Regulatory Role of PSA in Canada and lessons learned from its use in advanced reactor facilities

Canadian Nuclear Safety Commission, Ottawa, Canada
hayat.chatri@cnsc-ccsn.gc.ca
michael.xu@cnsc-ccsn.gc.ca
abderrazzaq.bounagui@cnsc-ccsn.gc.ca

Hayat Chatri[a*], Xu Michael[a] and Abderrazzaq Bounagui[a]
[a]Canadian Nuclear Safety Commission, Ottawa, Canada

Abstract: In Canada, Probabilistic Safety Assessment (PSA) is formally integrated in the licensing and regulatory oversight of nuclear power plants throughout the whole life cycle. The Canadian Nuclear Safety Commission (CNSC) uses the information provided by the PSA to support the licensing, regulatory oversight program, to reexamine the efficacy of regulatory systems, to enhance emergency preparedness and, to develop a risk-informed compliance verification process.

The CNSC regulatory document REGDOC-2.4.2, "Probabilistic Safety assessment (PSA) for Reactor Facilities" [1] sets high level requirements which call for the development and application of Level 1 and Level 2 PSA.

The CNSC's current licensing strategy for the new builds is established based on the practice and experiences from the licensing of CANDU Nuclear Power Plants (NPPs). As part of an application for a licence to construct, CNSC-REGDOC 1.1.2 "Licence Application Guide: Licence to Construct a Reactor Facility" [2] requires that the application shall include a PSA conducted in accordance with REGDOC-2.4.2. It is expected that the PSA scope and level of detail, at this stage, should be sufficient to provide preliminary risk results and insights, and to demonstrate that the PSA objectives established in REGDOC 2.4.2 and REGDOC 2.5.2 "Physical Design – Design of Reactor Facilities" [3] are met.
The CNSC REGDOC 2.5.2 requires that the PSA shall be developed to demonstrate that the safety objectives including safety goals, as well as other safety system performance indicators have been achieved. It also requires that the application describe how the results of the PSA have been used to identify plant vulnerabilities to severe accident, to support the Structures, systems and components (SSCs) classification, and how the results of the PSA are used to provide insights into the accident management program.

This paper aims to provide an overview of current status, application and the regulatory aspects of PSA in Canada. In addition, it will provide a brief background on the recent regulatory use of PSA for advanced reactor facilities and discuss its technical challenges and lessons learned.

Keywords: PRA, PSA, SMR, advanced reactor

1. INTRODUCTION

The probabilistic safety assessment (PSA) is a comprehensive, structured and logical analysis aimed at identifying, assessing and quantifying the risks in complex technological or natural systems. It is recognized as one of the key approaches to assess the safety critical aspects for the existing plants and for the new designs. PSA provides insights into the strengths and weaknesses of the design and operation of a nuclear power plant as it assists plant management and the regulator to target resources where the largest benefit for plant safety can be obtained.

In Canada, PSA is formally integrated in the regulatory process of nuclear power plants throughout the whole life cycle of a nuclear power plant starting from the early design phase and continuing through the construction and operation phases up to the decommissioning phase. High level regulatory requirements for the development and application of PSA are established in the Canadian Nuclear Safety Commission (CNSC) regulatory document REGDOC-2.4.2 "Probabilistic Safety assessment (PSA) for Reactor Facilities" [2]. The current CNSC regulatory requirements on PSA are adequate for new reactor designs including SMRs, however more guidance is needed on some topics such as the scope and level of detail of PSA at each licensing stage, as well as the applicability of the numerical safety goals. Guidance is also needed on some technical topics such as human reliability analysis, and the modelling of passive safety features SSCs and safety functions.

2. BRIEF HISTORY ON PSA USES IN CANADA

The use of PSA in, or as an adjunct to, plant licensing has had a long history in Canada, going back to the first commercial Canada Deuterium Uranium (CANDU) prototype (Douglas Point) ([4], [5]).

