U.S. NRC – CNSC Memorandum of Cooperation

INTERIM JOINT REPORT

concerning

Classification and Assignment of Engineering Design Rules to Structures, Systems and Components

June 2023

DISCLAIMER: The NRC and CNSC have prepared this interim report to inform stakeholders of the current project status for performing an assessment of similarities and differences between regulatory frameworks regarding safety classification and application of engineering design rules. The information contained in this document has not been subject to NRC and CNSC management and legal review, and its contents are subject to change and should not be interpreted as official agency positions.

USNRC ML23172A201 CNSC e-Docs 7057767

Preface

In August 2019, the Canadian Nuclear Safety Commission (CNSC) and the U.S. Nuclear Regulatory Commission (NRC) signed a memorandum of cooperation (MOC) to increase collaboration on technical reviews of advanced reactor and small modular reactor (SMR) technologies.

The MOC builds on the joint memorandum of understanding signed in August 2017 and further strengthens the CNSC and USNRC commitment to share best practices and experience from design reviews. Additional information on international agreements and the CNSC can be found at: https://nuclearsafety.gc.ca/eng/resources/international-cooperation/international-agreements.cfm

The outcomes of this cooperative activity are intended to help each jurisdiction leverage information from each other in reviewing advanced reactor designs and further facilitate the capability to perform joint technical reviews of advanced reactor designs that have been submitted for review in Canada and the United States. The activity aims to promote a mutual understanding of each organization's regulatory framework with a focus mainly on safety analysis expectations which are fundamental to the safety case that would support a licence application.

Cooperative activities between both organizations are established and governed under Terms of Reference (see https://nuclearsafety.gc.ca/eng/resources/international-cooperation/international-agreements/cnsc-usnrc-smr-advanced-reactor-moc-tor.cfm) and are designed to do the following:

- Contribute to better use of the regulators' resources by leveraging the technical knowledge and capabilities between the NRC and the CNSC.
- Enhance the depth and breadth of the respective CNSC and NRC staffs' understanding of the counterpart nation's regulatory review activities and requirements.
- Enhance the joint opportunities for learning about and understanding the advanced reactor and small modular reactor technologies being reviewed.

As part of the program of work, the CNSC and the NRC issued CNSC-NRC Report: "Technology Inclusive and Risk-Informed Reviews for Advanced Reactors: Comparing the Licensing Modernization Project with the Canadian Regulatory Approach" (CNSC Approach/LMP Comparison). The report recognizes the increased use of risk information in regulatory decision-making and focused on reviewing and comparing the technology-inclusive and risk-informed application approaches in each country. More specifically, it examined the technology-inclusive, risk-informed, and performance based (TI-RIPB) process developed as part of the Licensing Modernization Project (LMP) led by the U.S. nuclear industry, sponsored by the U.S. Department of Energy and endorsed by the NRC, and compared it with the requirements set out in CNSC regulatory requirements. In both approaches, vendors and applicants need to identify licensing basis events, to classify structures, systems and components (SSCs), and to ensure adequate defence-in-depth, which are the fundamental building blocks for establishing the licensing basis and content of a licence application.

The CNSC Approach/LMP Comparison report concludes there is much common ground in safety case assessment reviews and acceptance criteria that can be used as a foundation for technical reviews performed by one regulator to be leveraged by the other, in order to inform the independent regulatory findings and decisions required by law. The report suggested further work to assess the bases of key regulatory criteria where differences could exist and additional convergence could be achieved. The areas suggested for further work included classification of SSCs and investigation of the potential for greater harmonization through comparison of codes and standards related to quality assurance and management systems; and technical acceptance criteria in mechanical, electrical, structural, and instrumentation and control disciplines.

Executive Summary

As part of the program of work under the CNSC and NRC MOC, and in recognition of the recommendations of the CNSC Approach/LMP Comparison, a work plan was approved to document the results of the combined efforts of the CNSC and the NRC with a focus on:

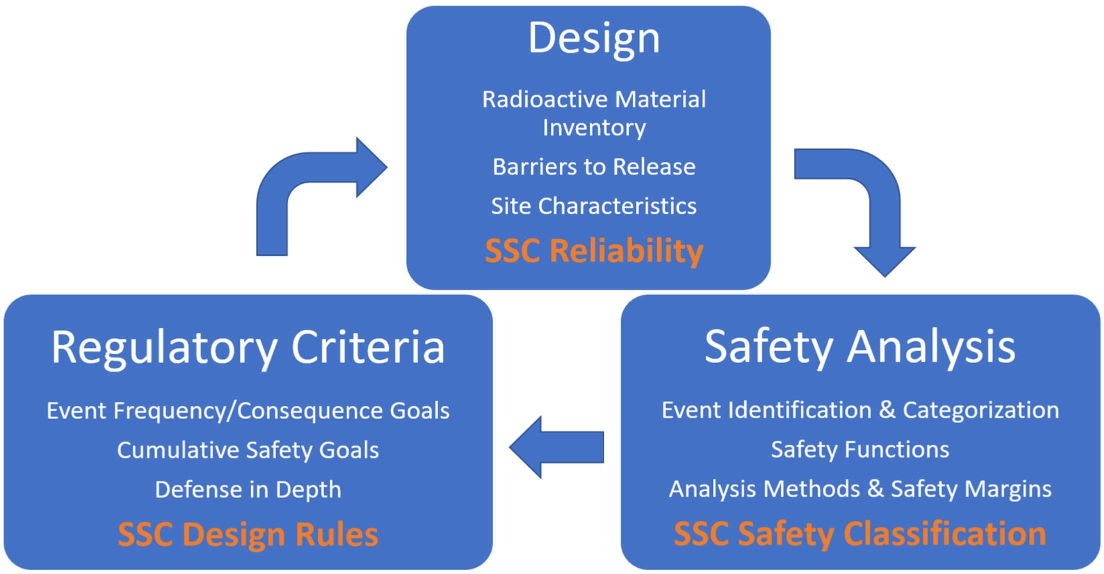
- Identification of key similarities and differences in the safety-significance determination process, the scope of SSCs subject to the process, and the process outcomes.
- Identification of key similarities and differences in the engineering design rules and specifications applied to each safety class and how this impacts the outcomes.
- Comparison of how each organization applies existing codes and standards and interacts with Standards Development Organizations to verify appropriate codes and standards are being developed, applied, and endorsed.

This report is an interim report that addresses the similarities and differences in the safety classification process, including scope and outcomes, and two areas involving application of engineering design rules: reliability assurance programs and pressure retaining components and supports. The purpose of providing an interim report is to enhance the understanding of applicants, potential applicants, interested stakeholders, and the respective staff of the CNSC and NRC on each nation's regulatory framework and requirements as they relate to safety classification of SSCs. The interim report also supports understanding of the relationship between safety classification and the application of engineering design rules in the two selected areas, which model potential approaches toward addressing other technical areas and related codes and standards in those areas.

The existing regulatory frameworks for each country have proven to be effective in protecting the health and safety of the public, ensuring the safety of workers, and protecting the environment. The regulatory frameworks considered in this report consist of the CNSC approach in Canada and the NRC traditional and LMP approaches in the United States. The CNSC Approach/LMP Comparison describes the more technology-neutral and risk informed regulatory approaches in place and available to both reactor designers/vendors and applicants for permits and licenses for potential new power reactors in Canada and the United States. This report summarizes information from the CNSC Approach/LMP Comparison report and adds comparable information regarding application of the NRC traditional regulatory framework to the licensing of water-cooled small modular reactors and advanced reactors. In addition, this report documents the results of the collaborative activities between CNSC and NRC on specific areas of the regulatory review process: the classification of structures, systems, and components (SSCs) important to safety and the assignment of engineering design rules to those SSCs based, in part, on their safety classification. In accomplishing these tasks, the CNSC and NRC staff considered the interrelationship of design, safety analyses, and regulatory criteria.

The following figure shows this relationship as an iterative process beginning with an initial reactor design. The design includes information related to reliability and availability of individual SSCs. The safety analysis process then evaluates how effective the design is at protecting the public health and safety and the environment around the reactor. The safety analysis process also identifies the SSCs performing important functions related to that protective capability and establishes safety classifications of individual SSCs based on the importance of their safety function. The outputs of the safety analysis are compared against regulatory criteria to determine the acceptability of the design for licensing. Designers may determine that modifications are necessary to satisfy the regulatory criteria and continue with iterative cycles through the design, safety analysis, and regulatory criteria evaluation steps until the design is sufficiently refined. The regulatory criteria include provisions for assignment of design rules to ensure the quality and reliability of individual SSCs are commensurate with the importance of their safety functions.

Interrelationship of Design, Safety Analyses, and Regulatory Criteria



The regulatory frameworks in both Canada and the United States reflect extensive experience with the licensing and oversight of water-cooled reactors. Both regulatory frameworks include provisions to demonstrate acceptable levels of safety through alternative means, such as under conditions where a regulatory requirement is not consistent with the proposed reactor technology. The NRC exemption process is more prescriptive, but both regulatory bodies consider similar criteria in evaluating alternative means of satisfying regulatory requirements. Therefore, the regulatory frameworks support licensing of small modular reactors and advanced reactors using the NRC traditional and the CNSC approaches. The NRC staff endorsed the LMP approach for licensing of advanced reactors, and ongoing NRC rulemaking is expected to provide for technology-neutral licensing using the risk-informed LMP approach.

The collaborative evaluation of the three approaches focused on identifying similarities and differences in each regulatory approach that could affect the safety classification process, the scope of SSCs considered safety significant, and the assignment of engineering design rules. Identified similarities include expectations regarding (1) the development and classification of event sequences, (2) the general incorporation of risk information into the safety analysis, (3) the identification of safety-significant functions, and (4) the classification of SSCs based on functions. Identified differences relate primarily to (1) the degree the regulatory approach is risk-informed, (2) the boundary values and specific acceptance criteria applied in the dose consequence and safety assessments, and (3) the process for assigning safety classifications to SSCs. The following table compares the use of probabilistic analyses, safety analysis approaches under each event classification, and the safety classification of SSCs relied on in the safety analyses.

Comparison of Licensing Approaches

Licensing Approach	NRC Traditional	CNSC	NRC LMP
Use of Probabilistic Analyses	Confirmatory and for Development of Risk Insights	Complementary to Deterministic Analyses	Foundational; Supported by Deterministic Analyses
Classification of Mitigating SSCs	Safety Related for Anticipated Operational Occurrences and Design Basis Accidents	Important to Safety	Safety Related for Design Basis Accidents and Certain Beyond Design Basis Events
Anticipated Operational Occurrences	Initiating Event Frequency; Conservative Analyses	Sequence Frequency; Best Estimate	Sequence Frequency; Best Estimate w/Uncertainty
Design Basis Accidents	Guidance for Initiating Event Selection; Conservative Analyses	Sequence Frequency; Conservative Analyses or Best Estimate w/Uncertainty	Sequence Frequency; Best Estimate w/Uncertainty
Beyond Design Basis Events	Special Regulations; Deterministic Requirements	Sequence Frequency; Best Estimate	Sequence Frequency; Best Estimate w/Uncertainty

The collaborative review by the CNSC and NRC determined that the tree regulatory approaches would support establishing comparable levels of safety consistent with the similar overall safety objectives of the two regulatory bodies. Similarities in the identification and categorization of events and the necessary safety functions that must be accomplished are expected to produce similar results in the identification and classification of SSCs performing important safety functions. The differences primarily reflect the degree the safety-analysis is risk-informed and the practice of compensating for uncertainty due to lower levels of risk information by increasing conservatism. This increase in conservatism may lead to a larger scope of SSCs having the highest safety classification in the less risk-informed approaches. However, the increased use of risk information would be expected to minimize this difference.

The review determined that the commonalities in the regulatory approaches would support joint reviews by the CNSC and NRC as well as provide the opportunity for applicants to leverage information developed for one regulatory body in developing an application for the other regulatory body. Applicants may minimize differences in the outcome of the safety analyses and SSC safety classification processes through selection of the most safety-significant systems for mitigation of design basis accidents in the safety analyses. The less prescriptive nature of the CNSC regulatory approach facilitates leveraging of NRC outcomes for development of an application to the CNSC. However, CNSC regulatory framework outcomes with respect to safety classification and assignment of engineering design rules may be leveraged for an application to the NRC with appropriate evaluation and reconciliation of regulatory requirements. These regulatory requirements relate primarily to the conformance with principal design criteria for the facility and appropriate assignment of engineering design rules in specific technical areas, including seismic qualification, quality assurance, and environmental qualification.

As an interim report, two pilot areas involving application of engineering design rules have been addressed: reliability assurance and pressure retaining components and supports. The collaborative review noted substantial alignment related to the implementation of reliability assurance programs and the assignment of specific engineering design rules to pressure retaining components and associated supports. Alignment in the implementation of the reliability assurance program may be increased through broader application of the program under the NRC traditional approach for water cooled and advanced reactors. For pressure retaining components and associated supports, the CNSC and NRC share similar water-cooled reactor code classification definitions and assign the same design code to each code classification. For advanced reactors, the code classifications and associated standards are expected to be similar.

The final report will build on this interim report. Specifically, the final report will include an evaluation similarities and differences in the application of engineering design rules and the development, endorsement, and application of codes and standards. The scope of this evaluation will address the following 10 subject areas (which include the two pilot areas addressed in this interim report) related to application of engineering design rules:

Technical Areas to be Evaluated for assignment of Engineering Design Rules

Programs

- Reliability Assurance
  - Diversity
  - Maintenance
  - Availability
- Quality Assurance
- Testing and Inspection

Standards

- Pressure Retaining Components
- Electrical Distribution
- Instrumentation and Control
- Civil Structures

Protection

- Seismic Design
- Fire Protection
- Environmental Qualification and Hazard Barriers

Table of Contents

List of Tables

List of Figures

Abbreviations

ADAMS	Agency wide Documents Access and Management Systems
AEA	Atomic Energy Act
ALARA	as low as reasonably achievable
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
BDBA	beyond design basis accident
BDBE	beyond design basis event
CANDU	Canada Deuterium Uranium
CDF	core damage frequency
CFR	*Code of Federal Regulations*
CNSC	Canadian Nuclear Safety Commission
COL	combined license
CP	construction permit
CSA	Canadian Standards Association
DBA	design basis accident
DBE	design basis event
DC	design certification
DEC	design extension condition
EAB	exclusion area boundary
EPA	Environmental Protection Agency
F-C	frequency-consequence
GEH	General Electric Hitachi
I&C	instrumentation and control
IAEA	International Atomic Energy Agency
IDP	integrated decision-making process
LBE	licensing basis event
LERF	large early release frequency
LMP	licensing modernization project
LRF	large release frequency
LWR	light water reactor
MOC	memorandum of cooperation

NCSA	Nuclear Safety and Control Act
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NPP	nuclear power plant
NSRST	nonsafety-related with special treatment
NST	no special treatment
PAG	Protective Action Guide
PIE	postulated initiating events
PRA	probabilistic risk assessment
PSA	probabilistic safety assessment
PSAR	preliminary safety analysis report
QHO	quantitative health objective
REGDOC	regulatory document
RG	regulatory guide
RIDM	risk-informed decision making
RIPB	risk-informed and performance-based
RSF	required safety function
SDA	standard design approval
SR	safety-related
SRP	standard review plan
SSCs	systems, structures, and components
TEDE	total effective dose equivalent

1. Introduction

The Canadian Nuclear Safety Commission (CNSC) and the U.S. Nuclear Regulatory Commission (NRC) signed a Memorandum of Cooperation (MOC) [1] in August of 2019 to further expand their cooperation on activities associated with small modular reactor (SMR)[1] and advanced reactor[2] technologies. This was done under the auspices of the CNSC-NRC Steering Committee (established in August 2017) and to further strengthen the CNSC and NRC commitment to share best practices and experience from design reviews.

Please note that this report maintains terminology and spelling that is consistent with use in the country of origin and no attempt to harmonize these is made in the report (e.g., license and licence; defense and defence, etc.).

Nothing in this report fetters the powers, duties, or discretion of CNSC or NRC designated officers, CNSC or NRC inspectors or the respective Commissions regarding regulatory decisions or taking regulatory action. Nothing in this report is to be construed or interpreted as affecting the jurisdiction and discretion of the CNSC in any assessment of any application for licensing purposes under the Nuclear Safety and Control Act (NSCA) [2], its associated regulations or the CNSC Rules of Procedure. Likewise, nothing in this report is to be construed or interpreted as affecting the jurisdiction and discretion of the NRC in any assessment of any application for licensing purposes under the Atomic Energy Act of 1954, as amended (AEA) [3], its associated regulations and the NRC Management Directives. This report does not involve the issuance of a licence under section 24 of the NSCA or under section 103 of the AEA. The conclusions in this collaborative report are the perspectives of the CNSC and NRC staff.

1.1. Purpose

This report as well as the other joint reports is intended to help applicants and potential applicants understand the relationships between various regulatory requirements in the U.S. and Canada and thereby support their decisions to facilitate gaining approvals of an SMR or advanced reactor design. The reports will also assist the staffs of both the CNSC and NRC when either agency is reviewing applications for a design that is under review or has been reviewed by the other agency.

The outcomes of this collaborative activity are intended to help each jurisdiction leverage information from each other in reviewing SMRs or advanced reactor designs. The activity is also expected to further facilitate the capability to perform joint technical reviews of advanced reactor designs that have been submitted for review in Canada and the United States.

A work plan [4] was approved to document the results of the combined efforts of the CNSC and the NRC with a focus on:

- Identification of key similarities and differences in the safety-significance determination process, the scope of SSCs subject to the process, and the process outcomes.

[1] For this report, the SMR designation refers to water-cooled reactors designed to generate 300 MW (electric) or less with passive design features that provide for enhanced safety relative to currently operating large, water-cooled reactors.
[2] The advanced reactor designation refers to non-LWRs with design features that provide for enhanced safety relative to currently operating large, water-cooled reactors.

- Identification of key similarities and differences in the engineering design rules and specifications applied to each safety class and how this impacts the outcomes.
- Comparison of how each organization applies existing codes and standards and interacts with Standards Development Organizations to verify appropriate codes and standards are being developed, applied, and endorsed.

This report is an interim report that addresses the similarities and differences in the safety classification process and two areas involving application of engineering design rules: reliability assurance and pressure retaining components and supports. The purpose of providing an interim report is to enhance the understanding of the respective staff of the CNSC and NRC on the counterpart nation’s regulatory review activities and requirements in the area of safety classification of SSCs. The interim report also supports continued development of the remaining sections involving application of engineering design rules.

1.2. Background

The objective of safety classification is to identify and classify those SSCs that are needed to protect people and the environment from harmful effects of ionizing radiation, based on their roles in preventing accidents, or limiting the radiological consequences of accidents should they occur.

The general approach is to provide a structure and method for identifying and classifying SSCs important to safety on the basis of their functions and safety significance. Once SSCs are classified, appropriate engineering design rules can be applied to ensure that they are designed, manufactured, constructed, installed, commissioned, operated, tested, inspected, and maintained with sufficient quality to fulfil the functions that they are expected to perform and, ultimately the main safety functions, in accordance with the safety requirements.

