant core degradation...</td>
<td>should not exceed $10^{-4}$ per reactor year.</td>
</tr>
<tr>
<td>Small Release Frequency (SRF)</td>
<td>A release to the environment of more than $10^{15}$ Bequerel of Iodine-131</td>
<td>should not exceed $10^{-4}$ per reactor year.</td>
</tr>
<tr>
<td>Large Release Frequency (LRF)</td>
<td>A release to the environment of more than $10^{14}$ Bequerel of Cesium-137...</td>
<td>should not exceed $10^{-5}$ per reactor year.</td>
</tr>
</tbody>
</table>

Source: The CDF and LRF are derived from INSAG-12, Basic Safety Principles for Nuclear Power Plants, United States Nuclear Regulatory Commission, and the safety goals adopted by Ontario Power Generation and Atomic Energy of Canada Limited. The SRF comes from RD-337, applying the same ratio as for CDF and LRF.

B.2 New Nuclear Power Plants
(First licensed after 2008)

Notwithstanding the high level of safety achieved by meeting or exceeding the safety goals set out in Table B.1, further improvements in accident prevention are expected due to advances in such areas as technology, design, and the optimization of human-machine interfaces. The goals for new nuclear power plants are taken from RD-337.

<table>
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<td>should not exceed $10^{-5}$ per reactor year.</td>
</tr>
<tr>
<td>Large Release Frequency (LRF)</td>
<td>A release to the environment of more than $10^{14}$ Bequerel of Cesium-137...</td>
<td>should not exceed $10^{-6}$ per reactor year.</td>
</tr>
</tbody>
</table>
GLOSSARY

Acceptance criteria
Specified bounds on the value of a functional or conditional indicator used to assess the ability of a structure, system, or component to meet its design and safety requirements.

ALARP (As Low as Reasonably Practicable)
Risk should be reduced when it does not result in a cost disproportionate with the effective improvement.

Anticipated operational occurrence (AOO)
Operational process deviating from normal operation, such as the loss of normal electrical power, which is expected to occur at least once during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions.

Beyond design basis accident (BDBA)
Accident conditions less frequent and more severe than a Design Basis Accident (DBA). For a nuclear reactor a BDBA may or may not involve core degradation.

Design basis accident (DBA)
Accident conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

Deterministic Safety Analysis
Analysis of responses to an event; performed using predetermined rules and assumptions (e.g., those concerning the initial operational state, availability and performance of the systems and operator actions). Deterministic analysis can use either conservative or best estimate methods.

Nuclear power plant (NPP, plant)
Any fission-reactor installation that has been constructed to generate electricity on a commercial scale. A nuclear power plant is a Class IA nuclear facility, as defined in the Class I Nuclear Facilities Regulations. For the purposes of this document we have differentiated nuclear power plants as follows:

(1) Existing nuclear power plant - an NPP first licensed before 2008.
(2) New nuclear power plant – an NPP first licensed after 2008.
**Probabilistic Safety Assessment (PSA)**

For any Class I facility, a comprehensive and integrated assessment of the safety of the facility. The probabilistic safety assessment considers the probability, progression, and consequences of equipment failures or transient conditions to derive numerical estimates that provide a consistent measure of the safety of the facility.

For a nuclear reactor the PSA is generally structured as follows:

(3) A Level 1 PSA identifies and quantifies the sequences of events that may lead to the loss of core structural integrity and to massive fuel failures;

(4) A Level 2 PSA starts from the Level 1 results, and analyses the containment behaviour, evaluates the radionuclides released from the failed fuel, and quantifies the releases to the environment; and

(5) A Level 3 PSA starts from the Level 2 results, and analyses the distribution of radionuclides in the environment and evaluates the resulting effect on public health.

A PSA may also be referred to as a Probabilistic Risk Assessment (PRA).

**Safety analysis**

Analysis by means of appropriate analytical tools that establishes and confirms the design basis for the items important to safety; ensures that the overall plant design is capable of meeting the acceptance criteria for each plant state.