The early designs of the CANDU reactors were based solely on deterministic rules and criteria such as, single/dual failure[1], and reliable design features such as redundancy, and diversity. The continuous effort for safety improvements, specifically by regularly trying to answer the "what if?" question, has led to concerns regarding the independence of the mitigating systems from the process systems based on knowing that the failure of the safety support systems could result in common cause (cross-link) failures of process and special safety systems. To address these concerns, Atomic Energy of Canada Limited (AECL) developed, in the late 1970's, an analysis approach called the Safety Design Matrix methodology. This is considered as the predecessor to PSA.

The first comprehensive PSA for CANDU reactors was completed by Ontario Hydro for the Darlington Station in 1987. As a pre-operational study, the Darlington Probabilistic Safety Evaluation (DPSE) played a significant role in optimizing the design of the station. A number of vulnerabilities were identified and corrected, for example.

Following the completion of DPSE, Ontario Hydro began a program of Probabilistic risk assessments (PRAs) for Pickering A and Pickering B nuclear power plants (NPPs). The Pickering A station PRA, completed in 1995, identified a number of single failures and changes were made to the moderator system during the Pickering A prior to Return to Service.

The Bruce B station PRA, completed in 1999, identified the need for improving the powerhouse venting system. AECL used its PSA methodology during the design process of its next product line, the CANDU 9 design, which incorporated a number of improved design features.

Currently, all licensees have completed the Level 1 and Level 2 PSAs for internal and external events during at-power and shutdown states. Internal hazards include hazards originating from the sources located on the site of the reactor facility that proceeds from a human error or from a failure of a SSC. External hazards are those hazards of natural or human-induced origin that originate outside the site (e.g., earthquakes, lightning, tornadoes, fires and floods, aircraft crash).

3. PSA IN REGULATORY FRAMEWORK

PSA is referred to in licensees' Nuclear Power Reactor Operating Licence (PROL) as a licence condition under the Safety and Control Area (SCA) "Safety Analysis". The objectives of the PSA are listed in section 3 of CNSC REDOC-2.4.2. PSA is also referred to in a number of CNSC regulatory documents as an approach that can be used to support other SCAs and regulatory programs, such as: Physical Design; Deterministic Safety Analysis (DSA); Reliability Program; Minimum Shift Complement; Maintenance Program; Aging Management; Accident Management, and Emergency Management.

[1] A "single failure" is a failure of any one process system. Dual failures, a much less likely event, are defined as a single failure coupled with the unavailability of one special safety system (one shutdown system, the emergency core cooling system, or impairments in containment).

3.1. PSA regulatory requirements

The CNSC issued the regulatory document REGDOC-2.4.2 [1] in April 2014 as an amendment of the previous CNSC standard S-294 [7] in response to CNSC Fukushima Task Force recommendations [8]. This regulatory document sets high level requirements which call for the development of Level 1 and Level 2 PSA by applying a formal quality assurance process for conducting the PSA. The scope of the PSA shall include internal and external events, both at-power and shutdown operational states, multi-unit impacts, as well as the inclusion of sensitivity analysis, uncertainty analysis, and importance measures. It also requires the licensees to seek CNSC acceptance of the methodology and computer codes to be used for the PSA, which means that the methodology and the computer codes have to be formally accepted by the CNSC prior to the submission of the PSA reports. As a basis for methodology acceptance, REGDOC-2.4.2 refers to the IAEA specific safety guides SSG-3 [9] and SSG-4 [10].

REGDOC 2.4.2, version 2 [1] was issued in May 2022 to take into account the recent developments of new nuclear and small modular reactor technologies. This document retained the PSA requirements and guidance included in the previous version with the clarification that these requirements are set out for a licence to construct or operate a reactor facility. It also clearly specifies that the PSA requirements are applicable to reactor facilities, including nuclear power plants, small reactor facilities, or non-power reactors (research and test reactors), using a graded approach.