Classification is a top-down process that begins with a basic understanding of the plant design, its safety analysis and how the main safety functions will be achieved. Using this information, the functions and design provisions required to fulfil the main safety functions are systematically identified for all plant states, including all modes of normal operation. Using information from safety assessments, such as the analysis of postulated initiating events, the functions are then categorized on the basis of their safety significance. The SSCs belonging to the categorized functions are then identified and classified on the basis of their role in achieving the function. Details on CNSC and US NRC processes and regulatory criteria supporting safety classification are provided in Section 3.

The next section of this report addresses engineering design rules. Engineering design rules include the following attributes considered in design:

- consensus codes and standards
- conservative safety margins
- reliability (e.g., redundancy, diversity, and separation of components)
- equipment qualification
- provisions for inspections, testing, and maintenance
- measures to ensure quality, such as design control measures

The application of engineering design rules was sorted into the following three categories: (1)

programmatic engineering design rules, (2) design of SSCs, and (3) plant-level engineering design rules for hazard protection. These considerations resulted in incorporation of the following specific areas related to engineering design rules into the scope of this report[3]:

- Programmatic engineering design rules:
    - Reliability assurance programs, including maintenance and availability
    - Quality assurance during design and construction
    - Design features to support inservice testing and inspection
- Design of specific structures, systems, and components:
    - Pressure retaining components and supports
    - Electrical distribution
    - Instrumentation and control
    - Civil structures
- Engineering design rules for hazard protections
    - Seismic design rules
    - Fire protection design rules
    - Environmental qualification and hazard barriers

Programmatic engineering design rules are often specified in consensus standards and incorporate features at the construction stage to support long-term reliability of SSCs. Reliability requirements for SSCs are established in the safety analysis process and affect the SSC safety classification and the assignment of engineering design rules. This report also addresses the relationship between classification and the scope of SSCs subject to operational programs, such as maintenance and inservice testing and inspection, intended to maintain reliability.

[3] As discussed in Section 1.1, the interim report will have information on reliability, and pressure retaining components and supports. The final report will have information on engineering design rules for all technical areas listed above.

2. Overview of Regulatory Framework and Safety Concepts

2.1. Regulatory Framework

The regulatory framework for licensing of new reactors in each nation has been established through laws issued by the respective governments. The laws mandate protection of the health and safety of the public, the protection of the environment, and the maintenance of security.

2.1.1. Canada

The NSCA, which became effective in May 2000, establishes the CNSC's mandate to regulate the development, production, and use of nuclear energy in Canada. The CNSC's regulatory framework includes a set of regulations that covers the full extent of the facilities and activities and practices regulated by the CNSC.

The CNSC's regulatory framework program aims to provide regulatory instruments that clearly state CNSC's regulatory expectations, and guidance material. Compliance with the higher-level elements of the NSCA and regulations is required. The CNSC has developed and published Regulatory Documents (REGDOCs) that clearly present expectations for compliance with the NSCA and its associated regulations, and with standards the CNSC has agreed to adopt. Requirements in REGDOCs need to be addressed; but, in certain instances, an applicant or licensee may put forward a case to demonstrate that the intent of a requirement is addressed by other means. Such a case must be demonstrated with supportable evidence. This does not mean that the requirement is waived; rather, it is an indication that the regulatory framework provides flexibility for licensees to propose alternative means of achieving the intent of the requirement. The Commission is ultimately responsible for licensing decisions.

2.1.2. United States

The NRC derives its regulatory authority from the AEA. The AEA directed that regulations be prepared that would protect public health and safety and the common defense and security.

The NRC's regulations are contained in Title 10, "Energy," of the *Code of Federal Regulations* (10 CFR). An applicant or licensee must comply with applicable regulations unless the Commission grants an exemption. The traditional NRC licensing process defined in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," [5] provides a similar sequential licensing process for an optional site preparation authorization, a construction permit (CP), and an operating license (OL). The NRC has also established a licensing process under 10 CFR Part 52 [6], "Licenses, Certifications, and Approvals for Nuclear Power Plants," that provides for conditional approval of sites, standardized reactor designs, and construction and operation of a reactor as a combined license. Each standard design approval (partial design) or standard design certification (complete design excluding certain site-specific elements) could be incorporated by reference in an application for a combined license at any number of sites that could accommodate the design.

The regulations provide for specific exemptions from individual requirements when special circumstances are met, and special circumstances include conditions where compliance is not necessary to meet the underlying purpose of the regulation.

The NRC staff develops and publishes Regulatory Guides (RGs) that describe methods that the NRC staff considers acceptable for use in implementing specific regulations and provide guidance to applicants. Compliance with RGs is not required, and methods that differ from those set forth in the RG are acceptable if they provide a basis for the regulatory findings necessary for the regulatory action. For light water reactors, the NRC staff makes standard review plans available to the public describing the acceptance criteria and methods of evaluation the NRC staff uses to establish compliance with regulations.

Consistent with the Nuclear Energy Innovation and Modernization Act (NEIMA), the NRC has endorsed published risk-informed, technology neutral guidance for the licensing of advanced reactors in Regulatory Guide (RG) 1.233 [7], “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors.” This RG endorses the Licensing Modernization Project (LMP) principles and methodology, as described in Nuclear Energy Institute (NEI) 18-04 [8], Revision 1, “Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development.” LMP is a technology-inclusive, risk-informed, and performance-based approach that can be implemented under the existing NRC regulatory framework with development of an appropriate probabilistic risk analysis (PRA). The LMP may be used in either the 10 CFR Part 50 or Part 52 licensing process, with appropriate supported exemptions. The NRC staff is also proposing that the LMP methodology would be one acceptable way of meeting the proposed 10 CFR Part 53 [9], “Risk-Informed, Technology -Inclusive Regulatory Framework for Commercial Nuclear Plants,” rulemaking specific to this risk-informed, technology-inclusive approach.

2.2. Reactor Facility Licensing

The regulatory framework for the licensing of new reactors by the CNSC and the NRC have both similarities and differences. The CNSC and NRC frameworks provide a flexible approach for licensing of new water-cooled SMRs and advanced reactors.

Section 2, “Overview of Regulatory Processes for New Designs,” of the CNSC-NRC Report: “Technology Inclusive and Risk-Informed Reviews for Advanced Reactors: Comparing the Licensing Modernization Project with the Canadian Regulatory Approach,” (CNSC Approach/LMP Comparison) [10] describes the licensing processes in Canada and the United States for new nuclear reactors. The section provides an overview of pre-application interactions, application interactions, and regulatory safety objectives that are applicable to all new reactors.

2.3. Safety Analysis Process

The process used to classify individual SSCs for importance starts with a safety analysis. In REGDOC 3.6, “Glossary of CNSC Terminology,” (CNSC Glossary) [11], the CNSC defines a safety analysis as a systematic evaluation of the potential hazards that are associated with the conduct of a proposed activity or facility and that considers the effectiveness of preventive measures and strategies in reducing the effects of such hazards. This section introduces the elements of a safety analysis and provides an overview of safety analysis methods and considerations applicable to both regulatory frameworks.

The safety analysis for each proposed reactor begins with the reactor design. The analyst uses the reactor design to identify and characterize radionuclide sources that could be released and barriers that prevent or slow the release of those radionuclides. Figure 3 provides examples of barriers that may be present in the conceptual design, including the fuel matrix, the fuel cladding, the reactor (primary) coolant, the reactor coolant boundary, containment or confinement volume, cleanup systems, and structure. The source term represents the quantity and composition of the radionuclides that may be released from the fuel through one or more barriers as a result of an accident sequence. Leakage of the radionuclides beyond all of the barriers defines the radionuclide release to the environment.

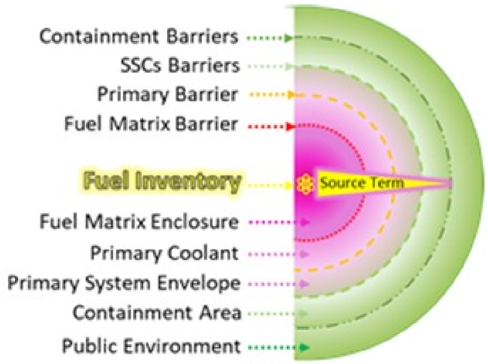


Figure 1: Fission Product Barriers

Source: NRC web page on "Nuclear Power Reactor Source Term" (https://www.nrc.gov/reactors/new-reactors/advanced/nuclear-power-reactor-source-term.html)

Considering the potential events that could challenge the integrity of barriers, the analyst identifies functions necessary to assure that the effectiveness of the barriers would be maintained. This step involves consideration of the barriers themselves, the functions that maintain adequate barrier performance, and supporting SSCs and operator actions. Together, these considerations establish the means of maintaining the fundamental safety functions of confinement of radioactive material, control of nuclear reactivity, and heat removal. The result of this analysis is the identification of SSCs and operator actions that perform safety functions, which are functions that contribute to protecting the barriers, preventing challenges to barrier integrity from developing, or enhancing the effectiveness of barriers in limiting releases of radioactive material.

2.4. Defense-in-Depth Considerations

According to the NRC glossary, defense-in-depth (DID) is:

> *An approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense in depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures.*

The CNSC Glossary provides the following similar definition:

> *A hierarchical deployment of different levels of diverse equipment and procedures to prevent the escalation of anticipated operational occurrences and to maintain the effectiveness of physical barriers placed between a radiation source or radioactive material and workers, members of the public or the environment, in operational states and, for some barriers, in accident conditions.*

The following figure, taken from NRC NUREG/KM-0009 [12], "Historical Review and Observations of Defense-in-Depth," illustrates the concept of layers of defense embodied in this philosophy. This process is consistent with the "levels of defense" concept advanced by the 2005 International Atomic Energy Agency (IAEA) Safety Report Series No. 46 [13], "Assessment of Defense in Depth for Nuclear Power Plants." The intent is to control disturbances during normal operation, control abnormal operating conditions to return to normal operations, maintain more significant events within the design basis of mitigating systems, control the effects of severe plant damage by mitigation of radionuclide releases, and prevent adverse public health and safety impacts from any release through emergency response capabilities.

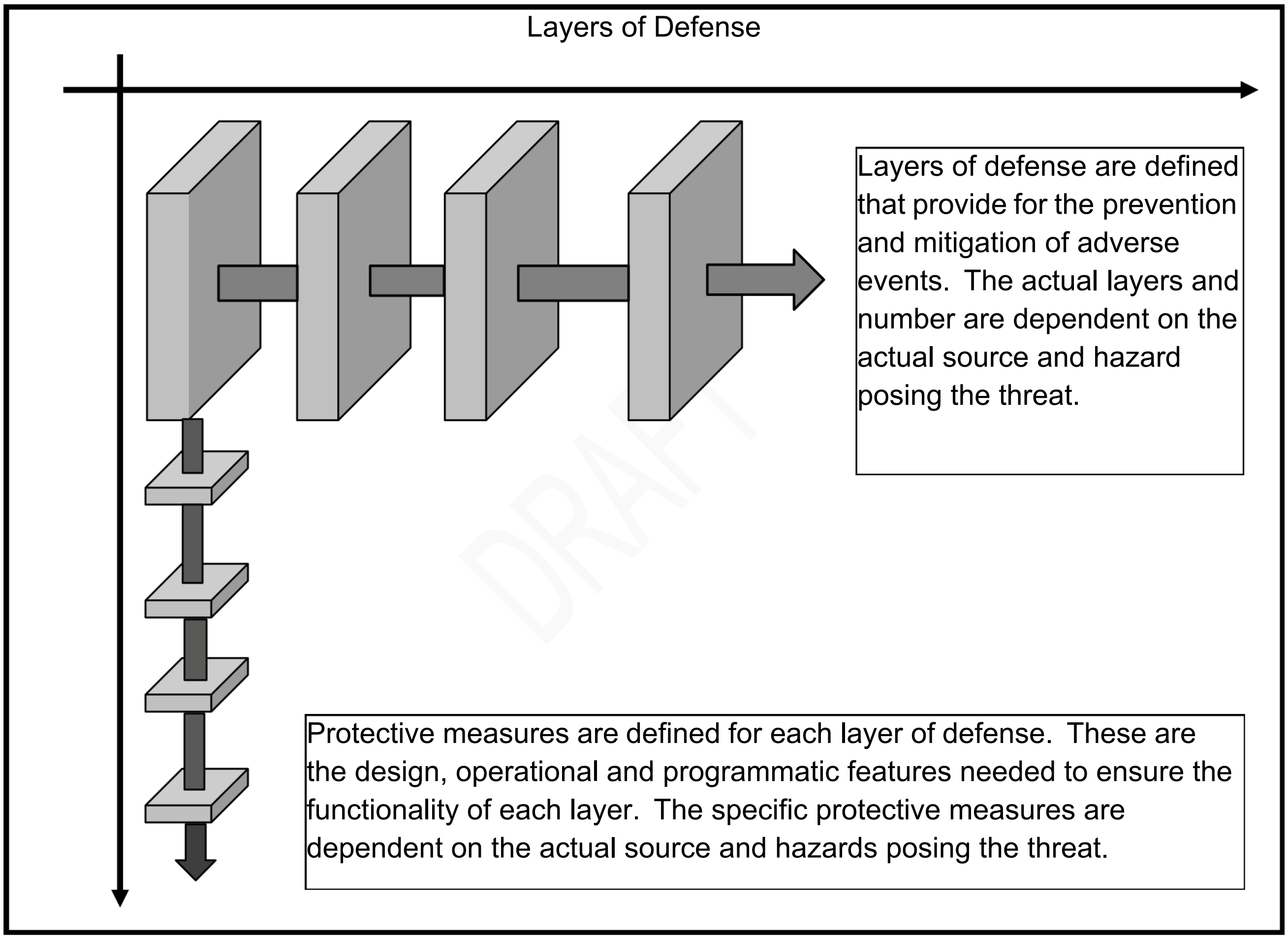


Figure 2: Concept of Defense-in-Depth

3. Licensing Approaches

This section provides a summary of the licensing approaches available under the current regulatory frameworks in Canada and the United States and the relationship of these approaches to safety classification of reactor SSCs.

3.1. CNSC Approach

Section 2.4.1 of the CNSC Approach/LMP Comparison provides a comprehensive discussion of the CNSC regulatory approach, including its evolution to a more risk-informed, technology neutral structure in line with the precepts of the IAEA. This section provides a summary of that information to simplify the application of this report to the safety classification of SSCs.

Under the CNSC regulatory approach, REGDOC 2.5.2 [14], "Design of Reactor Facilities: Nuclear Power Plants," provides requirements and guidance for the licensing of new nuclear power plants in the following areas:

- safety goals and objectives
- safety concepts and management principles applied to the design
- general plant design, including interfacing engineering aspects, plant features, and layout
- design of specific SSCs
- safety analysis

The requirements and guidance of REGDOC 2.5.2 apply directly to the design of new water-cooled nuclear power plants. However, the CNSC recognized the potential for application of the requirements to technologies other than water-cooled reactors and specified that other technologies would be subject to the safety objectives, high-level safety concepts and safety management requirements of REGDOC 2.5.2. The CNSC's strategy for "Readiness to Regulate Advanced Reactor Technologies" [15] describes the following approach:

> CNSC staff consider all relevant guidance when evaluating any proposal submitted. This includes application of the graded approach, and consideration of alternative means of meeting requirements.
>
> The graded approach is a systematic method or process by which elements such as the level of analysis, the depth of documentation, and the scope of actions necessary to comply with requirements are commensurate with:
>
> - the relative risks to health, safety, security, the environment, and the implementation of international obligations to which Canada has agreed
> - the particular characteristics of a nuclear facility or licensed activity
>
> In addition, as outlined in section 11 of REGDOC-2.5.2, the CNSC will consider alternative approaches to requirements of nuclear power plant design when:
>
> 1. the alternative approach would result in an equivalent or superior level of safety
> 2. the application of the requirements in this document conflicts with other

rules or requirements

3. the application of the requirements in this document would not serve the underlying purpose, or is not necessary to achieve the underlying purpose

Any alternative approach shall demonstrate equivalence to the outcomes associated with the use of established requirements.

Safety Analysis:

Applicants complete a deterministic safety analysis consistent with REGDOC 2.4.1 [16], “Deterministic Safety Analysis,” and a probabilistic safety analysis consistent with REGDOC 2.4.2 [17], “Probabilistic Safety Assessment (PSA) for Nuclear Power Plants,” to support the evaluation of the facility against the safety goals and dose acceptance criteria derived from the safety objectives.

In REGDOC 2.4.1, the CNSC describes methods for identifying and grouping initiating events, classifying events by frequency and type, establishing acceptance criteria, and conducting the safety analysis. The objectives of the deterministic analysis related to design are to:

- confirm that the design of a nuclear power plant (NPP) meets design and safety requirements
- derive or confirm operational limits and conditions that are consistent with the design and safety requirements for the NPP
- assist in demonstrating that safety goals are met

The guidance in REGDOC 2.4.1 states that the applicant performs the safety analysis for a set of events that could lead to challenges related to the nuclear power plant’s safety or control functions. These include events caused by SSC failures or human error, as well as human-induced or natural events, and consider credible combinations of events. The applicant should identify the set of events to be considered in safety analysis using a systematic process and by considering:

- reviews of the plant design using such methods as hazard and operability analysis, failure mode and effects analysis, and master logic diagrams
- lists of events developed for safety analysis of other NPPs, as applicable
- analysis of operating experience data for similar plants
- equipment failures, human errors and common-cause events identified iteratively with PSA

The identified events or event combinations are classified based on estimated frequency of occurrence into the following categories:

- Anticipated Operational Occurrences (AOOs) with a frequency of occurrence $\geq 10^{-2}$ per reactor year
- Design Basis Accidents (DBAs) with a frequency of occurrence $\geq 10^{-5}$ and $<10^{-2}$ per reactor year
- Beyond Design Basis Accidents (BDBAs) with a frequency of occurrence $<10^{-5}$ per reactor year

Plant states resulting from a subset of BDBAs, termed Design Extension Conditions (DECs) are considered in the facility design for mitigation on a best estimate basis through addition of complementary design features; the remainder of BDBAs are practically eliminated and not considered in the design. Events within a category are grouped based on similarities in initiating event, related phenomena, or expected plant responses. The analysis of events should consider the principles of DID in establishing the acceptance criteria. One or more event sequences may be bounding with respect to challenges to acceptance criteria, including the dose consequences or the maintenance of essential safety functions. An event sequence with a predicted frequency on the threshold between classifications or with substantial uncertainty in the frequency would be evaluated against acceptance criteria established for the higher frequency classification.

In REGDOC 2.4.2, the CNSC describesthe objectives of the PSA as follows:

- identify the sequences of events and their probabilities, which lead to challenges to fundamental safety functions, loss of integrity of key structures, release of radionuclides into the environment and public health effects
- develop a well-balanced NPP design
- assess the impact of changes to procedures and/or components on the likelihood of core damage

The PSA complements the deterministic safety assessment.