The Canadian Standards Association (CSA) Group standard on PSA, CSA-290.17 [11] provides guidance as to how a licensee can achieve requirements currently found in REGDOC-2.4.2. This standard includes clear requirements for the Licensees to develop and implement a systematic process to update a PSA.

3.2. Safety Goals, Criteria and Risk Metrics

At the CNSC, the ultimate output of safety is the prevention of unreasonable risk as endorsed in the Nuclear Safety and Control Act [12]. Safety Goals, as introduced in REGDOC 2.5.2, are defined in a more comprehensive way by combining both qualitative goals and quantitative goals.

Two qualitative safety goals are established based on the principle that nuclear risks posed by NPP operation should not be a significant addition to other societal risks. The qualitative safety goals are:

- Individual members of the public shall be provided a level of protection from the consequences of reactor facility operation, such that there is no significant additional risk to the life and health of individuals.
- Societal risks to life and health from reactor facility operation shall be comparable to or less than the risks of generating electricity by viable competing technologies, and shall not significantly add to other societal risks.

The quantitative safety goals are formulated in addition to the dose acceptance criteria and to deterministic design requirements to ensure that risk to the public originating from accidents outside the design basis is considered. Quantitative safety goals established in REGDOC 2.5.2 include:

- Core damage frequency (CDF): The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than 10^{-5} per reactor year.
- Small release frequency (SRF): 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} becquerels of iodine-131 shall be less than 10^{-5} per reactor year.
- Large release frequency (LRF): 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} becquerels of cesium-137 shall be less than 10^{-6} per reactor year.

The Safety Goals for existing NPPs are adopted by the licensees in conformance with International best practice, specifically the IAEA Safety Series No 75-INSAG-3 (1988) [13] which was updated in 1999 as INSAG-12 [14]. Canadian licensees use different terms to characterize these safety goals: Safety Goal;

Safety Goal target; Administrative Safety Goal. However, all these definitions come down to the same general principle of establishing safety goal not to be exceeded, and a Target "administrative safety goal" to strive for.

Safety goals adopted by Canadian licensees are shown in Table 1 below:

Table 1. Canadian Licensees definition of Safety Goals

	Safety Goal (per reactor.year)	Administrative Safety Goal "Target" (per reactor.year)
Severe Core Damage Frequency	10^{-4}	10^{-5}
Large Release Frequency	10^{-5}	10^{-6}

The industry application of Safety Goals to implement design upgrades is defined as follows:

- If assessed risk metrics are above safety goal: modifications are necessary.
- If assessed risk metrics are between safety goal and administrative safety goal: modifications are necessary if practicable.
- If assessed risk metrics are below the administrative safety goal: modifications are expected if practicable.

3.3. PSA regulatory role

In 2018, CNSC issued a regulatory fundamentals document REGDOC-3.5.3 "Regulatory Fundamentals" [15] which describes risk informed approach to licensing and compliance activities where the focus is put on issues of higher risk for effectiveness and efficiency, but it does not refer to the use of PSA risk insights (qualitative and/or quantitative results), nor to the safety goals. This REGDOC has been updated in 2023 to expand information on the CNSC's risk-informed and graded approaches to nuclear regulating, particularly in regulation of new technologies such as small modular reactors (SMRs).

In addition, all REGDOCs prefaces evoke the use of the "Risk informed approach". In this approach, the PSA results can be used to complement the deterministic approach, and Risk Informed Decision Making (RIDM) key principles (PSA limitations and uncertainty should be considered). Figure 1. below illustrates the CNSC RIDM process.