Safety Functions:

In REGDOC 2.5.2, the CNSC identifies that the following fundamental safety functions shall be available during operational states, DBAs and DECs, except where the postulated accident involves a loss of that function:

- control of reactivity
- removal of heat from the fuel
- confinement of radioactive material
- shielding against radiation
- control of operational discharges and hazardous substances, as well as limitation of accidental releases
- monitoring of safety-critical parameters to guide operator actions

Defence-in-Depth:

Section 4.3.1, “Defence-in-depth,” of REGDOC 2.5.2 specifies the application of five levels of DID in the design of nuclear power plants such that a series of measures are established aimed at preventing accidents and ensuring appropriate protection in the event that prevention fails. This structure follows the IAEA recommended approach described in SSR-2/1, “Safety of Nuclear Power Plants: Design” [18] and ISAG Series 10, “Defense in Depth in Nuclear Safety” [19]. Section 6.1, “Application of defence-in-depth,” of REGDOC 2.5.2 includes additional guidance. Table 1 (presented as Table 4 in the CNSC/LMP Comparison report) presents design-related elements of the CNSC DID levels based on information from REGDOC 2.5.2:

Table 1: Application of Design Related Elements of Defence-in-Depth

DID Level	Objective	Essential Means
Level 1	To prevent deviations from normal operation, and to prevent failures of SSCs important to safety	Conservative design High quality construction (e.g., appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication and plant construction, operational experience)
Level 2	To detect and intercept deviations from normal operation, to prevent AOOs from escalating to accident conditions and to return the plant to a state of normal operation	Inherent and engineered design features to minimize or exclude uncontrolled transients to the extent possible
Level 3	To minimize the consequences of accidents, and prevent escalation to beyond design basis accidents	Inherent safety features, Fail-safe design, engineered design features, and procedures that minimize consequences of DBAs
Level 4	To ensure that radioactive releases caused by severe accidents or Design Extension Conditions are kept as low as practicable	Equipment and procedures to manage accidents and mitigate their consequences as far as practicable, Robust containment design, Complementary design features to prevent accident progression and to mitigate the consequences of Design Extension Conditions, Severe accident management procedures
Level 5	To mitigate the radiological consequences of potential releases of radioactive materials that may result from accident conditions	Emergency support facilities, Onsite and offsite emergency response plans

The elements of Levels 2, 3, and 4 should be considered in establishment of acceptance criteria for analyzed event sequences appropriate for the event category. REGDOC 2.5.2 specifies that the levels of defence-in-depth be independent to the extent practicable. The intent of the CNSC DID implementation is to minimize the challenges to physical barriers, prevent their failure if there is a challenge, and minimize the probability of propagation of a failure from one level of defence to the next. If a failure were to occur, the DID approach allows the failure to be detected, and to be compensated for or corrected.

Design Basis Dose Assessment:

In REGDOC 2.5.2, the CNSC specifies that the calculated individual event sequence radiation dose assessment results be less than the committed whole-body dose acceptance criteria for average members of critical groups over the 30 days following the event of 0.5 millisievert (mSv) for AOOs and 20 mSv for DBAs.

Safety Objective Assessment

In REGDOC 2.5.2, the CNSC lists the following qualitative safety goals:

> Individual members of the public shall be provided a level of protection from the consequences of nuclear power plant operation, such that there is no significant additional risk to the life and health of individuals.
>
> Societal risks to life and health from nuclear power plant operation shall be

comparable to or less than the risks of generating electricity by viable competing technologies and shall not significantly add to other societal risks.

Consistent with these CNSC qualitative safety goals, the CNSC specifies the following quantitative safety goals:

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 radioactive material release frequency: The sum of frequencies of all event sequences that can lead to a release to the environment of more than 10^{15} becquerels of iodine-131 shall be less than 10^{-5} per reactor year. A greater release may require temporary evacuation of the local population.

Large release frequency: The sum of frequencies of all event sequences that can lead to a release to the environment of more than 10^{14} becquerels of cesium-137 shall be less than 10^{-6} per reactor year.

Summary of Safety Analysis Acceptance Criteria

Table 2 summarizes analysis methods, DID considerations, and acceptance criteria for AOOs, DBAs, and DEC states for BDBAs

Table 2: CNSC Safety Analysis Acceptance Criteria

Initiating Event Category	AOO	DBA (or AOO with DID Level 2 Failure)	BDBA
SSC Availability	No Single Failure	Single-Failure Affecting Safety System Group	No Single Failure
Analysis Methods	Best Estimate (DID Level 2)	Conservative Analysis or Best Estimate plus Evaluation of Uncertainties (DID Level 3)	Best Estimate (DID Level 4)
Fuel and SSC Limits	Within Specified Acceptable Design Limits; No Unanalyzed Conditions	Within Specified Acceptable Design Limits; No Unanalyzed Conditions	Evaluate Ability to Restore or Maintain Safety Functions
Dose	0.5 millisievert (mSv)	20 mSv	Safety Goals
Consequential Failures	Prevented to the Extent Practicable	Prevented to the Extent Practicable	Avoid Cliff-Edge Effects; Prevent Early Containment Failure

Safety Classification:

In the CNSC framework, the designer/applicant is expected to classify SSCs, as important to safety or not important to safety, using a consistent and clearly defined classification methodology and design, construct, and maintain those SSCs such that their quality and reliability is commensurate with the classification. Beyond establishing SSCs as systems important to safety, the vendor/applicant may propose a graded classification of systems from most important to least important to safety. The number of categories is left to the discretion of the vendor/applicant. All SSCs are identified as either important to safety or not important to safety with safety-significance based on:

- safety function(s) to be performed
- consequence(s) of failure
- probability that the SSC will be called upon to perform the safety function
- the time following a PIE at which the SSC will be called upon to operate, and the expected duration of that operation

In evaluating the consequences of failure, the severity should be based on the consequences of a failure assuming that safety functions assigned to subsequent levels of defence-in-depth remain functional. Those SSCs that provide essential support to frontline SSCs should be assigned to the same safety class as the frontline SSC, and those SSCs performing several safety functions should be assigned to the safety class associated with the function with the highest safety-significance.

The establishment of appropriate engineering design rules is expected to be commensurate with the selected safety class and should be an output of the safety classification process. The CNSC allows the use of a graded approach to quality assurance requirements and other engineering design rules that is commensurate with these safety classifications. Figure 3, which was drawn from IAEA SSG-30 [20], “Safety Classification of Structures, Systems and Components in Nuclear Power Plants,” reflects the CNSC process used to identify important to safety functions, the linkage of the functions to specific SSCs, and the classification of those SSCs.

Assignment of Engineering Design Rules:

Section 7.5 of REGDOC 2.5.2 provides guidance for assigning engineering design rules. The engineering design rules should be determined based on the safety classification and include the following categories, as applicable:

- identified codes and standards
- conservative safety margins
- reliability
- equipment qualification
- provisions for inspections, testing, and maintenance
- management system application (i.e., organizational quality assurance)

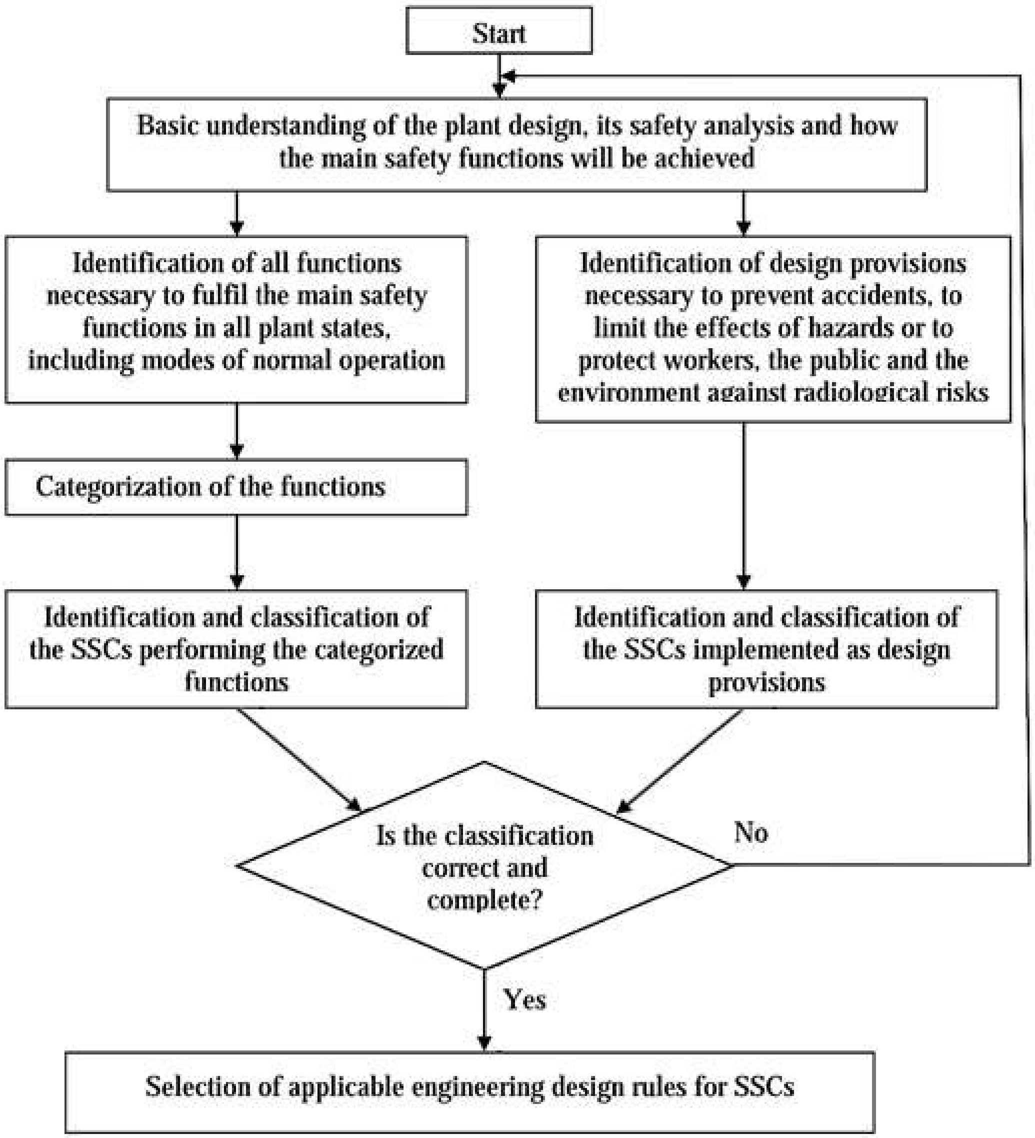


Figure 3: IAEA Safety Classification Flow Diagram

Source: IAEA SSG-30

3.2. NRC Licensing Pathways

NRC regulations establish a variety of regulatory frameworks for commercial nuclear plant licensing, which provides designers and applicants considerable flexibility while also ensuring an acceptable level of safety. This report focuses on licensing approaches pursuant to 10 CFR Part 50 or 10 CFR Part 52. These licensing frameworks reflect significant licensing and operational experience related to light water reactors (LWRs). Applicants for LWR certificates, approvals, permits, or licenses under 10 CFR Part 50 or 10 CFR Part 52 must include an evaluation of the facility against the Standard Review Plan (SRP) [21] in effect 6 months prior to docketing of the application. The SRP provides comprehensive guidance for completion of the safety analysis report, including identification of specific groups of initiating events and the associated safety analysis.

The NRC staff has focused on developing technology-inclusive guidance rather than SRPs for non-LWRs due to wide variation among potential non-LWR designs. The NRC staff has developed a draft analysis of the applicability of NRC regulations to advanced reactors [22] indicating that 10 CFR Part 50 and Part 52 would support licensing of non-LWR advanced reactors with appropriately supported exemptions. The NRC staff has also developed a draft white paper associated with the Advanced Reactor Content of Application Project (ARCAP), titled “Guidance for Performing the Review of a Technology-Inclusive Advanced Reactor Application - Review Roadmap” [23].

The draft guidance that the NRC staff is developing in Draft Guide (DG) 1413 [24], “Technology-Inclusive Identification of Licensing Events for Commercial Nuclear Plants,” provides a technology-inclusive approach for both LWR and non-LWR applicants for performing a comprehensive and systematic search for initiating events and delineating a comprehensive set of licensing event sequences without preconceptions or reliance on predefined lists. Table 3 shows licensing pathways applicable to water-cooled SMRs and advanced reactors within the 10 CFR Part 50 or 10 CFR Part 52 frameworks, with or without application of the LMP, and is adapted from DG 1413.

Event categorization differs between the traditional licensing approach and the LMP. Under the traditional licensing approach, design basis events (DBEs) are defined as conditions of normal operation, including anticipated operational occurrences (AOOs), design basis accidents (DBAs), external events, and natural phenomena. An AOO is an event expected to occur one or more times over the life of the facility, and a DBA is a specific event sequence that bounds similar events with respect to challenging an essential safety function or an event sequence used to assess dose consequences for a class of events. Beyond design basis events (BDBEs) are specified events included in the regulations that are considered to ensure the principles of DID are maintained. Under the LMP, events considered in the licensing of the facility are collectively referred to as licensing basis events (LBEs) rather than DBEs, and frequency-based definitions apply to AOOs, DBEs, and BDBEs. The LMP DBAs are derived from the DBEs by evaluating the event with only safety-related SSCs available for mitigation.

Table 3: NRC Licensing Pathways and Event Categorization

<table>
<tr><th>Regulation and Application Type</th><th>Reactor Type</th><th>Use of LMP[a]</th><th>Licensing Event Categories</th><th>Risk Evaluation</th></tr>
<tr><td>Part 50
Construction Permit (CP), Operating License (OL)</td><td rowspan="2">LWR</td><td rowspan="2">n/a</td><td rowspan="4">DBEs[b] - this term is used in the § 50.2 definition of safety-related SSCs; § 50.49 identifies four subcategories of DBEs as follows:
AOOs
DBAs (i.e., postulated accidents)
External events
Natural phenomena
BDBEs
Anticipated Transients Without Scram (ATWS)
Station Blackout (SBO)</td><td>Evaluation against SRP Chapter 19[c]</td></tr>
<tr><td>Part 52
Design Certification (DC), Standard Design Approval (SDA), Manufacturing License (ML), Combined Operating License (COL)</td><td>PRA required</td></tr>
<tr><td>Part 50
CP, OL</td><td rowspan="2">non-LWR</td><td rowspan="2">no</td><td>Not required[c]</td></tr>
<tr><td>Part 52
DC, SDA, ML, COL</td><td>PRA required</td></tr>
<tr><td>Part 50
CP, OL</td><td rowspan="2">non-LWR</td><td rowspan="2">yes</td><td rowspan="2">Licensing events are collectively referred to as licensing basis events (LBEs), which include the following categories:
AOOs
DBEs
BDBEs
DBAs</td><td>PRA necessary for LMP[a]</td></tr>
<tr><td>Part 52
DC, SDA, ML, COL</td><td>PRA required</td></tr>
<tr><td colspan="5">[a] The Licensing Modernization Project (LMP) guidance, which is provided in NEI 18-04, Rev. 1 and endorsed in RG 1.233, provides a voluntary technology-inclusive approach to LBE selection for non-LWRs licensed under Parts 50 or 52 and includes an expanded role for PRA beyond that currently required.
[b] Although 10 CFR Parts 50 and 52 include normal operation in the design basis, the risk evaluation focuses on departures from normal operation.
[c] SECY-22-0052 (ML21159A055) describes NRC proposed changes to the regulations in 10 CFR Part 50 and 10 CFR Part 52 to align reactor licensing processes and incorporate lessons learned from new reactor licensing into the regulations. The NRC is proposing to add new regulations, 50.34(a)(14) and 50.34(b)(14), to require CP and OL applicants to submit a description of the plant-specific probabilistic risk assessment (PRA) and its results. Chapter 19 the SRP addresses the use of PRA and evaluation of severe accidents.</td></tr>
</table>

The licensing approach under 10 CFR Part 52 has been used for the NuScale SMR design certification through approval of a final rule to be incorporated into a new Appendix to 10 CFR Part 52 [25]. Other commercial power SMRs and advanced reactors are in the preapplication phase.

3.2.1. Traditional NRC Licensing Approach

For the traditional NRC licensing approaches under either 10 CFR Part 50 or 10 CFR Part 52, the regulations require a safety analysis report supported by deterministic analyses and other information.

Safety Analysis:

The safety analysis includes the following design-related elements:

- A safety assessment of the site and facility, including:
    - the nature and inventory of contained radioactive materials
    - the extent of application of engineering standards to facility design
    - safety features and those barriers that must be breeched as a result of an accident to release radioactive material
    - an analysis of a postulated fission product release to evaluate the offsite radiological consequences
- An assessment of the design of the facility, including:
    - the principal design criteria (PDC)
    - the relationship of the facility design bases to the PDC
    - an analysis and evaluation of the design and performance of SSCs to assess the risk to public health and safety

Applicants must complete an assessment of the design and performance of SSCs. This assessment establishes that the necessary SSC performance characteristics have been incorporated into the design to assure safety functions are accomplished, considering both the site and the reactor design characteristics. Chapter 15 of the SRP provides guidance for identification and classification of PIEs and evaluation of the performance of LWR SSCs. The guidance provided in DG-1413 supports a systematic method for identification of PIEs for all reactor types,

The safety analysis guidance specifies conservative analysis of both AOOs and DBAs, which are classified based on the initiating event frequency. The guidance calls for verification that only safety-related systems and components were specified for mitigation and the analysis has considered the effect of single active failures of those systems and components. Components that are not safety-related may continue in operation if unaffected by the initiating event and, on a case-by-case basis with appropriate technical justification, may be specified for mitigation.

The PDC provide deterministic criteria for evaluating the overall design of the facility and the performance of SSCs. The General Design Criteria (GDC) of Appendix A to 10 CFR Part 50 establish the minimum requirements for PDC for water-cooled nuclear power plants and provide guidance to applicants for PDC for other types of nuclear power units. The NRC staff issued RG 1.232 [26], “Developing Principal Design Criteria for Non-Light Water Reactors,” to provide specific guidance for PDCs for non-LWRs. Deterministic design criteria included in the PDC provide assurance that safety functions are adequately maintained following postulated events.

The GDC include acceptance criteria relevant to both AOOs and DBAs. For example, Criterion 20, “Protection System Functions,” states that the protection system shall be designed:

1. to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences; and

2. to sense accident conditions and to initiate the operation of systems and components important to safety.

Additionally, Criterion 29, “Protection against Anticipated Operational Occurrences,” states that the protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences. The advanced reactor design criteria in RG 1.232 corresponding to GDC 20 and 29 are identical to the GDC.

Specific deterministic regulations apply to LWRs. For example, the regulations in 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors,” specify fuel and core acceptance criteria for hypothetical loss of coolant accidents.

Probabilistic Analysis:

Commission policy supports the use of probabilistic analysis methods in all regulatory areas. In August 1995, the NRC issued a final Commission policy statement on the use of PRA methods in nuclear regulatory activities, titled “Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement” [27]. The statement adopted, in part, the following policy:

- The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC’s deterministic approach and supports the NRC’s traditional defense-in-depth philosophy.
- PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, RGs, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109, “Backfitting” (Ref. 23). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. This policy intends compliance with existing rules and regulations unless these rules and regulations are revised.