The RIDM process:

- helps in prioritizing goals and activities,
- ensures that all aspects of a given issue are identified and considered when making decisions,
- is easily understandable, well documented and structured,
- is adaptable to various decision-making situations,
- enables decision-makers to explain decisions,
- respects stakeholder consultation, and
- is founded on open communication

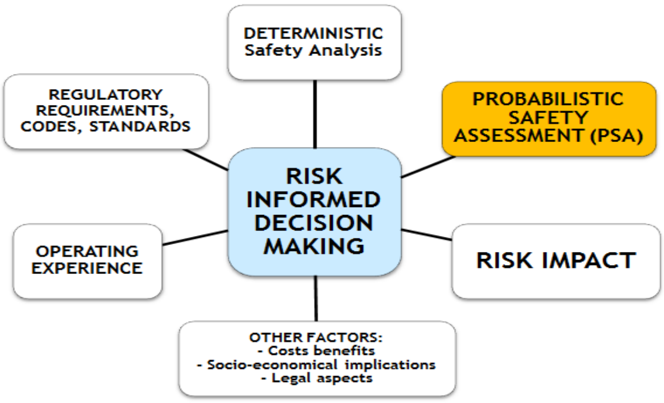

Figure 1. CNSC RIDM Process

As a comparison of PSA uses in other countries, US as an example, US NRC issued different Regulatory Guides and other RIDM processes which promote the use of the PSA results and provide guidance on the thresholds for change in core damage frequency (ΔCDF) and Large Early Release Frequency (ΔLERF) for characterizing the risk significance of issues. As an example,

- RG 1.174 "*An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*" [20],
- NRR Office Instruction LIC-504 "*Integrated Risk-Informed Decision-making Process for Emergent Issues*" [21].

In both these documents, acceptance guidelines regarding incremental risk calculated by the PSA (ΔCDF and ΔLRF) are provided. As an example, ΔCDF < $1x10^{-6}$ per reactor year and ΔLERF < $1x10^{-7}$ per reactor year serve as thresholds to show that issues are of very low risk significance.

The regulatory role of the PSA, using the RIDM approach, can be grouped in the following broad regulatory activities:

3.3.1 Licensing: comparison of PSA results against safety goals

For the CNSC, Safety Goals are quantitative indicators of the overall safety of the plant. The safety Goals form part of the licensing basis to identify design improvements to enhance safety and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.

The main tool for demonstrating that the NPP design meets Safety Goals is PSA; Level 1 for the assessment of plant failures and responses of reactor systems (core damage frequency), and Level 2 for the assessment of containment response (large release frequency). The consequences to the public (doses) and to the environment are assessed using the Level 3 PSA. Consistently with the international practice, this level of assessment is not required by the CNSC.

The PSA results are compared against the established safety goals. However, it is internationally accepted that the probabilistic safety goals should not be used as the sole basis for regulatory decisions. It is also recognized that a firm regulatory limit cannot be stated on a risk metric that is assessed using a purely probability-based approach.

3.3.2 Regulatory oversight

CNSC staff use the information provided by the PSA to support the regulatory oversight program. Indeed, as part of the ongoing risk-informed, performance-based approach in support of the regulatory compliance verification program, including site inspection activities, CNSC staff completed an internal project, called the Risk Handbook, to summarize the licensees' PSA results, important risk insights and the performance of important safety systems for CNSC inspectors. The project has two main objectives: to provide introductory

PSA training to CNSC site inspectors, and to develop a user-friendly, web-based tool summarizing licensees' PSA results and insights.

The handbook can be used by CNSC inspectors to:

1. Optimize inspection planning and improve efficiency:
   - Focus on risk significant SSCs, Initiating Events, and Human Actions, Specific hazard information
2. Provide risk insights to define the scope of existing regulatory inspection procedures:
   - Electrical/Mechanical Systems Inspections
   - Human performance
   - environmental qualification (EQ), and seismic qualification inspections
   - Fire and flood hazards inspections
3. Evaluate inspection results, and provide an understanding of the risk impacts in situations where a SSC is taken out of service, or a system experiences an impairment level
4. Help in determining the safety significance of operational events

3.3.3 Operational event evaluation and abnormal plant configurations

CNSC staff use the PSA in combination with the DSA to evaluate the potential consequences of an operational event. PSA-based event analysis provides a quantitative assessment of the risk significance of the event by calculating the risk increase induced by the event. Operational event review is used to provide an input into the consideration of what changes could be made to reduce the likelihood of recurrence of such operating events.