Consistent with this policy and as indicated in Table 3, the NRC has developed a draft proposed rulemaking that would modify regulations to provide consistent requirements to complete a PRA across the 10 CFR Part 50 and 52 licensing pathways. Currently, applicants for DCs, COLs, SDAs, and MLs under 10 CFR Part 52 must develop probabilistic risk assessments (PRAs) to support the applications. Per policy, applicants under 10 CFR Part 50 are also expected to develop PRAs[4], and the NRC has initiated rulemaking to align the Part 50 regulations with Part 52. Currently, applicants for LWR CPs or OLs under 10 CFR Part 50 must include an evaluation of the facility against the SRP, and Section 19.0 of the SRP addresses PRA and severe accident evaluation methods for new LWRs.

The traditional NRC approach is risk-informed through the quantitative and qualitative consideration of risk in the development of the regulations. Compliance with the regulations presumptively provides reasonable assurance that the standard of adequate protection of public health and safety has been met. The design-specific PRA identifies risk insights, identifies any severe accident vulnerabilities, and ensures that the QHOs are met.

Safety Functions

For the NRC traditional licensing approach, safety functions are identified through the PDC established for the specific reactor design. For LWRs, the prescribed GDC identify important safety functions through consideration of single failures or other measures enhancing redundancy or diversity. The GDC include such measures for performance of the following safety functions: electric power, protective system actuation, reactivity control, residual heat removal, emergency core cooling, containment heat removal, containment atmosphere cleanup, safety equipment cooling, and containment isolation.

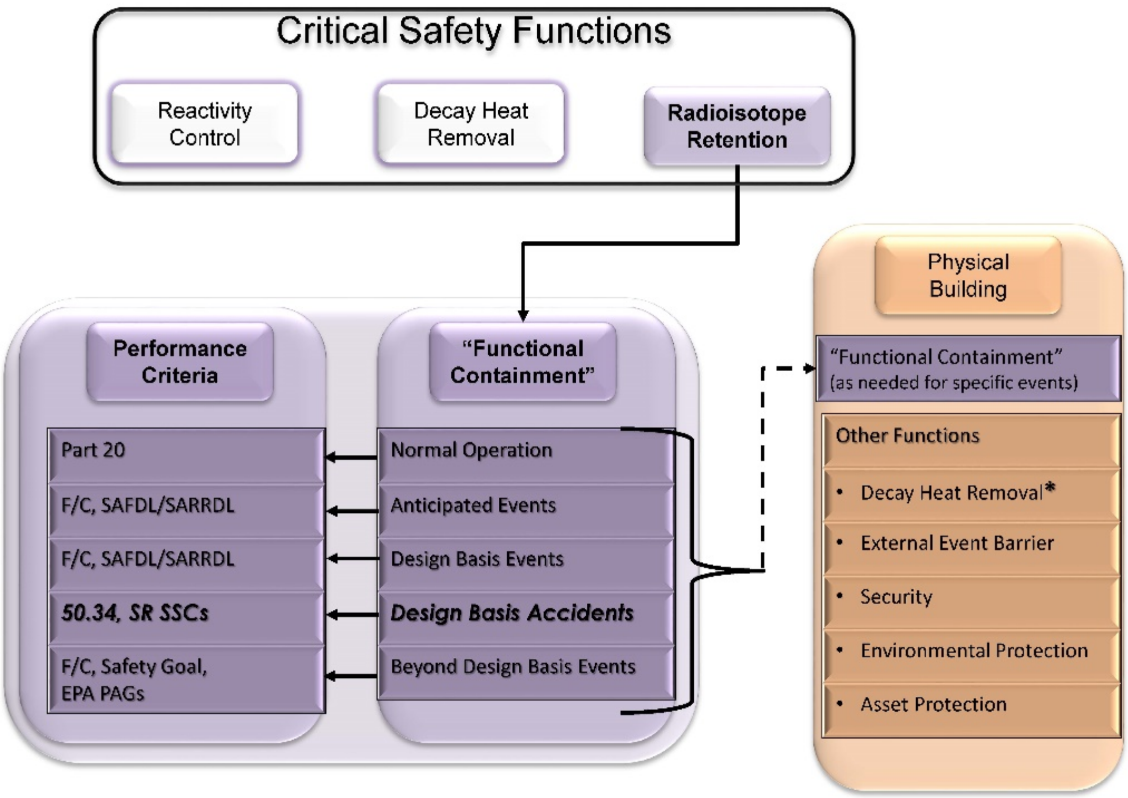
For non-LWRs, the guidance for developing proposed PDC for advanced reactor types included in RG 1.232 considers single failures for similar safety functions, with the exception that the containment-related functions do not consider single failures when the conceptual design credits functional containment rather than a single structural containment boundary. This consideration of single failures in systems performing safety functions helps ensure reliability of those functions and reflects consideration of those functions as particularly safety significant.

The NRC addressed the concept of functional containment and necessary performance in SECY 18-0096 [28], “Functional Containment Performance Criteria for Non-Light-Water-Reactors.” In the case of a functional containment, the radioisotope retention function of a low-leakage structure is supplemented or replaced by multiple barriers or components providing radionuclide retention. Instead of performance being based on prevention of leakage, the acceptance criteria would be established for each event classification to meet specified frequency-consequence targets (F/C), specified acceptable fuel design limits (SAFDL), or specified acceptable radionuclide retention limits (SARRDL) . This correlation of event classification to functional containment acceptance criteria is depicted in Figure 4, which was presented in SECY 180-0096. For BDBEs, the performance would be assessed against the safety goals.

---

[4] NRC SECY 15-002, “Proposed Updates of Licensing Policies Rules, and Guidance for Future New Reactor Applications,” and associated SRM (ADAMS Accession Nos. ML13277A420 and ML15266A023, respectively) approved initiation of rulemaking to align Part 50 requirements with Part 52, including submittal of PRA information with new reactor applications. Proposed rulemaking under Regulation Identifier Number (RIN) 3150-AI66 requested public comments on consistency in new reactor licensing reviews (86 FR 7513), and the Commission is considering specific regulatory changes proposed in SECY 22-052 (ADAMS Accession No. ML21159A055).

Figure 4: Functional Containment Performance Criteria



Defense-in-Depth:

The NRC traditional licensing process has established DID measures through the regulations of 10 CFR Part 50 and Part 52. These measures include:

- the PDC, which provide for high quality and conservative design to maintain reliable operation under normal conditions and prevent site disturbances, plant transients, and accidents from becoming more severe
- the emergency core cooling requirements, reactor vessel fracture toughness requirements, combustible gas control requirements, and offsite dose consequence performance requirements that prevent design basis accidents from becoming more severe and provide radionuclide release mitigation if the accident is severe[5]
- the special event regulations (e.g., regulations addressing fire protection, anticipated transient without scram, loss of all alternating current power, and extensive damage mitigation requirements)[6] that manage specific conditions beyond the facility design basis

[5] These requirements include: 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors"; 10 CFR 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events"; and 10 CFR 50.67, "Accident Source Term".
[6] The special event regulations referenced include: 10 CFR 50.48, "Fire Protection"; 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"; 10 CFR

- the emergency planning regulations (i.e., 10 CFR 50.47, “Emergency Plans,” and Appendix E to 10 CFR Part 50, “Emergency Planning and Preparedness for Production and Utilization Facilities”) to help ensure public health and safety if, despite the other requirements, an event progresses to a large radionuclide release

Design Basis Accident Dose Assessment:

To assess barrier performance, the regulations require an evaluation of a major hypothetical fission product release into the containment. This release has historically been based on a prescribed release which effectively occurs due to the presumed failure of two fission product barriers in a large LWR, the fuel cladding and the reactor coolant pressure boundary. The regulation requires evaluation of the release that would progress to the environment through containment leakage paths. The dose reference value in this scenario is 250mSv (25 rem) total effective dose equivalent (TEDE) to both (1) an individual at the exclusion area boundary for two hours and (2) an individual at the low population zone boundary for the entire period of the fission product release passage. However, this dose reference value applies to an event that postulates a large release to containment without identifying a mechanism for that result (i.e., a conservative analysis that disregards the design basis of the emergency core cooling system to mitigate loss of coolant events without major fuel damage) to test containment performance. For non-LWR reactors crediting a functional containment, the applicant could propose consideration of attributes that serve as effective barriers to radionuclide release related to fuel design, inherent safety features, and other design elements that contribute to the retention of radionuclides, in addition to consideration of the radionuclide inventory present during operation at the proposed power level. Such applications may need an exemption from the regulatory requirements. Other DBEs with postulated releases are evaluated to dose criteria that are a fraction (10 to 25 percent, depending on event frequency) of the above dose criteria. Normal operational releases and AOOs are evaluated against the criteria of 10 CFR Part 20 [29], “Standards for Protection Against Radiation.”

Safety Objective Assessment

The traditional performance goals include the requirements defined in 10 CFR 50.40, “Common Standards,” which states that in issuing a CP or OL under 10 CFR Part 50 or an ESP, COL, or ML under Part 52, the Commission will be guided, in part, by:

- reasonable assurance of compliance with the regulations of 10 CFR Part 50
- adequate protection of the public health and safety.

Among the regulations are the requirements that applicants for certificates, approvals, permits, or licenses under 10 CFR Part 50 or 10 CFR Part 52 provide, in part, the following information:

> *An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents*

50.62, “Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants”; 10 CFR 50.63, “Loss of All Alternating Current Power”; and 10 CFR 50.155, “Mitigation of Beyond-Design-Basis Events”.

and the mitigation of the consequences of accidents.

The deterministic safety analysis structure under the NRC traditional framework provides the analysis and evaluation necessary to address these regulatory requirements. The transient events anticipated during the life of the facility are represented by the AOOs considered within the design basis. The adequacy of SSCs provided for prevention of accidents include those SSCs that provide the fundamental safety functions during normal operating conditions, including the effects of AOOs. Other SSCs provide for mitigation of the consequences of accidents by fulfilling fundamental safety functions through alternate means following accidents.

The acceptance criteria established by the plant-specific PDC and applicable regulations address the integrity of barriers to radioactive material release; conditions indicative of safe shutdown, including reactivity control and heat removal; and mitigation of consequences to protect public health and safety. The acceptance criteria also implicitly address DID principles through the prevention of accidents, providing alternate means of maintaining fundamental safety functions under accident conditions, and mitigating the consequences of accidents when fundamental safety functions are not maintained. The overall safety of this approach may be confirmed by a probabilistic risk assessment.

The NRC established qualitative safety goals and quantitative objectives to gauge achievement of the safety goals, which are contained in its Reactor Safety Goal Policy Statement [30]. The NRC staff has established the following radiation exposure guideline values to meet its Quantitative Health Objectives (QHOs) for early or latent health effects:

1. The average individual risk of early fatality within 1.6 kilometers (1 mile) of the exclusion area boundary from all reactor accidents shall not exceed 5 x 10^{-7}/plant-year to ensure that the plant meets the NRC safety goal quantitative health objective for early fatality risk.

2. The average individual risk of latent cancer fatalities within 16 kilometers (10 miles) of the exclusion area boundary from all reactor accidents shall not exceed 2 x 10^{-6}/plant-year to ensure that the plant meets the NRC safety goal quantitative health objective for latent cancer fatality risk.

The NRC traditional approach uses probabilistic analyses to confirm the NRC QHOs have been met.

Summary of Safety Analysis Acceptance Criteria

Table 4 presents a summary of safety analysis acceptance criteria presented in Chapter 15 of the SRP for water-cooled reactors. These acceptance criteria reflect the GDC of Appendix A to 10 CFR Part 50, compliance with appliable regulations, consistency with the 10 CFR 50.2 definition of safety-related SSCs, and consideration of DID principals. However, these acceptance criteria were developed to work within a deterministic framework. Consistent with the NRC Policy Statement on the use of PRA methods, applicants may propose alternate acceptance criteria that maintain compliance with NRC regulations and are appropriately supported by a risk-informed evaluation.

Table 4: NRC Traditional AOO and DBA Analysis Acceptance Criteria

Initiating Event Category	AOO	DBA
SSC Availability	Safety-Related SSCs with Single Failure; with and without Offsite Power; limited credit for other SSCs with Technical Justification	Safety-Related SSCs with Single Failure; with and without Offsite Power
Pressure Boundary	Within 110% of Design	Within Acceptable Design Limits
Fuel	Within Specified Acceptable Fuel Design Limits	Cladding Failure if Specified Acceptable Fuel Design Limit Exceeded
Dose	10 CFR Part 20	Accident Dose Limit (25 Rem TEDE) or Small Fraction of Limit
Consequential Failures	No Escalation without other Independent Faults	No Consequential Failures of SSCs Necessary to Mitigate Fault
Loss of Coolant Accident	Not Applicable	10 CFR 50.46 Criteria

Safety Classification:

The traditional licensing paths using 10 CFR Part 50 or 10 CFR Part 52, incorporate deterministic criteria to classify SSCs as safety related. In addition, regulations, guidance, and policy establish an important to safety class of SSCs, which encompasses the safety-related SSCs, but also includes SSCs defined by other deterministic or risk-informed criteria.

For applicants for power reactor licenses under Part 50 or 52, the NRC has defined the term "Safety-Related" in 10 CFR 50.2, "Definitions," in the following manner:

> *Safety-related structures, systems and components* means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:
>
> - The integrity of the reactor coolant pressure boundary;
> - The capability to shut down the reactor and maintain it in a safe shutdown condition; or
> - The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable

guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

The functions identified here can be related to the fundamental safety functions of confinement of radioactive material, controlling nuclear reactivity, and removing heat. The first function of assuring the integrity of the reactor coolant pressure boundary primarily relates to maintaining barriers to release of radioactive material, since that is a function of the reactor coolant pressure boundary in an LWR. Although the other functions related to reactivity control and heat removal also help ensure the integrity of the pressure boundary, these functions are more closely associated with shutting down the reactor and maintaining it in a safe shutdown condition. The capability to prevent or mitigate the consequences of accidents which could result in substantial offsite exposures addresses the SSCs that perform one of the fundamental safety functions or provide essential support functions to those SSCs. For non-LWRs that lack a reactor coolant pressure boundary performing a comparable confinement function, the NRC staff expects that applicants/designers would seek an exemption to redefine the term “safety-related structures, systems and components” for the specific reactor design.

The NRC regulations provide for a voluntary risk-informed classification process that considers the concept of DID in 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors.” The NRC provided guidance for implementation of this regulation in RG 1.201 [31], “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety-Significance.” This regulation and associated guidance provide a risk-informed method to reduce the scope of engineering design rules and other regulations applied to SSCs that perform low safety significant functions. Implementation of this risk-informed classification process requires development of a PRA and an integrated process to characterize SSC importance that considers DID concepts. An applicant for an LWR CP or OL under 10 CFR Part 50 or an applicant for an LWR SDA, COL, or ML under 10 CFR Part 52 may implement this classification method provided that the applicant includes the required information in its application and the NRC approves implementation.

Although several NRC regulations use the term SSCs important to safety, that term has not been formally defined in regulations[7]. The introduction to the GDC in Appendix A to 10 CFR Part 50 describes important to safety in the following manner:

> *The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.*

Thus, important to safety SSCs includes safety-related SSCs, SSCs that are not safety-related but could affect the performance of safety functions, and SSCs that are not safety-related but perform a function important to DID.

The NRC has identified certain functions important to safety for advanced passive LWRs that

[7] NRC SRM-SECY-21-0112, "Denial of Petition for Rulemaking on Determining which Structures, Systems, Components and Functions are Important to Safety (PRM-50-112;NRC-2015-0213)," describes the NRC basis for not adding a definition of “important to safety” to 10 CFR 50.2 (ADAMS Accession No. ML22026A409).

are also applicable to SMRs. SRP Section 19.3, “Regulatory Treatment of Nonsafety Systems for Passive Advanced Light Water Reactors,’, Revision 0, June 2014 (ADAMS Accession No. ML14035A149) provides guidance on identifying nonsafety-related SSCs that perform risk-significant functions in a passive plant design and are candidates for regulatory oversight.

The use of the terms “nonsafety-related” or “not safety-related” mean only that the associated SSCs do not perform any of the functions identified under the definition for safety-related in 10 CFR 50.2. Risk- or safety-significant equipment that does not meet the definition of safety-related would be considered important to safety.

Assignment of Engineering Design Rules:

The GDC presented in Appendix A to 10 CFR Part 50 contains the minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission and many design criteria are generally applicable to other types of nuclear power units. GDC 1, “Quality Standards and Records,” is applicable to SMRs and advanced reactors, and requires, in part, that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Where generally recognized codes and standards are used, GDC 1 provides that they be identified and evaluated to determine their applicability, adequacy, and sufficiency and be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function. This provision supports a graded application of engineering design rules based on the significance of the safety function performed by the SSC.

In addition to the use of the term “important to safety” to describe applicability of the GDC in Appendix A to 10 CFR Part 50, requirements that apply to equipment described in the regulation as “important to safety” include environmental qualification of important to safety electrical equipment in 10 CFR 50.49 and protection of important to safety equipment from fire in 10 CFR 50.48.

Several regulations in 10 CFR Part 50 use the term “safety-related” to require application of specific engineering design rules to SSCs satisfying that definition. These requirements include:

- environmental qualification of electrical equipment, as required by 10 CFR 50.49
- inservice testing and inspection of safety-related LWR pressure vessels, piping, pumps and valves, and their supports (including access), as required by 10 CFR 50.55a
- monitoring the effectiveness of maintenance, as required by 10 CFR 50.65
- quality assurance, as required by 10 CFR 50.34 and Appendix B to10 CFR Part 50, for activities affecting the safety-related functions of SSCs that prevent or mitigate the consequences of postulated accidents
- earthquake engineering design, as required by 10 CFR 50.34 and Appendix S to10 CFR Part 50

The NRC has developed a policy for nonsafety-related SSCs that perform risk-significant functions in LWRs with passive safety systems in NRC SECY-95-132 [32], “Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems (RTNSS) in Passive Plant Designs.” Guidance in Chapters 17 and 19 of the SRP specifies the application of select engineering design rules to those components that perform the following risk significant functions:

A. SSC functions relied on to meet beyond design basis deterministic NRC performance requirements.

B. SSC functions relied on to ensure long-term safety (the period beginning 72 hours after a design basis event and lasting the following 4 days) and to address seismic events.

C. SSC functions relied on during power-operating and shutdown conditions to meet the Commission goals of a CDF of less than $1x10^{-4}$ each reactor year and a LRF of less than $1x10^{-6}$ each reactor year.

D. SSC functions needed to meet the containment performance goal, including containment bypass, during severe accidents.

E. SSC functions relied on to prevent significant adverse systems interactions between passive safety systems and active nonsafety SSCs.

The RTNSS policy includes establishment of appropriate levels of reliability and availability through the Reliability Assurance Process discussed in Section 6.1 of this report

3.2.2. NRC Risk-Informed, Technology-Inclusive Approach (LMP)

Section 2.4.2 of the CNSC Approach/LMP Comparison provides a more detailed discussion of the LMP. This section provides a summary of that information to simplify the application of this report to the safety classification of SSCs.