PSA is also used as an integrated model to evaluate the risk increase due to an abnormal plant configuration where multiple SSCs unavailability has occurred. This unavailability could be due to equipment random failures or to situations where SSCs are taken out of service for maintenance activities.

3.3.4 Changes to licensing basis

CNSC staff use the PSA results and insights in combination to the DSA to assess the impacts of licensees' proposed changes to the licensing basis. These changes may include:

- Changes in test/maintenance interval
- Changes in Operating Policies and Procedures (OP&Ps)
- Request for extending allowed outage time of an SSC.

3.3.5 PSA use in Severe Accident Management and emergency preparedness.

Since the severe core damage accidents at Three Mile Island, Chernobyl, and Fukushima, the identification and mitigation of severe accidents has become a major part of nuclear safety. PSA provides most of the input to this process: it systematically identifies severe accidents, provides a frequency so that mitigation measures can be applied in a risk-effective manner, and predicts the consequences (through PSA support analyses) so there is a sound basis for on-site accident management and off-site emergency planning.

In Canada, the CNSC is promoting the use of PSA insights in defining the strategies to cope with the consequences of severe accidents. Indeed, the Level 2 PSA has been used to identify accident management measures that could be carried out to mitigate the effects of a severe accident. This has led to the implementations of generic or plant specific Severe Accident Management Guidelines (SAMG) to guide operators in the event of a severe accident.

In addition, the source terms and frequencies produced by the Level 2 PSA have been used as the basis for Emergency Preparedness Drills and Exercises.

4. CNSC REGULATORY USE OF PSA FOR ADVANCED REACTOR FACILITIES

4.1. CNSC Licensing process and PSA Regulatory Requirements for advanced reactor facilities

The Canadian nuclear industry has shown strong interest in the development and deployment of advanced and small modular reactors (SMRs) within Canada[2].

The CNSC has been preparing for the regulation and licensing of SMRs and advanced reactors for over a decade and is continuing to invest efforts towards readiness activities. The CNSC is also currently reviewing license applications for advanced reactors/SMRs.

The CNSC licensing process for all reactor facilities include the following:

- Licence to prepare a site: This includes the preparation of an environmental assessment.
- Licence to construct: a preliminary safety report (PSAR) as well as a PSA shall be prepared as part this application.
- Licence to operate: a final safety report (FSAR), a PSA reflecting the as-built as-operated Conditions, and a reliability program for systems important to safety, should be included as part of this application.
- Licence to Decommission.
- Licence to Abandon.

CNSC also provides a pre-licensing service known as Vendor Design Reviews (VDR) [16] to provide early feedback on how the vendor may comply with CNSC regulatory requirements. Several SMR vendors have engaged the CNSC through the VDR Program.

As per the CNSC Regulatory document REGDOC 2.5.2, a safety analysis of the plant design shall include hazard analysis (HA), DSA, and PSA. The safety analysis shall demonstrate achievement of all levels of defence in depth and confirm that the design can meet the applicable expectations, dose acceptance criteria and safety goals. The PSA shall be conducted in accordance with the requirements specified in REGDOC-2.4.2 [1].

As part of an application for a licence to construct, REGDOC 1.1.2 requires that the application shall include a PSA conducted in accordance with REGDOC-2.4.2 . It also requires that the application should describe how the results of the PSA have been used to identify any vulnerabilities, and how the results of the PSA are used to provide insights into the severe accident management program, and how these results meet the safety goals.

For a licence to operate application, REGDOC 2.4.2 [1] requires that the PSA shall reflect the as-built as-operated conditions. A reliability program, in accordance with REGDOC 2.6.1 [17], is also required for the purpose of identifying systems important to safety (SIS), establishing reliability targets, and reporting annually to the CNSC on the performance of these SISs.