The NRC staff issued RG 1.233 to endorse the principles and methodology in NEI 18-04 as one acceptable method for informing the licensing basis and determining the appropriate scope and level of detail for parts of applications for licenses, certifications, and approvals for non-LWRs. Applicants may use the guidance to inform the content of applications for non-LWR applicants applying for permits, licenses, certifications, and approvals under 10 CFR Part 50 or 10 CFR Part 52. In order to implement the risk-informed, technology-inclusive NRC licensing process endorsed by this RG (referred to as the LMP), development of a PRA that includes evaluation of dose consequences to identified populations is necessary. The LMP methodology includes the following processes:

- Systematic definition, categorization, and evaluation of event sequences for selection of licensing basis events (LBEs)[8], which include AOOs (mean frequency $\geq 10^{-2}$ per plant year), DBEs (mean frequency $\geq 10^{-4}$ and $<10^{-2}$ per plant year), DBAs, and BDBEs (mean frequency $\geq 5 \times 10^{-7}$ and $<10^{-4}$ per plant year)

- Systematic safety classification of SSCs, development of SSC performance requirements, and selection of engineering design rules

- Evaluation of DID adequacy

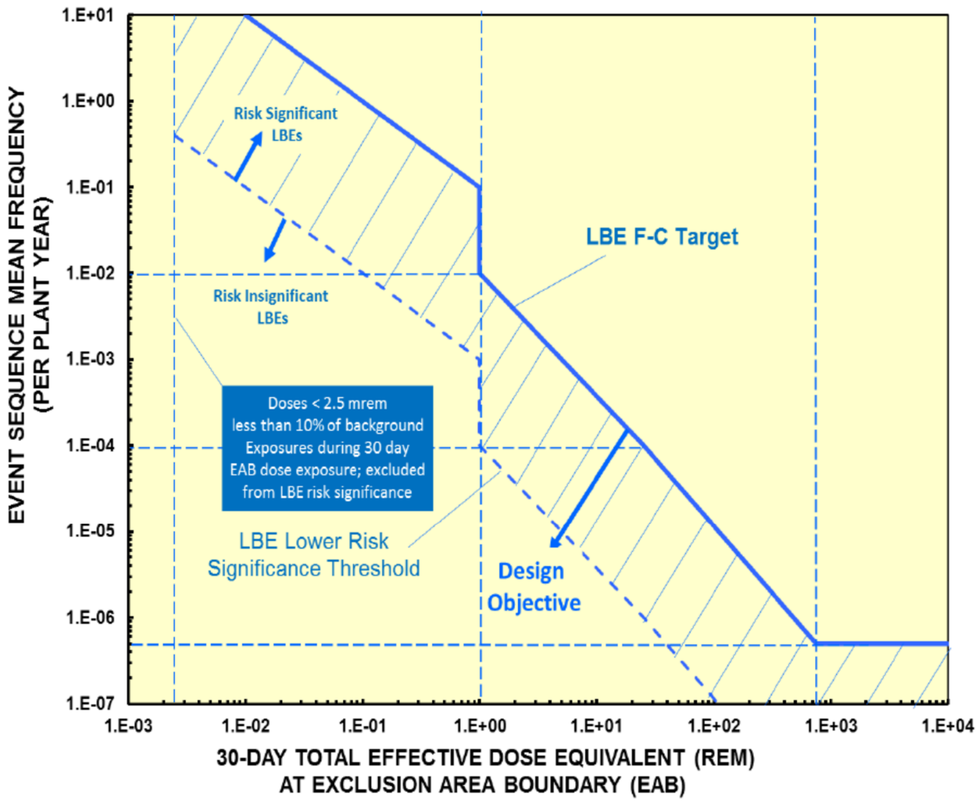
It is anticipated that non-LWR applicants using the LMP with Part 50 or Part 52 regulations would request exemptions from some LWR-oriented requirements, including the definition of safety-related SSCs in 10 CFR 50.2, to accommodate non-LWR designs.

Safety Analysis:

[8] LBEs are the entire collection of event sequences considered in the design and licensing basis of the plant, which may include one or more reactor modules. LBEs include AOOs, DBEs, BDBEs, and DBAs.

The process for evaluation of LBEs involves comparison of individual event sequence risk and cumulative integrated risk against performance targets. This comparison involves early introduction of a PRA into the design process in combination with deterministic evaluations to establish SSC performance during LBEs. The early introduction of PRA facilitates an iterative process by the designer to incorporate risk-informed design decisions related to SSC performance and reliability. The LMP process individually compares event sequence families, with consideration of uncertainty, against a frequency-consequence (F-C) target curve. Collectively, the LMP process compares the integrated risk of all LBEs against a cumulative risk target equal to the annual exposure limits to members of the public in 10 CFR Part 20 and the integrated risk of all LBEs against cumulative risk target values derived from the NRC QHOs. Figure 5 (from NEI 18-04) depicts the F-C targets for LBEs based on overall plant frequency of a specific class of LBEs (AOOs, DBEs, and BDBEs) compared to the 30-day post-accident committed total effective dose equivalent at the exclusion area boundary.[9]

Figure 5: Licensing Basis Event F-C Target Curve



The LMP includes a process for identification of safety functions that supports development of design criteria for SSCs. The LMP uses an evaluation of DBEs and BDBEs to establish the reactor-specific required safety functions (RSFs), which are those functions necessary to ensure the F-C targets are met. These RSFs inform development of design-appropriate PDC, considering the Advanced Reactor Design Criteria (ARDCs) presented in Appendix A of RG

[9] *Exclusion area* means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. The complete definition is provided in 10 CFR 50.2.

1.232. The LMP also includes a risk-informed evaluation of DID, which may identify additional SSCs that perform safety-significant functions. Because the LMP for advanced reactors considers combinations of inherent, passive, and active design features to accomplish RSFs and includes a full risk-informed assessment of event sequences and DID, it obviates the need to apply the single failure criterion as specified in the ARDCs.

Safety Functions:

The LMP process begins with the following technology inclusive fundamental safety functions defined by an IAEA Technical Report [33]:

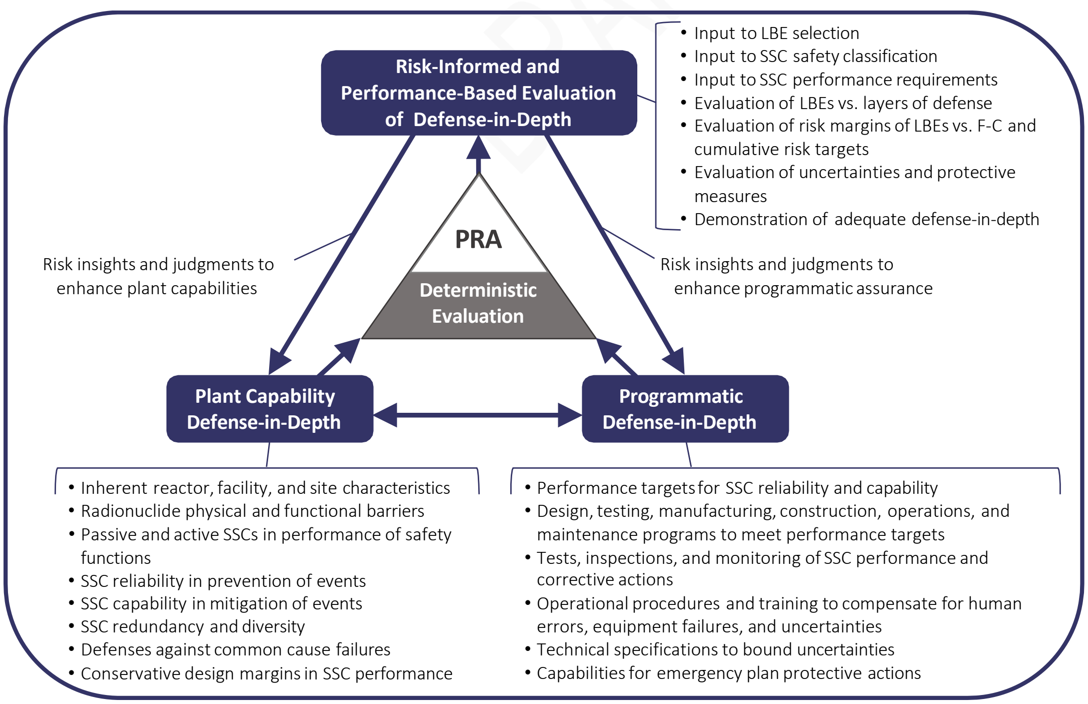
- control of the reactor power
- removal of heat from the fuel
- confinement of radioactive material

From these functions, the designer considers the unique characteristics of the conceptual reactor design to develop the reactor technology-specific required safety functions.

Defense-in-Depth Considerations:

The LMP includes a specific process for consideration of DID principles in the classification of SSCs. Figure 6 is from NEI 18-04 and shows this process:

Figure 6: LMP Framework for Establishing DID Adequacy



The relationship of this LMP process to design is shown in the factors that contribute to plant capability and how the risk-informed and performance-based evaluation provides an input to SSC classification and performance requirements.

Design Basis Accident Dose Assessment

The LMP includes other elements that address conformance with NRC regulatory requirements, including requirements for evaluation of very unlikely events, which are addressed under current requirements by the need to assume a “major accident” with appreciable quantities of fission products released to containment. For the evaluations under the LMP methodology, the applicant selects a group of SSCs that are capable of performing all RSFs necessary to meet the F-C targets for all DBEs and BDBEs with high consequences. This group of SSCs is classified as safety-related. The applicant deterministically evaluates a set of DBAs derived from the DBEs to ensure the dose reference values for a major hypothetical fission product release under 10 CFR Part 50 or Part 52 would be met using only the SSCs classified as safety-related and assuming all other SSCs are not available. Applicants using the LMP may need to request an exemption from the regulations if the most severe DBA does not involve the equivalent of significant core damage because the existing regulations require an assumed “major accident” to confirm the calculated doses to individuals are below applicable reference values.

Safety Objective Assessment

An evaluation against the QHOs is inherent in the LMP process. The LMP provides for evaluation of classes of AOOs, DBEs, and BDBEs against the F-C targets derived from regulatory limits and safety objectives to ensure the design includes adequate safety margins and DID. The LMP also provides for a comparison of the cumulative risk of all LBEs against the NRC QHOs to ensure an appropriate overall level of safety.

Safety Classification:

The LMP includes a risk-informed process to classify SSCs in the following three classes: safety-related (SR), nonsafety-related with special treatment (NSRST), and nonsafety-related with no special treatment (NST). The LMP provides an approach to SSC safety classification that begins with an evaluation of all PRA-modeled LBEs to identify safety-significant functions. Safety-significant functions include functions that contribute to meeting the F-C target values, that are significant in relation to one of the LBE cumulative risk metrics, or to meeting DID criteria. RSFs are a subset of safety-significant functions identified from the safety analysis as those functions modeled in a PRA necessary to (1) maintain the consequences of a postulated DBE or the frequency of a high-consequence BDBE within the LBE F-C target and (2) ensure that the accident dose reference values can be conservatively met.

Safety-significant SSCs include all those SSCs relied upon to perform the safety-significant functions. Risk-significant SSCs are those SSCs necessary to perform an RSF to mitigate consequences of DBEs having consequences within one percent of the F-C target values and are a subset of the set of safety-significant SSCs. The Safety-Related SSCs are a set of SSCs selected by the plant designer that are capable of:

- performing all the RSFs necessary and sufficient to mitigate DBEs within the F-C target
- ensuring that the accident dose reference values can be conservatively met for DBAs selected from the DBEs
- performing a RSF to prevent escalation of high-consequence BDBEs to beyond the F-C target values within the DBE frequency band

The Venn diagram in Figure 7 shows the relationships among the safety-related SSCs, risk-significant SSCs, safety-significant SSCs, and the SSCs modeled in the PRA. Figure 12 provides a summary of SSC classifications under the LMP and the associated definitions. Both figures were drawn from NEI 18-04, Rev. 1.

Figure 7: Relationship Between LMP Categories of SSCs

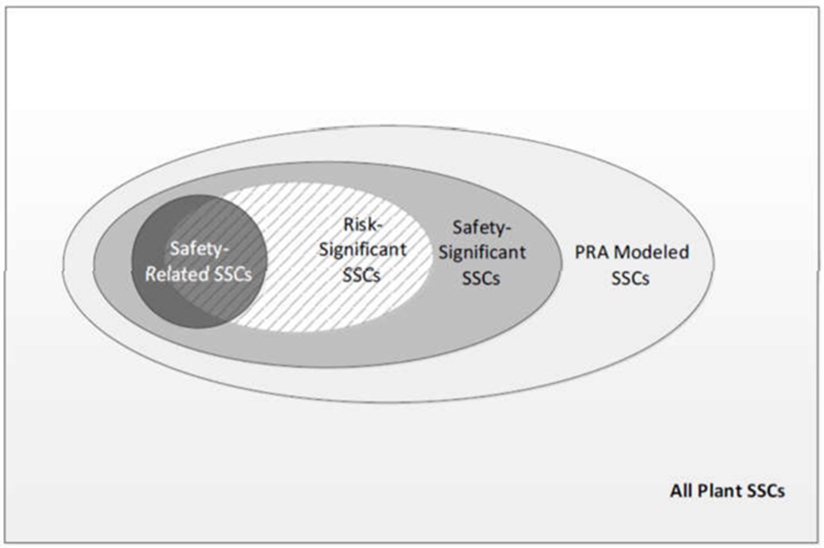
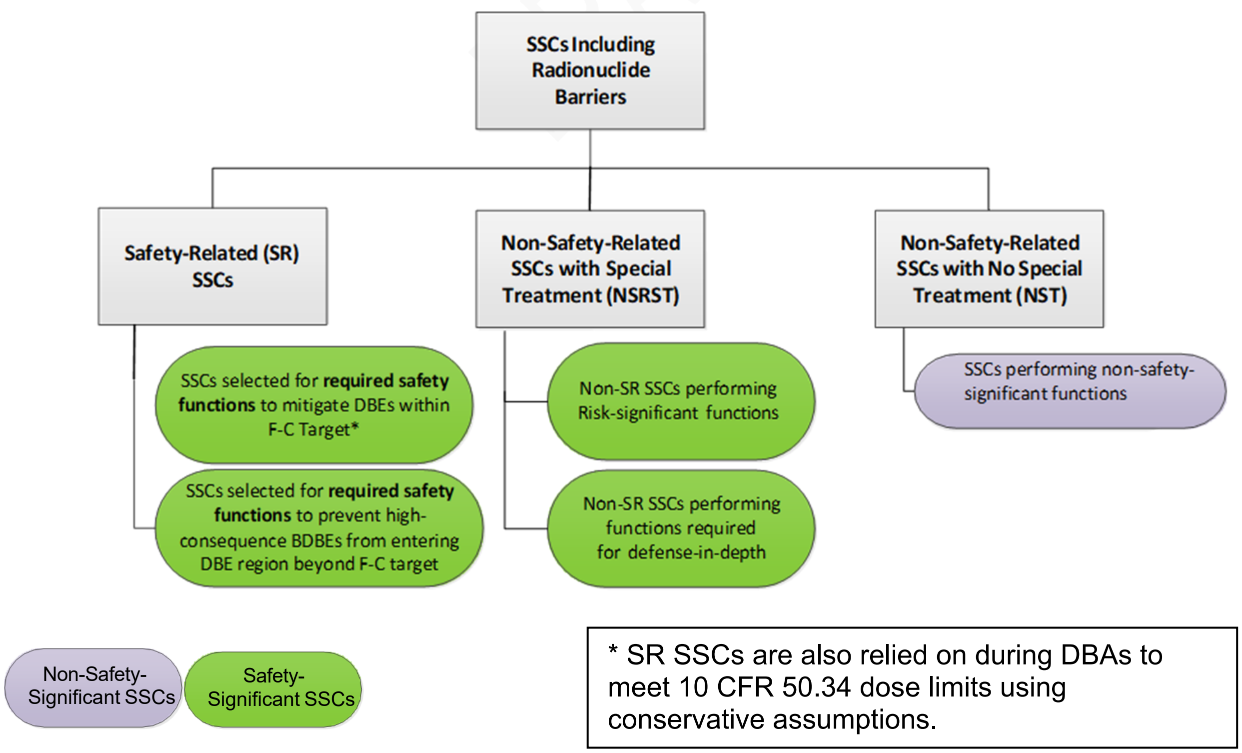


Figure 8: Relationship Between LMP SSC Classifications



Assignment of Engineering Design Rules:

Table 5 is an abbreviated version of Table 4-1, “Summary of Special Treatments for SR and NSRST SSCs,” from NEI-18-04, Rev. 1.

Table 5: Applicability of Engineering Design Rules in under LMP

Engineering Design Rule	LMP SR SSCs	LMP NSRST SSCs
Reliability Assurance	In Scope	In Scope
Maintenance Program	In Scope	In Scope
Quality Assurance	10 CFR Part 50, Appendix B	Extent Necessary for Reliability Assurance
Seismic Qualification	Full Qualification	No Interference with SR RSFs following SSE
Protection Against Design Basis External Events	In Scope	Not Specified
Equipment Qualification	10 CFR 50.49	Not Specified
Pre-Service and Inservice Inspection and Testing	In Scope	Extent Necessary for Reliability Assurance

Details regarding application of these engineering design rules based on safety classification are discussed in Sections 7, 8, and 9 of this report.

4. Comparison of Safety Classification Approaches

The licensing approaches available under the CNSC and NRC regulatory frameworks have many similarities and some differences. The similarities include expectations regarding (1) the development and classification of event sequences, (2) the general incorporation of risk information into the safety analysis, (3) the identification of safety-significant functions, and (4) the classification of SSCs based on functions. The differences relate primarily to (1) the degree the regulatory approach is risk-informed, (2) the boundary values and specific acceptance criteria applied in the dose consequence and safety assessments, and (3) the process for assigning safety classifications to SSCs.

4.1. Safety Analysis

All regulatory approaches provide requirements and guidance to systematically identify and classify an appropriate set of PIEs for the conceptual reactor design, with consideration of the proposed site. Section 3.3 of the CNSC Approach/LMP Comparison provides a detailed description of the identification and classification of PIEs/LBEs for those two approaches. For the NRC traditional approach, the SRP and DG-1413 provide comparable guidance. As such, the set of PIEs and resulting outcomes in identifying and classifying events for analysis for a conceptual design at a site with similar characteristics are expected to be similar. Therefore, this readily supports a joint review of initiating event identification and classification.

Another area in which the regulatory approaches are similar is in the establishment of design criteria. The NRC traditional approach requires PDC, with minimum requirements for LWR PDC established in the GDC in Appendix A to 10 CFR Part 50 and guidance for non-LWR PDC provided in RG 1.232. The reliability of specified functions is established through deterministic criteria, such as the assumption of a single failure. The CNSC provides similar criteria in the plant and SSC design information included in REGDOC 2.5.2. In addition, the CNSC provides a quantitative failure on demand criterion for systems performing important to safety functions. The LMP again uses a fully risk-informed methodology to establish the necessary functional reliability to achieve the safety target values. These similarities support a joint review of the application and are expected to result in similar performance and reliability outcomes for individual SSC designs.

4.2. Probabilistic Analysis

The degree a regulatory approach is risk-informed can be considered on a spectrum between prescriptive rules derived from experience combined with conservative deterministic analyses at one end and performance criteria derived from detailed probabilistic analyses that consider uncertainties at the other extreme. The NRC traditional approach, the CNSC approach, and the LMP are distributed on this spectrum.

Although the NRC traditional approach includes risk information, it is incorporated qualitatively in deterministic criteria included in the regulations. Risk information from quantitative analysis methods is employed to develop insights and to verify that cumulative safety objectives have been met. An example of qualitative incorporation of risk information is the event classification process and development of the associated acceptance criteria. The Traditional NRC licensing approach qualitatively groups credible events into AOOs or DBAs, and deterministic acceptance criteria are established for each class of event considering the likelihood. Another example is the use of principal design criteria to require redundancy for more safety-significant functions. In addition, certain BDBEs have been directly incorporated into the regulations with deterministic evaluation criteria established, in part, through consideration of risk. Classification of SSCs is determined by a qualitative assessment of the safety importance of the functions performed by an SSC. The voluntary use of the risk-informed classification process defined in 10 CFR 50.69 by an applicant for an LWR permit or license increases the consideration of risk in the classification process and the application of engineering design rules. However, the deterministic functional classification criteria remain part of the classification process.

The CNSC approach, as described in REGDOC 2.4.1, is more risk-informed than NRC's traditional approach to SSC classification in that probabilistic safety assessment information is incorporated more fully in the safety analysis. Postulated event sequences are classified based on the estimated frequency of the sequence, and each postulated event sequence class has a consequence target value established considering the frequency of occurrence of the class of events. In selecting events for the different class of events, (AOO, DBA, BDBA,) engineering judgement and operating experience are also taken into account in addition to the PSA information. Safety assessment information is also incorporated in assessing the reliability of individual systems. Furthermore, defence-in-depth considerations are integrated with safety assessment information to provide a risk-informed perspective on classification.

The LMP is fully risk-informed. The frequency of the postulated event sequence is evaluated with consideration of uncertainty, and the target value for the sequence consequences varies with the frequency. Defense -in -depth is incorporated in the assessment of event sequences in the PRA. The LMP provides a risk-informed method of classification that considers the importance of the functions performed by each SSC.

Figure 9: Classification Approaches on Risk-Informed Continuum

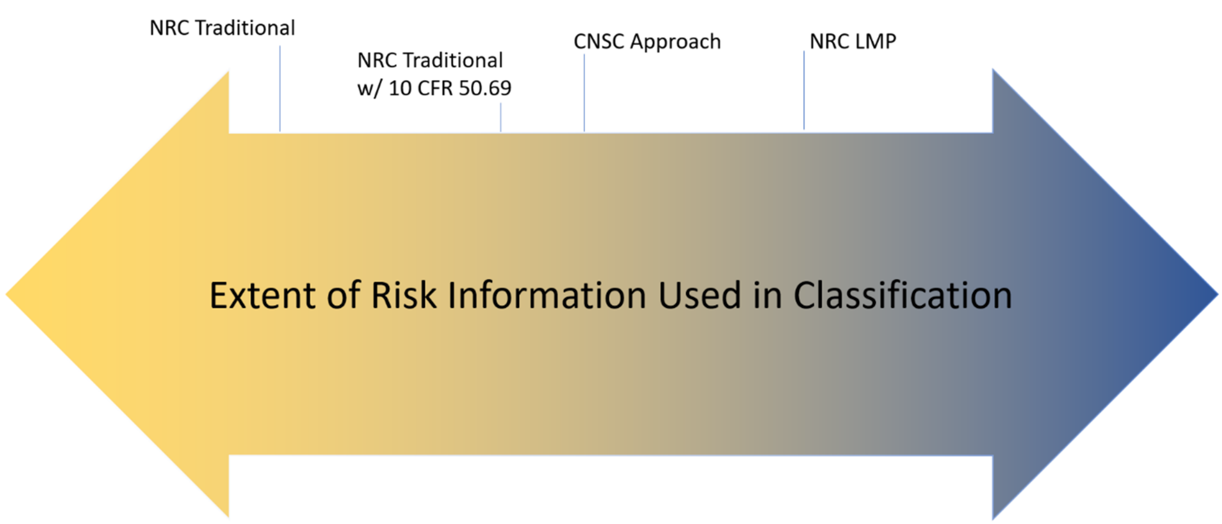


Table 10, “General PRA Topics and Risk Metrics within CNSC and NRC Frameworks,” in the CNSC Approach/LMP Comparison provides a thorough listing of the use of probabilistic information in each approach. Although presented as a comparison between the LMP and CNSC approach, the table includes LWR information that would also be applicable to the traditional NRC licensing approach for SMRs.

4.3. Safety Functions

Table 6 provides a comparison between the safety-significant functions as referenced in CNSC REGDOC-2.5.2, “Design of Reactor Facilities: Nuclear Power Plants,” the NRC traditional approach as reflected by the GDC of Appendix A to 10 CFR Part 50 that specify consideration of single failures for light water reactors, and NRC RG 1.233.

Table 6: Safety Functions

Function Category	CNSC (REGDOC-2.5.2)	NRC Traditional (GDC Reliability Considerations)[Note]	NRC LMP (RG 1.233)
Control of reactivity	Control of reactivity	• Inherent reactivity feedback (GDC 11) • Protection system reliability (GDC 21) • Reactivity control system redundancy (GDC26)	Reactivity and Power Control
Heat removal	Removal of heat from fuel	• Residual heat removal (GDC 34) • Emergency core cooling (GDC35) • Containment heat removal (GDC38)	Heat removal
Containment of radioactive material and radiation control	• Confinement of radioactive material • Shielding against radiation • Control of operational discharges; limitation of accidental releases	• Reactor coolant pressure boundary design (GDC 14) • Containment design (GDC 50 and 51) • Containment isolation (GDC 54-57)	Radioactive material retention
Support and monitoring systems	Monitoring of safety-critical parameters to guide operator actions	• Electric power (GDC 17) • Equipment cooling water (GDC44)	Required safety functions may rely on support systems, and instrumentation may support required operator actions

Note: The NRC staff has identified similar ARDC for non-water-cooled reactors in RG 1.232.

All methods applied to a proposed reactor design establish the safety-significance of SSCs through application of the following simplified steps:

- Identification of radionuclide sources and barriers to release
- Determination of safety functions
- Selection of PIEs and event combinations considered for licensing (licensing basis events)
- Identification of a set of SSCs that perform the safety functions with the necessary reliability to meet performance goals

For a given preliminary plant design, the CNSC and NRC expect the process of selecting PIEs and determining the functions that must be accomplished to satisfy the fundamental safety functions to be similar for each regulatory framework. That outcome is a result of the commonality in fundamental safety functions and the necessary safety functions being a natural outcome of the design. Therefore, the SSCs performing the fundamental safety functions are expected to be designed to similar performance criteria. These similarities support a joint review of the application and are expected to result in similar performance and reliability outcomes for individual SSC designs.

4.4. Defense-in-Depth

All safety analyses methods include measures for consideration of DID. Section 3.6 of the CNSC Approach/LMP Comparison describes the frameworks for assessment of DID as similar and generally consistent with the concept of layers of defense described in IAEA standards. The NRC traditional approach incorporates elements of DID through application of the PDC, compliance with certain regulations, and consideration of PRA results to provide reasonable assurance that DID principles have been effectively incorporated into the reactor design.

Selected GDC from Appendix A to 10CFR Part 50 align well with the CNSC Defence Levels (DLs). CNSC DL 1 includes SSCs that reliably perform the fundamental safety functions during normal operation through conservative design. These SSCs may also perform functions at other defence levels. Several NRC criteria define conservative performance capabilities for SSCs preforming fundamental safety functions during normal operations (e.g., GDC 10, Reactor Design; GDC 13, Instrumentation and Control; GDC 14, Reactor Coolant Pressure Boundary; GDC 17, Electric Power Systems; and GDC 44, Cooling Water).

The CNSC second level of defence aligns predominantly with those SSCs that detect and respond to component or system failures categorized as AOOs without exceeding conservative operational limits for Defence Level (DL) 1 components and have sufficient reliability such that the frequency of sequences including the AOO and failure of the DL 2 mitigation function fall within the DBA frequency band. Several NRC design criteria establish conservative design limits for SSCs important to maintain fundamental safety functions following AOOs (e.g., GDC 15, Reactor Coolant System Design; GDC 17, Electric Power Systems; GDC 20, Protection System Function; GDC 26, Reactivity Control System Redundancy and Capability; GDC 33, Reactor Coolant Makeup; and GDC 34, Residual Heat Removal). In addition, GDC 29, Protection against AOOs, specify that protection and reactivity control systems be designed with an extremely high probability of accomplishing their safety functions in the event of an AOO.

The CNSC third level of defence includes those SSCs that perform functions to mitigate DBAs such that severe damage conditions are prevented. The CNSC DL 3 SSCs should have significant independence from DL 1 and DL 2 SSCs in order to satisfy safety objectives. Several NRC design criteria and certain regulations ensure fundamental safety functions would be satisfied to specified performance levels that prevent severe damage under accident conditions (e.g., GDC 17, Electric Power Systems; GDC 20, Protection System Function; GDC 27, Combined Reactivity Control System Capability; GDC 35, Emergency Core Cooling; and 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors).

The fourth level of defence includes both SSCs designed to minimize any radiological release from DBAs (e.g., a structural containment and its isolation system) and SSCs placed in service during DECs to limit further damage or mitigate releases (e.g., temporary systems to perform fundamental functions or SSCs designed to preserve containment during severe accident conditions). The DL 4 SSCs should also have significant independence from the DL 3 SSCs. Several NRC design criteria and certain regulations provide for the capability to mitigate radioactive material releases and limit further damage under accident conditions, including beyond design basis accidents (e.g., GDC 16, Containment Design; GDC 38, Containment Heat Removal; GDC 41, Containment Atmosphere Cleanup; GDC 50, Containment Design Basis; 10 CFR 50.44, Combustible Gas Control for Nuclear Power Reactors; and 10 CFR 50.155, Mitigation of Beyond-Design-Basis Events). The NRC traditional approach specifies design criteria for reliable containment function that is independent of accident mitigation functions to the extent practicable, and beyond-design-basis accident mitigation capability is independent of design-basis accident mitigation capabilities.

Thus, the three regulatory approaches provide robust measures for establishment of design related DID measures. The NRC LMP and CNSC licensing approaches provide for discrete assessments to verify acceptable DID. The NRC traditional approach specifies design criteria for reliable accomplishment of safety functions that align well with the design related DLs 1 through 4 as outlined above. Also, the NRC traditional approach incorporates conservatisms, including:

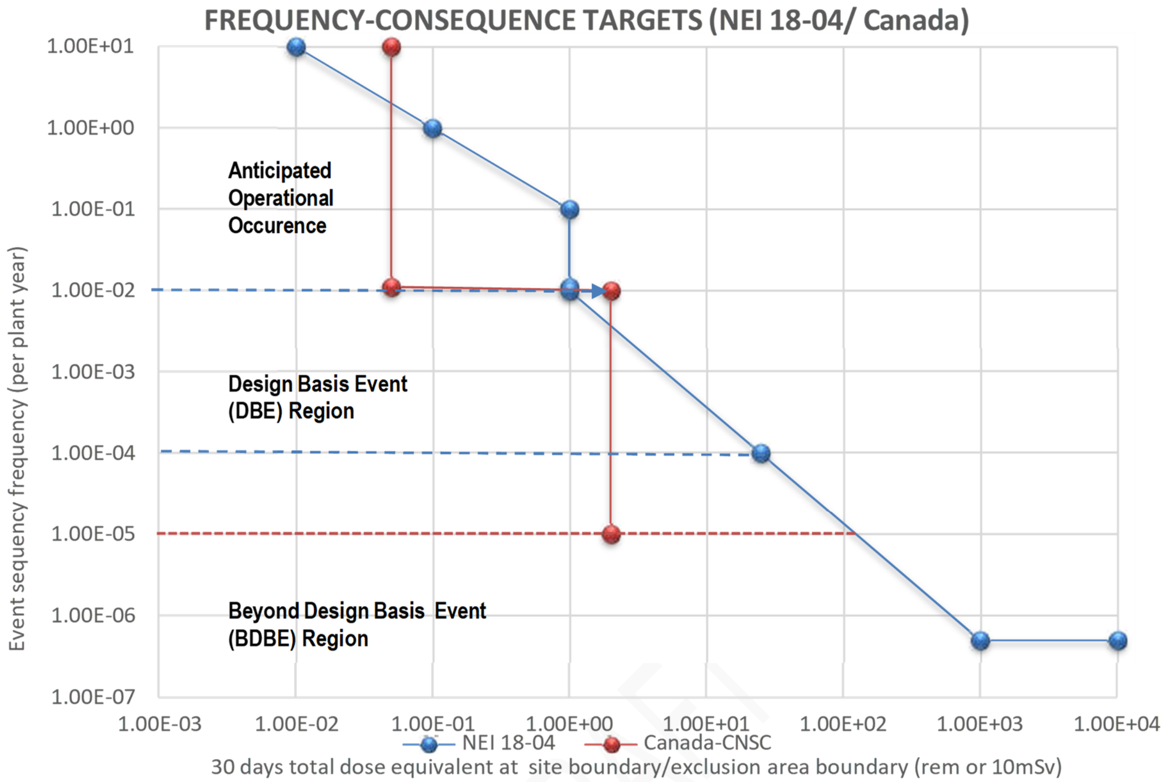
- evaluation of events against acceptance criteria established considering the initiating event frequency rather than the event sequence frequency
- consideration of SSCs with the highest safety classification for mitigation of AOOs

These considerations help ensure that the principles of DID are met, and probabilistic analyses may be used to confirm DID adequacy. Therefore, the comparable consideration of DID in each regulatory framework supports a joint review of applications.

4.5. Design Basis Accident Dose and Safety Objective Assessment

Both the NRC and CNSC regulatory frameworks consider the likelihood and consequences of the postulated event sequences in establishing performance goals, with more likely event sequences having lower acceptable consequences. The NRC and the CNSC have independently established risk-informed performance goals for new reactor licensing that establish quantitative bounds, with methods to address uncertainty in the values estimated for the frequency and consequences of postulated LBEs. Figure 10 (originally presented as Figure 9 of the CNSC Approach/LMP Comparison) illustrates a comparison of the frequency-consequence (F-C) targets by superimposing the NRC endorsed F-C target curve with a plot of CNSC frequency thresholds and dose acceptance criteria.

Figure 10: Comparison of NRC and CNSC Frequency-Consequence Targets



Under traditional NRC licensing approaches, acceptable functional reliability has been achieved through application of deterministic criteria to safety functions. This deterministic approach generally provides for conservative results. The guidance for the evaluations and analyses specifies conservative assumptions, and the GDC include provisions to ensure many safety functions can be accomplished assuming a single failure and using either onsite or offsite power alone. The GDC also establish conservative acceptance criteria for the An important distinction from the CNSC and the LMP approaches in the conduct of the evaluations and analyses is that the NRC traditional acceptance criteria reflect the frequency of the initiating event rather than the frequency of the event sequence that leads to the evaluated end state. Another difference is the SRP guidance specifying reliance on safety-related SSCs to ensure the acceptance criteria are satisfied following postulated transients and accidents. The CNSC and LMP approaches generally consider availability of SSCs in a more risk-informed way, although the CNSC approach requires consideration of single failures in accident analysis and the LMP approach specifies that selected SSCs necessary and sufficient to perform required safety functions for mitigation of DBAs or satisfy the dose assessment requirements of 10 CFR Part 50 or 10 CFR Part 52 be classified as safety-related.

Table 7 provides a summary of the safety analyses and regulatory considerations associated with reactor operating licensing paths under the CNSC framework, 10 CFR Part 50, 10 CFR Part 52, and the LMP:

Table 7: Comparison of Licensing Paths and Safety Analysis Criteria

Safety Analyses and Event Groups	CNSC Regulatory Approach	NRC Traditional Licensing (Part 50 or 52)	NRC Risk-Informed, Technology-Inclusive Licensing (LMP)
Deterministic Safety Analysis	Deterministic design criteria	Required by regulation; includes development of principle design criteria	Necessary for development of principle design criteria and DBA dose analysis
Probabilistic Safety Analysis	PSA including release frequency complements deterministic analysis	PRA including release frequency confirms acceptable level of safety	PRA necessary for evaluation of LBE frequency and dose consequences
AOO Frequency	Event sequence $\geq 10^{-2}$ per reactor year	One or more occurrences of initiating event during life of reactor	Event sequence $\geq 10^{-2}$ per plant year
AOO Acceptance Criteria	Important to Safety (ITS) SSCs undamaged, safety functions met, and no escalation	Acceptable fuel design limits, fundamental safety functions met, and no escalation using SR SSCs	Meet SSC functional design criteria and prevent escalation
AOO Dose Criteria	≤0.5 mSv (0.05 rem) per event - PSA and best estimate calculations	Part 20 dose limit - ≤1 mSv (0.1Rem) per year	F-C Target per event – PRA w/uncertainty; Cumulative dose below Part 20 limits
DBA / DBE Frequency	Event sequence $\geq 10^{-5}$ and $<10^{-2}$ per reactor year	Event/accident selection guidance	Event sequence $\geq 10^{-4}$ and $<10^{-2}$ per plant-year
DBA / DBE Acceptance Criteria	Fuel within DBA limits; Mitigating SSCs meet functional design criteria with single failure	Fundamental safety functions met using SR SSCs 10 CFR 50.46 LWR criteria	F-C Target per event; Mitigating SR SSCs meet functional design criteria
DBA Dose Criteria	≤20 mSv (2 rem) for any DBA[Note A] – conservative deterministic or best estimate plus uncertainty	Hypothetical release ≤250 mSV (25 Rem) TEDE[Note B]; lower dose consequence targets for other mechanistic DBAs	DBAs mitigated by SR SSCs only ≤250 mSV (25 Rem) TEDE [Note B]
DEC or BDBE Frequency	Below 10^{-5} occurrences per reactor year – best estimate	Deterministic design criteria for certain events	From 5 x 10^{-7} to 10^{-4} per plant year – w/uncertainty
DEC or BDBE Constraints	Prevent high-consequence events	QHOs considering PRA uncertainty	F-C Target per event
Cumulative Constraints	CNSC Safety Goals[Note C]	Targets derived from QHOs	Targets derived from QHOs

Table 5 Notes:

A Per REGDOC-2.5.2: The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, shall be calculated in the deterministic safety analysis for a period of 30 days after the analyzed event.