4.2. Regulatory Review of PSA Technical Elements and challenges for SMRs

SMRs incorporate novel aspects and features that represent a departure from traditional CANDU technology. The design of these reactors encompasses a wide array of technologies, most of which have limited operational experience.

PSA for SMRs is recognized as an important approach to achieve improved safety for the future nuclear facilities. However, the application of PSA to SMRs, which have limited operational experience, encounters concurrent challenges which can include a lack of design detail, a lack of empirical data and the possibility of failure scenarios that differ in character from those treated in CANDU PSAs. These challenges may lead to PSA results that do not reflect the future as-built, as-operated plant.

While the current CNSC regulatory requirements on PSA are adequate for SMRs, additional guidance is needed on some PSA elements. The following sections include a discussion about the PSA technical

[2] The term SMR used throughout the paper includes advanced nuclear reactors which may not be small in size.

elements and the challenges associated with their applicability to SMRs, and the CNSC staff guidance and expectations regarding each PSA technical element.

4.2.1 PSA scope and level of detail

For a licence to construct application, the applicant is required to seek CNSC acceptance of the methodology and computer codes used for the PSA. The scope and level of detail should be consistent with the risk impact, using a graded approach.
The challenge encountered is whether a full scope and detailed PSAs should be required for all reactor types and sizes at the different licensing stages, and the extent to which a graded approach can be used.
As an example, for some designs, the relatively low consequences may lead to the use of simple PSA models, to provide an indicative risk estimate. The scope may also be limited to at-power state only, and some internal and External Hazards.
The expectation from a regulatory perspective is that the applicant should document and justify the scope reduction of the PSA and to seek CNSC staff acceptance.

4.2.2 PSA Quantitative Safety Goals

The challenge about the application of the current quantitative safety goals, established in REGDOC 2.5.2, is that for designs using pebble-bed fuel (TRISO), or liquid fuel, CDF, SRF and LRF may not be considered to be applicable. The regulatory expectation is that whenever the quantitative safety goals are shown to be not applicable, the applicant shall demonstrate that the qualitative safety goals are met, including the surrogates for SRF and LRF quantitative safety goals, i.e., the temporary evacuation and the long-term relocation of the local population, respectively. CNSC is also considering the development of a Quantitative Health Objective (QHO) based on cancer incidence, as well as establishing a dose acceptance criteria for the Design Extension Conditions (DEC) which would be applicable to all nuclear power plants and SMR designs.

4.2.3 Initiating Event Analysis

The applicant is required to apply a systematic approach to provide a reasonably complete list of postulated initiating events (PIEs). For SMRs and unique designs, the challenge is the lack of operating experience (OPEX) and the absence of similar reactors to compare the list of PIEs to ensure completeness of the list and its use as a reference for deriving PIEs frequencies. The challenge may also include the lack of understanding of certain phenomena that could result in omitting some initiating events.
As indicated in IAEA Safety Report Series 25 [18], it is recognized that it is not possible to demonstrate completeness for first of a kind projects. However, by using a combination of inductive and deductive analysis methods, it should be possible to gain confidence that the contribution to the risk from initiating events which have not been identified would be small.
The regulatory guidance and expectations suggest the use of both a top-down approach, such as the Master Logic Diagram (MLD) and the bottom-up approach such as the Failure Mode and Effects Analysis (FMEA), and to provide the rationale for screening out any initiating event. It may also be necessary to use of Phenomena Identification and Ranking Technique (PIRT) to rank the importance of some phenomena.

4.2.4 Accident Sequence Analysis and System Analysis

The objective of the accident sequence analysis is to adequately model the systems responses and operator actions that affect the key safety functions for each initiating event.
For SMRs and unique designs, the lack of experience and reference studies may result in an incomplete understanding of the mitigation systems' responses and success criteria, Operator actions, as well as certain phenomena in the event sequences. There could be a challenge in modelling the plant specific dependencies, and the passive safety features, including:

- Identification of failure modes from natural phenomena.
- Modelling a suitable range of environmental and operational conditions due to less driving forces.
- Phenomena-based parameters (e.g., heat sink temperature, non-condensable gas fraction, convective heat transfer, etc.), including interdependencies.