B Per 10 CFR 50.34(a)(1): A deterministic evaluation determines that an individual at the following locations would not receive a dose in excess of 25 rem TEDE:

1. any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release; and
2. any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage)

C Core degradation and small release frequency <10^{-5} per reactor year; LRF <10^{-6} per reactor year

All regulatory approaches provide requirements and guidance to evaluate LBEs and DBAs and assess the performance measures against appropriate evaluation criteria. Section 3.3 of the CNSC Approach/LMP Comparison provides a detailed description of the methods to assess the reactor SSC performance following LBEs in each event class (i.e., AOOs, DBEs/DBAs, and BDBEs/BDBAs/DECs) against defined criteria. For the NRC traditional approach, the PDCs and regulations, such as 10 CFR 50.34(a)(1) and 10 CFR 50.46, provide evaluation criteria for certain classes of accidents, some applicable only to LWRs. Regulatory guidance supplements these regulations for evaluation of specific mechanistic DBAs applicable to LWRs. For advanced reactors, the evaluation criteria for dose consequence analyses remain the same as for LWRs, but evaluation criteria for other aspects applicable to advanced reactors, such as fuel and functional containment evaluation criteria, would be established on a case-by-case basis considering the PDC established for the design. For all three regulatory approaches, the evaluation criteria provide for comparable design performance, although the specific evaluation criteria vary. Additionally, the dose consequence evaluations are affected by the site configuration as well as the reactor design. Therefore, the CNSC and NRC staffs expect similar outcomes in identifying the necessary SSC safety functions to meet the evaluation criteria for all event classes.

4.6. Similarities and Differences in Safety Classification

Licensing approaches, including the LMP and the CNSC licensing approach for new reactors, are more directly risk-informed by the results of probabilistic analyses. The adequacy of the SSCs identified for mitigation of the individual PIEs would be assessed against the performance goals expressed in terms of event frequency and resulting consequences. Although there are differences in the performance goals and analysis conditions, a comparison of the CNSC and NRC approaches determined that similar outcomes in the classification of SSCs could be expected [Section 3.5 of CNSC Approach/LMP Comparison]. The resulting classification groups would be similar because the licensing processes have shared characteristics that support accurate ranking of SSC safety importance.

The traditional NRC classification approach is similar to the LMP with respect to the number of classification groups; however, the terms and definitions are not consistent. The regulations in 10 CFR Part 50 support classifications of safety-related, important to safety, and not important to safety (NITS). Under the NRC traditional approach, the applicant determines SSC safety classification solely based on the functions performed by the SSC. The CNSC approach is informed by both the deterministic safety analysis and probabilistic safety analysis.

An important distinction in the classification of SSCs under CNSC requirements and guidance compared to the NRC classification systems (LMP or Traditional) is the role of safety significance in the classification process. Under the CNSC approach, the vendor/applicant defines a number of important to safety classification categories, and the classification of important to safety SSCs within those categories is determined by the relative safety significance of each SSC. The safety significance in the CNSC approach is based on:

1. safety function(s) to be performed
2. consequence(s) of failure
3. probability that the SSC will be called upon to perform the safety function
4. the time following a PIE at which the SSC will be called upon to operate, and the expected duration of that operation

This approach considers all the functions of the SSC, including both safety functions and contribution to DID.

The two NRC classification systems (traditional and LMP) use three categories of classification: safety-related, important to safety, and not important to safety under the traditional approach; and SR, NSRST, and NST under the LMP. Under either approach, the vendor/applicant selects a set of SSCs to be classified as safety-related. This set of SSCs must perform all safety-related functions as specified in the definition of safety-related SSC under the traditional classification process or perform all required safety functions under the LMP classification process. Required safety functions under LMP are those functions both necessary and sufficient to (1) maintain the consequences of a postulated DBE or the frequency of a high-consequence BDBE within the LBE F-C target and (2) ensure that the accident dose reference values can be conservatively met. The SSCs performing risk-significant functions or functions important to DID that were not selected as safety-related are classified as NSRST under the LMP process or important to safety under the NRC traditional process.

5. Impact of Safety Classification Differences

The CNSC and NRC expect that the safety classification results for a proposed new reactor design would be similar in effect. Any differences in the engineering design rules applied to a particular SSC could be reconciled through application of existing risk-informed processes. The CNSC regulatory framework provides for gradation of classification of important to safety SSCs. The NRC regulatory framework provides for classification of a designer-selected set of SSCs as safety-related, where the selected SSCs are capable of performing all functions captured within the definition of safety related associated with the regulatory approach (either traditional or LMP). The remaining safety significant SSCs further reduce risk by providing independent means of accomplishing safety functions, providing protection against certain hazards, or otherwise enhancing DID, and these SSCs are classified as important to safety (not safety-related) under the NRC traditional approach or NSRST under the LMP. Therefore, the CNSC framework provides applicants the flexibility to define two or more graded classifications of important to safety SSCs, while the NRC process classifies a nearly identical set of SSCs into a broad important to safety classification and an included safety-related classification.

5.1. Classification Exercise

To get a better sense of the opportunities and challenges regarding reviewing a new reactor licensing application, the CNSC and NRC staffs elected to conceptually apply each regulatory framework to the safety classification of SSCs and assess the means of reconciling differences that may affect the assignment of engineering design rules.

This exercise is focused on the safety classification process. In order to simplify this safety classification exercise, the following assumptions were used:

Table 8: Assumptions for Classification Exercise

Topic	Assumption	Basis
Plant Design	Identical design of a single-reactor plant for deployment in U.S. and Canada	A common design with assumed bounding site hazards (i.e., external transportation conditions and natural phenomena) would support cost-effective deployment. Single-reactor plant design avoids complication with LMP F/C targets established on a per-plant-year rather than a per-reactor-year basis.
Event Identification	Identical consideration of postulated events	Each regulatory framework provides comparable guidance for identification of events to be considered in the design basis. The plant design assumptions result in equivalent consideration of internal events, external events, and natural phenomena.
Safety Significant Functions	SSCs are capable of performing all necessary functions under each regulatory framework	Each regulatory framework identifies essentially identical key safety functions. With an identical plant design, identical consideration of postulated events, and

		common safety functions, SSCs are expected to be capable of performing necessary safety functions under each regulatory framework.
Safety Analysis	Applicant uses analysis methodologies consistent with the selected regulatory framework to establish the necessary SSC performance characteristics	Each regulatory framework includes specific degrees of conservatism aligned with the licensing approach. The established SSC performance characteristics are an iterative output of the design and the analysis framework applied to the evaluation of the design.
Probabilistic Analysis	Quality probabilistic analyses support the safety analysis and assessment of DID	Confirms that safety objectives associated with selected regulatory approach have been satisfied.
Dose Consequence Assessment	Deterministic dose assessments demonstrate bounding accident consequences satisfy the target values under each regulatory framework	Under both the CNSC and NRC regulatory frameworks, conservative analyses of bounding accidents use similar methods and assumptions to demonstrate target values would be met. A common design is expected to satisfy regulatory requirements under each framework (including assessment of a major core damage event under NRC regulations or an approved exemption from that requirement (See *Design Basis Accident Dose Assessment* subheading in Section 3.2.1 of this report)).

5.2. Leveraging of CNSC Framework Outcome for an Application to NRC

Application of the CNSC framework involves consideration of both SSC safety function and functions that contribute to DID (as shown in Figure 6, CNSC classification encompasses SSCs that perform safety functions, prevent accidents, limit the effect of hazards, and protect against radiological releases).

Application of the requirements and guidance of REGDOC 2.5.2 support evaluation of a water-cooled reactor design and may be adapted to other reactor technologies. Section 6.2 of REGDOC 2.5.2 requires that fundamental safety functions (i.e., those listed in Table 6 of this report under the CNSC heading) be available during normal operational states, DBAs, and DECs, except when the postulated accident involves loss of that function. The CNSC specifies that SSCs important to safety are those SSCs that directly perform the fundamental safety functions and their necessary support systems, complementary design feature SSCs that contribute during DECs, and certain process and control systems. The guidance in REGDOC 2.5.2 permits applicants to sort SSCs important to safety into a graded sub-classification for assignment of engineering design rules.

The CNSC regulatory framework in REGDOC 2.5.2 also specifies that a systematic review of the design be performed to verify that measures at all five levels of DID (see Table 1 of this report) have been established and there is acceptable independence between the SSCs performing functions at each level of defence.. The systematic review for DID should involve consideration of the deterministic and probabilistic analyses of the design, and information from the analysis should be used to determine assignment of SSCs to specific defence levels and ensuring that acceptable independence between levels of defence has been established.

The safety classification process typically would result in SSCs preventing failures during normal operation and responding to design basis accidents to establish a stable condition having the highest classification among important to safety SSCs. These functions include, for example, maintaining reactor core geometry or maintaining reactor heat removal, where failure could result in unacceptable consequences because fundamental safety functions would not be accomplished.

Considering the assumptions listed in Table 8 that result in essentially identical sets of SSCs performing identical functions for a specific reactor design, an assessment of adaptations necessary to use the CNSC classification process output under the NRC framework was performed.

In the NRC's regulatory framework, one of the more significant requirements related to design involves establishing the PDC for the facility and the relationship of the facility design basis to the PDC. The GDC of Appendix A to 10 CFR Part 50 establish the minimum requirements for PDC for water-cooled nuclear power plants and provide guidance to applicants for development of PDC for other types of nuclear power units.

The PDC provide criteria for safety functions and reliability standards that relate to safety classification. Safety classification based on safety significance for assignment of engineering design rules, as described in REGDOC 2.5.2, is consistent with GDC-1.

The scope of design criteria included in REGDOC 2.5.2, including the elements related to DID, is similar to the scope of the GDC. However, applicants should verify a design based on REGDOC 2.5.2 conforms with the proposed facility PDC in the NRC Traditional Framework or in the NRC LMP Framework, as appropriate.

As noted in Section 3.2.1, the NRC regulatory framework includes several special purpose regulations to enhance DID that may be applicable to the proposed reactor design. With respect to safety classification of SCCs, the applicant should establish whether compliance with these regulations would require alteration of the reactor design. Alternatively, the applicant may seek NRC approval of an appropriately justified request for exemption from applicable regulations. Related guidance in NRC RGs may specify application of particular engineering design rules to SSCs necessary to satisfy the requirements of these special purpose rules, and deviations from the guidance should be justified on a risk informed basis.

As discussed in Section 3.2.1 of this report, several NRC regulations apply to safety-related SSCs, which are defined in 10 CFR 50.2. A safety classification approach using the CNSC regulatory framework likely identifies a broader set of SSCs as important to safety than those SSCs that would meet the definition of safety-related SSCs under the NRC framework.

In order to use the CNSC classification approach to define a set of SSCs under the scope of the NRC regulations, the applicant should compare the criteria for determination of the highest safety classification in the CNSC license application to identify differences in scope from the 10 CFR 50.2 definition of safety related. If exceptions to NRC regulations exist, then a request for approval of exemptions from the regulations would be needed. The exemption request may be informed by a risk-informed justification, but must meet the requirements of 10 CFR 50.12, “Specific Exemptions.“ This report provides no evaluation of the acceptability of any such exemption requests.

As noted in Section 3.2 of this report, applicants for LWR certificates, approvals, permits, or licenses under 10 CFR Part 50 or 10 CFR Part 52 must include an evaluation of the facility against the SRP. The safety classification of SSCs is particularly relevant to conformance with the guidance in SRP Chapter 3 related to SSC design and classification (i.e., seismic design criteria), SRP Chapter 15 related to transient and accident analyses (i.e., SSCs assumed to function in the analyses are classified as safety-related), SRP Chapter 17 related to quality assurance (i.e., Section 17.4 related to reliability assurance and special treatments), and SRP Chapter 19 related to severe accidents and DID (i.e., Section 19.3 on RTNSS classification). However, safety classification is discussed in several other SRP chapters. These applicants should provide a risk informed justification supporting deviations from the guidance in the SRP.

5.3. Leveraging of NRC Framework Outcome for an Application to CNSC

The less prescriptive CNSC design and classification requirements supports a more flexible, but still rigorous method of reconciling NRC and CNSC SSC classification approaches. The CNSC SSC safety classification process supports progressive classification from relatively low safety significant SSCs to high safety significant SSCs based on consideration of specific factors (described in Section 3.1), and the application of engineering design rules becomes progressively stricter as safety significance increases, consistent with the graded approach used in the CNSC framework.

The SSCs classified as safety-related under the NRC classification frameworks would also have high safety significance under the CNSC approach because of the functions performed and the likelihood the functions would be needed to prevent or mitigate development of adverse consequences. However, there are potential circumstances where an SSC with high safety significance would be classified as NSRST or important to safety (not safety-related) under the NRC classification process when another SSC can perform the same function.

As an example, an SMR or advanced reactor with a passive heat removal system may also include in the design a reliable, active heat removal system. Assuming the active system is less reliable but set to actuate to prevent initiation of the passive system, the active system may have about the same safety significance and, therefore, the same applied engineering design rules, as the passive system under the CNSC safety classification process.

In the NRC classification process, if the vendor/applicant selects the passive system to perform the safety function, the passive system would be classified as safety-related with the more comprehensive engineering design rules. The active system would be treated as important to safety but not safety-related in the traditional classification method and NSRST under the LMP, which results in lesser application of engineering design rules. This situation could result in different application of engineering design rules to the active and passive heat removal systems in Canada and the U.S. because of potentially different relative safety classifications.

The consistency in application of engineering design rules would be improved by the use of a risk informed process that considers DID in assigning those rules. These processes are present in the reliability assurance program applicable to NSRST SSCs under the LMP or the RTNSS (important to safety) SSCs under the NRC traditional process, as discussed in Section 6.1 of this report. Under the NRC traditional framework, the reliability assurance program described in SRP Section 17.4, or the risk informed classification process defined in 10 CFR 50.69, could be used to ensure that a systematic assessment of DID had been completed and that appropriate engineering design rules would be applied to SSCs commensurate with their importance to safety.

5.4. Summary of Reconciliation Approaches

In summary, the CNSC and NRC staff anticipate that applicants would choose a common design that complies with requirements imposed by both regulatory bodies due to significant alignment in the CNSC and NRC regulatory frameworks.

Minor deviations from regulations or requirements could be justified on a risk informed basis provided that the underlying intent of the regulation or requirement continues to be satisfied (i.e., NRC exemption or CNSC alternative approach). Deviations from associated guidance are explained and justified using a risk informed decision-making approach. Table 9 outlines expected actions, given the assumptions outlined in Table 8, to align safety classifications when either regulatory framework is used for the initial SSC safety classification process:

Table 9: Leveraging Prior SSC Classification Process Outcomes

Process	*Leveraging of CNSC Classification Framework Outcome for an NRC Application*	*Leveraging of NRC Classification Framework Outcome for a CNSC Application*
Original Classification Process	CNSC safety significance process supported by defence level confirmation	NRC Traditional approach with reliability assurance program (RAP) or risk informed safety classification (RISC); or LMP [RAP, RISC, and LMP processes each support confirmation of DID.]
Compliance with Requirements	• Development of PDC and verification of design conformance with PDC. [For water-cooled reactors, the GDC represent the minimum requirements for PDC, so an applicant may need to seek approval of exemption request for deviations from GDC.] • Reconcile any differences in safety analysis approach necessary to satisfy PDC and establish SSC design basis. • Ensure SSCs in highest CNSC safety class meet all functions specified in existing 10 CFR 50.2	• Conformance expected based on assumptions in Table 8. • Justify any deviations from REGDOC 2.5.2 requirements, leveraging information from risk informed processes (NRC RISC or LMP evaluation processes).

	definition of "safety-related," modify the SSCs within the safety class such that all functions specified in the definition of "safety-related" are met or seek approval of an exemption to redefine the scope of regulations that establish scope using the term "safety-related." (It's anticipated that many advanced reactor NRC permit and license applicants will seek approval of an exemption to essentially redefine "safety-related.") • Conform with applicable special purpose regulations or seek approval of exemption. • Address conformance with SRP (Especially Chapters 3, 15, 17, and 19) (water-cooled reactors only).	
Defense in Depth	• For NRC traditional approach, DID is presumptively assured by conformance with the regulations (including conformance with PDC) and confirmed through a risk informed evaluation process, which may consider input from CNSC graded licensing approach. • LMP verifies acceptable DID through an Integrated Decision-making Process Panel, which may consider input from CNSC graded licensing approach.	Demonstrate conformance with defense levels by leveraging information from risk informed classification processes.
Assignment of Design Rules	• Apply design requirements for SSCs classified as safety-related or as established by exemptions (e.g., Appendices B and S to 10 CFR Part 50) for NRC Traditional and LMP • Demonstrate that CNSC safety significance classification process provides acceptable graded assignment of rules per GDC 1 for other SSCs classified as important to safety in NRC Traditional framework.	Demonstrate that assignment of design rules supported by information from risk informed classification processes provides acceptable assignment of design rules consistent with the CNSC safety significance classification process.

	• CNSC safety significance classification process may inform assignment of design rules to SSCs classified as NSRST under the LMP.	

Table 10 indicates the expected alignment of SSC functional description and SSC Classification under the three regulatory frameworks. The anticipated differences are small except for the classification differences when an SSC reliably performs a fundamental safety function but is not among the SSCs designated as safety-related for that function within the NRC framework (dark blue row in Table 10). When this circumstance exists in a reactor design, the alignment of applied engineering design rules may be established by considering the risk informed processes used in the original classification to ensure assignment of engineering design rules commensurate with the safety significance of the SSC.

Table 10: Comparison of Safety Classification Outcomes

SSC Description	CNSC Classification	NRC Traditional Classification	NRC LMP Classification
System or Structure Passively Performing or Reliably Actuated to Perform Fundamental Safety Function (FSF)	Important to Safety (ITS) (High to Medium Safety Significance)	Safety-Related (Selected to Perform Function)	Safety-Related (Selected to Perform Function)
System or Structure Automatically or Manually Actuated to Perform (or Normally Performing) FSF	ITS (High to Low Safety Significance)	ITS (Defense-in-Depth and Regulatory Treatment of Nonsafety Systems)	NSRST (For Risk-Significant Function or Defense-in-Depth)
Complementary Design Features or SSCs (Enhancement, Preservation, or Restoration of FSFs)	ITS (Medium to Low Safety Significance)	ITS (Defense-in-Depth)	Safety Related if to Mitigate High-Consequence BDBE, NSRST (Defense-in-Depth), or NST
Essential Support Systems for SSCs Performing FSFs	Same as Supported SSC	Same as Supported SSC	Same as Supported SSC
SSCs whose Failure Could Adversely Affect SSCs that Perform FSFs	Generally Same as Affected SSC	ITS	NSRST or NST

The following table provides an overview of expected safety classifications relative to safety significance under each of the three regulatory frameworks:

Table 11: Safety Classification Overview (Illustrative Purposes Only)

Safety Significance	**High**			**Low**
CNSC	Important to Safety (ITS)			Not Important to Safety (NITS)
	ITS - High	ITS - Medium	ITS - Low	NITS
NRC LMP	LMP Risk Significant (and Safety Significant)	LMP Safety Significant		Not Safety Significant
	Safety-Related			Non-Safety-Related No Special Treatment
	Non-Safety-Related with Special Treatment			
NRC Traditional	Important to Safety			NITS
	Safety-Related (RISC-1)		Safety-Related (RISC-3)	NITS
	ITS (Not Safety-Related) (RISC-2)		ITS (RISC-4)	

The more safety-significant SSCs are expected to be universally included in a safety classification imposing the most comprehensive engineering design rules. The NRC LMP process is anticipated to result in the smallest group of SSCs subject to engineering design rules because the classification methodology is founded in a risk-informed probabilistic framework with full consideration of uncertainties and DID principles. portion of SSCs designated as LMP Safety-Related, and thus subject to the most comprehensive engineering design rules, is expected to be smaller than the portion of SSCs designated NRC Traditional Safety-Related because the process for selection of the safety-related SSCs under the LMP is fully risk-informed, whereas the NRC traditional approach relies on a more conservative deterministic approach.