It is, therefore, expected that the safety functions required to prevent core damage for the individual initiating events are identified, along with specific plant systems required to perform those safety functions, the systems success criteria, and related operating procedures. It is also recommended to develop event-sequence

diagrams (ESDs) for each group of initiating events to facilitate the collection and display of information required for developing system event trees, especially if the large event tree/small fault tree approach is used. Furthermore, the qualified safety analysis tools may not be yet available to support Level 1 and Level 2 PSA, specifically for the source term quantification.
The challenge also includes the PSA end states definition and whether there is a need for a separate Level 1 and Level 2 PSA.
For system analysis, the challenges include the fault tree development for new systems and system level failure criteria, modelling of passive systems, Digital I&C, mission times, time windows for operator actions, and the modelling of intersystem and intra-system dependencies.
It is expected that the applicant will identify and fully document the boundaries of each system, and to include direct functional dependencies that are incorporated directly into the fault or event trees, and the dependencies with no direct causal relationships. This task can be facilitated by the development of dependency matrices.

4.2.5 Human Reliability Analysis (HRA)

SMRs present new concepts of operation enabling a smaller operating crew for operating a single reactor or, multiple reactor units in parallel. As noted in Ref [19], the unique aspects of the modern human-system interface (HSI) in the SMR cannot be readily accounted for in the Technique for Human Error Rate Prediction (THERP), and newer HRA methods may more readily generalize to the types of technologies and concepts of operation found in SMRs.
The general challenges in HRA for SMRs include the lack of HRA data, and the lack of understanding of human actions' impact on system unavailability and the dependency between human actions, specifically at the design stage or during an application for a licence to construct.
The applicant is expected to document the HRA methodology used for the identification of the human failure events (HFEs) as well as the quantification of the human error probabilities (HEPs) taking into consideration the specific design features of SMRs such as the modern HIS, digital controls and displays, computerized procedures, and automated actions. The methodology should also describe instances where the operator may need to take recovery actions when passive safety system function is impaired.

4.2.6 Data analysis

Data analysis includes the statistical analysis of raw information, the use of generic and specific data, and/or the use of expert judgment data. The necessary data include the frequencies of initiating events, component-failure rates, common-cause probabilities, and operational data such as repair times, test frequencies and test downtimes.
The challenges related to data analysis for SMRs and unique designs include the lack of operational data to derive initiating events' frequencies and component or system unavailabilities specifically for first of a kind (FOAK) components.
The applicant is expected to document the data analysis methods. The data may be generic industry data or plant-specific data, or a combination of both, and engineering judgment supplemented with an uncertainty characterisation.

5. CONCLUSION

The Canadian Nuclear Safety Commission (CNSC) uses risk-informed approaches to support its regulatory activities. CNSC staff use the licensees' PSA results and insights to support regulatory activities, such as licensing, regulatory oversight, review and evaluation of operational events and abnormal plant configurations.

PSA for SMRs has been used from the very early stage of SMR design phase as an important approach to achieve improved safety for the future reactor facilities.

The current CNSC regulatory requirements on PSA are adequate for SMRs, however more guidance is needed on some PSA topics such as the scope and level of detail of PSA at each licensing stage, as well as the applicability of the numerical safety goals. Guidance is also needed on some technical topics such as human reliability analysis and the modelling of passive safety features (SSCs and safety functions). This new guidance will be forthcoming once more experience is gained. Indeed, the CNSC applies a risk-informed approach to regulation focusing on the safety risks for each licensing phase. The CNSC has the regulatory tools and legislative power to make decisions as new guidance is being developed in an evolving environment.

References

[1] Canadian Nuclear Safety Commission (CNSC), regulatory document REGDOC-2.4.2 Version 2, "Probabilistic Safety Assessment (PSA) for Reactor Facilities", Canada, May 2022.

[2] Canadian Nuclear Safety Commission (CNSC), regulatory document REGDOC-1.1.2 Version 2, "Reactor Facilities: Licence Application Guide: Licence to Construct a Reactor Facility", Canada, Oct. 2022.

[3] Canadian Nuclear Safety Commission (CNSC), Regulatory Document REGDOC-2.5.2, "Physical Design, Design of Reactor Facilities", Version 2.1, Canada, May 2023.

[4] V.G. Snell, "Probabilistic Safety Assessment Goals in Canada", in "Status, Experience and Future Prospects for the Development of Probabilistic Safety Criteria", Report of a Technical Committee Meeting, International Atomic Energy Agency report IAEA-TECDOC-524, Vienna, 1986.

[5] Y. Zeng, P. Webster & P. Hessel, "Probabilistic Safety Assessment in Canada", Reliability Engineering and System Safety 62 (1998) 33–42.

[6] Canadian Nuclear Safety Commission (CNSC), regulatory document REGDOC-2.4.2, "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants", Canada, May 2014.

[7] Canadian Nuclear Safety Commission (CNSC), "Probabilistic Safety Assessment for Nuclear Power Plants", regulatory document S-294, Canada, April 2005.

[8] Canadian Nuclear Safety Commission (CNSC), "CNSC Fukushima Task Force Report", CNSC INFO-0824, Canada, October 2011.

[9] International Atomic Energy Agency (IAEA), "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants", IAEA Safety Standards Series No. SSG-3, IAEA, Vienna (2010).

[10] International Atomic Energy Agency (IAEA), "Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants", IAEA Safety Standards Series No. SSG-4, IAEA, Vienna (2010).

[11] Canadian Standards Association, "Probabilistic safety assessment for nuclear power plants", CSA-N290.17, Canada, November 2023.

[12] MINISTER OF JUSTICE OF CANADA, "Nuclear Safety and Control Act", (1997). http://laws-lois.justice.gc.ca

[13] International Atomic Energy Agency (IAEA), International Nuclear Safety Advisory Group (INSAG), "Basic Safety Principles for Nuclear Power Plants" IAEA Safety Series No 75-INSAG-3 (1988).

[14] International Atomic Energy Agency (IAEA), International Nuclear Safety Advisory Group (INSAG), "Basic Safety Principles for Nuclear Power Plants" IAEA Safety Series No 75-INSAG-3 Rev 1 INSAG-12, (1999).

[15] Canadian Nuclear Safety Commission (CNSC), Regulatory document REGDOC-3.5.3, Version 2.1 "Regulatory Fundamentals", February 2022, REGDOC-3.5.3, Regulatory Fundamentals, Version 2.1 - Canadian Nuclear Safety Commission

[16] Canadian Nuclear Safety Commission (CNSC), Regulatory document REGDOC-3.5.4, "CNSC Processes and Practices: Pre-Licensing Review of a Vendor's Reactor Design", November 2018,

[17] Canadian Nuclear Safety Commission (CNSC), Regulatory document REGDOC-2.6.1, "Fitness for Service: Reliability Programs for Nuclear Power Plants", August 2017,

[18] International Atomic Energy Agency (IAEA): "Review of Probabilistic Safety Assessments by Regulatory Bodies", IAEA Safety Report Series 25, Vienna, 2002

[19] Ronald L. Boring and David I. Gertman, "Human Reliability Analysis for Small Modular Reactors", INL/CON-12-25622, PSAM-11, June 2012.

[20] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" January 2018.

[21] U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, NRR Office Instruction, LIC-504 Revision 5: "Integrated Risk-Informed Decision-Making Process for Emergent Issues", March 4, 2020.