The CNSC approach may result in a somewhat larger scope of SSCs subject to engineering design rules due to consideration of deterministic analyses and more conservative performance targets; however, there will be flexibility afforded to applicants/vendors due to the CNSC"s risk-informed approach. The NRC traditional licensing approach may also include a larger scope of SSCs important to safety due to the potential for a fully deterministic safety analysis and deterministic classification process.

6. Programmatic Engineering Design Rules

6.1. Reliability Assurance Programs, Including Maintenance and Availability

Reliability programs play an important role in establishing further application of engineering design rules in risk-informed, technology neutral licensing processes, such as the CNSC regulatory approach for new reactors and the NRC LMP process. The reliability programs operate in an iterative fashion with the licensing probabilistic safety assessment (PSA) because reliability of SSCs is an input to the PSA and changes to SSC reliability through changes in redundancy, diversity, and independence may help the PSA results meet performance goals included in the licensing process. Accordingly, reliability programs are included in CNSC REGDOC 2.5.2, SRP Section 17.4, and NEI 18-04 as part of the process to establish engineering design rules applied to the design of nuclear reactor SSCs.

Reliability assessment plays an important role for nuclear reactor designs that rely on passive design features for safety-related functions. In that type of reactor, over-reliance on assumed single failures of active components and other prescribed reliability measures may not provide appropriate DID. The reliability assurance program supports evaluation of DID considerations in these reactor types.

Both the CNSC and NRC rely on the maintenance program scope to ensure that SSC reliability in operation remains consistent with the importance of individual SSCs to reactor safety and DID.

6.1.1. Canada

REQUIRED ENGINEERING RULES – RELIABILITY IN DESIGN

Requirements related to reliability can be found throughout REGDOC-2.5.2, with Section 7.6 being dedicated to design for reliability.

This section requires that all SSCs important to safety be designed with sufficient quality and reliability to meet the design limits and that a reliability analysis be performed for each of these SSCs. Where possible, the design must provide for testing to demonstrate that the reliability requirements will be met during operation. This section also states that the safety systems and their support systems be designed to ensure that the probability of a safety system failure on demand from all causes is lower than 1×10^{-3}.

The reliability model for each system is expected to use realistic failure criteria and best estimate failure rates, considering the anticipated demand on the system from PIEs. Design for reliability must take account of mission times for SSCs important to safety and the availability of offsite services upon which the safety of the plant and protection of the public may depend, such as the electricity supply and external emergency response services.

Additional sections with reliability related requirements are given below.

REGDOC-2.5.2, Section 7.5: Design rules and limits

The design authority shall specify the engineering design rules for all SSCs. These rules shall comply with appropriate accepted engineering practices. The design shall also identify SSCs to which design limits are applicable. These design limits shall be specified for operational states, DBAs and DECs

The engineering design rules for all SSCs should be determined based on their importance to safety, as determined using the criteria in section 7.1. The design rules should include, as applicable:

- identified codes and standards
- conservative safety margins
- reliability and availability:
  - material selection
  - single failure criterion
  - redundancy
  - separation
  - diversity
  - independence
  - fail-safe design
- equipment qualification:
  - environmental qualification
  - seismic qualification
  - qualification against electromagnetic interference
- operational considerations:
  - testability
  - inspectability
  - maintainability
  - aging management
- management system

The design for reliability is based on meeting applicable regulatory requirements and industry standards. The design should provide assurance that the requirements of CNSC REGDOC 2.6.1 [34], “Reliability Programs for Nuclear Power Plants,” will be met during operation. Not all SSCs important to safety identified in the design phase will necessarily be included in the reliability program.

The following principles are applied for SSCs important to safety:

- the plant is designed, constructed, and operated in a manner that is consistent with the assumptions and risk importance of these SSCs

- these SSCs do not degrade to an unacceptable level during plant operations

- the frequency of transients posing challenges to SSCs is minimized

- these SSCs function reliably when challenged.

Deterministic analysis or other methods may be used if the PSA lacks effective models or data to evaluate the reliability of SSCs.

REGDOC-2.5.2, Section 8.4 Means of shutdown

The design shall permit ongoing demonstration that each means of shutdown is being operated and maintained in a manner that ensures continued adherence to reliability and effectiveness requirements.

Periodic testing of the systems and their components shall be scheduled at a frequency commensurate with applicable requirements.

The reliability calculation should include sensing the need for shutdown, initiation of shutdown, and insertion of negative reactivity. All elements necessary to complete the shutdown function should be included.

The reliability of the shutdown function should be such that the cumulative frequency of failure to shut down on demand is less than 10^{-5} failures per demand, and the contribution of all sequences involving failure to shut down to the LRF of the safety goals is less than 10^{-7} per year. This considers the likelihood of the initiating event and recognizes that the two shutdown means may not be completely independent.

CNSC REQUIREMENTS – MAINTENANCE IN DESIGN

Maintenance requirements in design are mostly related to ensure that provision is made to ensure that SSCs are capable of performing their function throughout the lifetime of the plant. For example, section 5.1 specifies that the design authority must ensure the availability of the design information that is needed for safe plant operation and maintenance. Similarly, section 5.2 on Design Management requires that safety design information necessary for safe operation and maintenance of the plant and for subsequent modifications be preserved. It also requires a design objective that the plant design itself facilitate maintenance and aging management.

Section 7.14 on inservice testing, maintenance, repair, inspection, and monitoring, sets design requirements related to the maintenance activities necessary to maintain an NPP within the boundaries of the design over the lifetime of the plant. It states that such activities be performed to standards commensurate the importance of the safety function. Guidance in section 7.14 promotes the development of strategies and programs to address maintenance as a necessary aspect of the plant design phase.

6.1.2. United States

DESIGN RELIABILITY ASSURANCE PROGRAM (D-RAP)

The reliability assurance program (RAP) is implemented in two stages. The first stage, the design reliability assurance program (D-RAP), encompasses reliability assurance activities that occur before initial fuel load to ensure that the plant is designed and constructed in a manner consistent with risk insights and key assumptions. The second stage comprises the reliability assurance activities conducted during the operations phase of the plant’s license.

The RAP applies to those SSCs, both safety-related and nonsafety-related, identified as risk-significant (or significant contributors to plant safety). The SSCs within the scope of the RAP are identified by using a combination of probabilistic, deterministic, and other methods of analysis to identify and quantify risk. Appropriate quality assurance (QA) controls for the design activities for nonsafety-related RAP SSCs are implemented in accordance with Part V of SRP Section 17.5.

If certain types of risk-significant SSCs are excluded from the RAP, the applicant provides a rationale for excluding these SSCs and addresses how other programs and requirements ensure that these SSCs do not degrade to an unacceptable level of reliability, availability, or condition during plant operations and will function reliably when challenged.

The second stage of the RAP comprises the reliability assurance activities conducted during the operations phase of the plant license to ensure that the reliability and availability of RAP SSCs are maintained commensurate with their risk significance. During the operations phase the RAP is implemented through regulatory requirements for SSCs, including (1) 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, (2) Application of the Quality Assurance Program requirements for safety-related SSCs, (3) QA controls for nonsafety-related RAP SSCs (4) the inservice inspection, inservice testing, surveillance testing, and maintenance programs. Prior to initial fuel load, the licensee identifies dominant failure modes and integrates RAP into operational programs. During the operations phase of the plant, performance and condition monitoring is implemented to provide reasonable assurance that these RAP SSCs do not degrade to an unacceptable level of reliability, availability, or condition.

Requirements and Guidance

The Commission approved the Design Reliability Assurance Program in NRC SECY-95-132 [35], “Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems (RTNSS) in Passive Plant Designs,” Item E, “Reliability Assurance Program,” dated May 22, 1995. The staff guidance for implementation of a D-RAP is primarily derived from SECY-95-132.

Applications for new LWR licenses, approvals, or certifications must evaluate the proposed design against the SRP in effect 6 months before the date of the application. NUREG-0800, the SRP, provides guidance to NRC staff in performing safety reviews of LWR CP or OL applications under 10 CFR Part 50 and DC, COL, SDA, or ML applications under 10 CFR Part 52. The principal purpose of the SRP is to assure the quality and uniformity of staff safety reviews.

SRP Section 17.4, “Reliability Assurance Program,” Revision 1, May 2014 (ADAMS Accession No. ML13296A435), addresses the Reliability Assurance Program.

SRP Section 19.3, “Regulatory Treatment of Nonsafety Systems for Passive Advanced Light Water Reactors,”, Revision 0, June 2014 (ADAMS Accession No. ML14035A149) provides guidance on identifying nonsafety-related SSCs that perform risk-significant functions in a passive plant design and are candidates for regulatory oversight. RTNSS SSCs are one input to the D-RAP program.

Specific Codes and Standards

The Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) for passive plants (Volume III, Chapter 1) issued March 1999 by the Electric Power Research Institute (EPRI) specifies standards concerning the design and performance of nonsafety-related active systems and equipment that perform functions which support safe operation of the facility. Appropriate levels of reliability and availability for these systems are established with the RAP and RTNSS process.

Additional Information

SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Nonsafety Systems in Passive Plant Designs," dated March 28, 1994, and associated SRM, dated June 30, 1994 (ADAMS Accession No. ML003708068) and SRM approved D-RAP and disapproved the operational reliability assurance program (O-RAP). The Commission directed the staff to ensure that the objectives of the O-RAP are incorporated into existing programs for maintenance or QA.

SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems (RTNSS) in Passive Plant Designs," Item E, "Reliability Assurance Program," dated May 22, 1995, approved by the Commission in the SRM to SECY-95-132, dated June 28, 1995, describes the details of RAP and provides guidance for developing an effective RAP.

NEI 18-04, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactors," defines special treatment standards for SSCs in Table 4.1, "Summary of Special Treatments for SR, NSRST SSCs." This table specifies application of a RAP to all LMP safety-significant SSCs, including reliability and availability targets for performance of PRA safety functions. The table describes the program as essentially the same as the Reliability Assurance Program in SRP Section 17.4.

MAINTENANCE RULE (10 CFR 50.65)

Overview

The Maintenance Rule regulation, 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," was issued in 1991. The requirements in 10 CFR 50.65(b) relate to the criteria used to determine which SSCs are to be scoped into the Maintenance Rule. While the Maintenance Rule is a performance-based regulation and not generally proscriptive, the scoping requirements detail specific criteria for both safety-related and nonsafety-related SSCs. Licensees apply these scoping requirements to establish the population of plant SSCs that will be monitored under the facility's Maintenance Rule program.

Requirements and Guidance

10 CFR 50.65 does not list specific SSCs as being required to be monitored under the Maintenance Rule. 10 CFR 50.65(b)(1) addresses SSCs classified as safety-related and prescribes scoping criteria to determine if a safety-related component is required to be monitored under the Maintenance Rule. Monitoring under the Maintenance Rule is required for safety-related SSCs that are relied upon to remain functional during and following design basis events to ensure:

- the integrity of the reactor coolant pressure boundary,
- the capability to shut down the reactor and maintain it in a safe shutdown condition, and
- the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the accident dose reference values in regulations.

10 CFR 50.65(b)(2) addresses SSCs classified as non-safety related. Monitoring under the Maintenance Rule is required for non-safety related SSCs:

- that are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs); or

- whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related function; or

- whose failure could cause a reactor scram or actuation of a safety-related system.

The risk-informed SSC categorization process of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," allows LWR licensees to establish alternative treatments in lieu of compliance with 10 CFR 50.65 requirements for performance monitoring and goal setting for SSCs that are categorized as RISC-3 (safety-related, but low safety-significant) or RISC-4 (nonsafety-related and low safety-significant). In cases where an alternative treatment has been established for RISC-3 and RISC-4 SSCs, licensees are not required to scope these SSCs into the Maintenance Rule per 50.65(b) for monitoring under 50.65(a)(1), (a)(2), and (a)(3). However, the requirements under 10 CFR 50.65(a)(4) for assessment of increases to plant risk due to maintenance activities remain in-force. For advanced reactor applicants proposing to use the LMP, the SSCs expected to be addressed by the maintenance rule would be similar to those determined to be within the scope under 10 CFR 50.69.

NRC RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 4 (ADAMS Accession No. ML18220B281) endorses, with exemptions and clarifications, the industry guidance contained in Nuclear Management and Resources Council (NUMARC) 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 4F (ADAMS Accession No. ML18120A069).

Much of the guidance in NUMARC 93-01 is based on establishing reliability and availability performance criteria to establish thresholds to assess effective maintenance for scoped SSCs. NUMARC 93-01, Section 8.0, "Methodology to Select Plant Structures, Systems, and Components," contains guidance related to determining the SSCs that should be scoped into the Maintenance Rule. The guidance focuses on a safety function approach to determine in scope SSCs. Specific guidance is also provided related to each of the scoping criteria for both safety-related and nonsafety-related SSCs contained in 10 CFR 50.65(b). This guidance includes considerations such as explicit and implicit use of SSCs in EOPs, determining SSC functions, and the use of industry operating experience.

Specific codes and standards

IEEE Standards Association, IEEE 933-2013, "IEEE Guide for the Definition of Reliability Program Plans for Nuclear Generating Stations and Other Nuclear Facilities," approved December 11, 2013, provides guidance for constructing a reliability program to help achieve improved plant safety and performance. The standard includes a brief, high-level overview of the Maintenance Rule, including several excerpts from the revision of NUMARC 93-01 guidance in effect at that time. The overview notes that Maintenance Rule concepts related to determining SSC scoped equipment can indicate a structure for input to the scope and detail of a reliability program. However, no specific links to the Maintenance Rule are identified.

Additional information

Nuclear Energy Institute (NEI) 18-10, "Monitoring the Effectiveness of Nuclear Power Plant Maintenance," Revision 0 (ADAMS Accession No. ML19302F210) was issued in July 2019. Industry has not requested endorsement of NEI 18-10 and describes the guidance as a more economical approach to ensure compliance with the Maintenance Rule as compared to the guidance in

NUMARC 93-01. All U.S. nuclear plants adopted and implemented the NRC endorsed guidance of NUMARC 93-01 until 2019. Since 2019, several (less than 15) U.S. plants have revised their Maintenance Rule programs based on the non-endorsed guidance of NEI 18-10. Thus far, none of the adopting facilities strictly follow the NEI 18-10 guidance and Maintenance Rule programs based on NEI 18-10 will have some differences from plant-to-plant.

Unlike the NUMARC 93-01 guidance, NEI 18-10 does not establish reliability and availability performance criteria thresholds to assess effective maintenance for scoped SSCs. The guidance in NEI 18-10 places a greater focus on SSCs determined to have high safety-significance (regardless of SSC classification) and relies more heavily on automated failure trending and impact to plant risk. However, similar to NUMARC 93-01, NEI 18-10, Section 6.0, “Scoping, Determining Safety-Significance, and Establishing and Implementing Maintenance Strategy,” focuses on a safety function approach to determine in scope SSCs.

6.1.3. Similarities and Differences

Both the CNSC and the NRC have established reliability programs as part of the facility licensing review to ensure engineering design rules applied to an SSC provide predicted reliability commensurate with its importance to safety. These programs apply to important to safety SSCs (i.e., for the NRC, the scope includes safety-related and important to safety classifications, such as RTNSS and NSRST).

The CNSC reliability goals are broader than the associated NRC Reliability Assurance Program guidance in that the goals address system failure on demand probabilities and other capabilities that are satisfied, in part, by design of the systems, such as redundancy, diversity, independence, and protection from hazards. The corresponding design elements in the NRC regulatory framework are established (1) for the traditional licensing approach through application of the single failure criterion and other design considerations included in the facility PDC or in NRC regulations, with confirmation of safety through probabilistic risk assessment results, and (2), for the LMP approach, through the probabilistic risk assessment of the design to ensuring appropriate reliability and capability targets have been developed for safety-significant SSCs. However, the reliability programs, defined in both the CNSC regulatory framework and the NRC guidance, share many other attributes.

The shared reliability program attributes relate to QA, testing, inspection, availability management, and maintenance. As a design focused review, this assessment focuses on the QA, tests, and inspections conducted as part of initial construction and how the scope of SSCs subject to availability management, operational testing, and maintenance would be determined based on the safety classification. The CNSC regulatory framework provides for graded association of these attributes with SSCs, commensurate with each SSC’s importance to safety, which is reflected in the safety classification. Within the NRC framework, regulations require the application of QA, inspection, testing, maintenance, and availability management to safety-related SSCs, with graded application commensurate to the safety significance of each SSC defined in associated standards. For design of important to safety SSCs under the NRC framework, GDC 1 specifies that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. A risk-informed classification process defined in 10 CFR 50.69 may be applied to limit full application of reliability measures to the more safety-significant SSCs among those classified as safety-related. For the SSCs classified as important to safety but not safety-related (e.g., SSCs classified as NSRST under the LMP or under the RTNSS policy for passive LWRs), the D-RAP review process ensures appropriate reliability attributes have been established for components with this safety classification, commensurate with the SSC’s safety-significance.

This review includes assurance that the nonsafety-related SSCs included in the D-RAP are also within the scope of 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.”

6.1.4. Assessment of Impact

The CNSC and NRC expect that the application of reliability attributes based on safety classification would have similar outcomes. The most safety-significant SSCs would be subject to the most complete QA, inspection, testing, maintenance, and availability management. Similarly, the important to safety components with lower safety-significance would have reduced, but still substantial, reliability attributes assigned commensurate with the SSC’s importance to safety. Although the CNSC approach offers additional flexibility with respect to assignment of reliability attributes, the overall impact of differences is expected to be small because the most important attributes for reliability are the most likely to be retained. Conversely, the CNSC regulatory framework includes specific, quantitative reliability goals for system performance that may be more conservative than those accepted under the NRC regulatory framework. These quantitative goals may result in more conservative system designs than those specified by NRC design requirements and guidance. However, the impact would remain small because probabilistic assessment methods would identify unwarranted risk increases resulting from insufficient redundancy or independence and result in design changes to correct these conditions.

6.2. Quality Assurance During Design and Construction (Future)

6.2.1. Canada

6.2.2. United States

6.2.3. Similarities and Differences

6.2.4. Assessment of Impact

6.3. Scope of Inservice Testing and Inspection *(Future)*

6.3.1. Canada

6.3.2. United States

6.3.3. Similarities and Differences

6.3.4. Assessment of Impact

7. Design of Specific Structures, Systems, and Components

7.1. Pressure-Retaining Components and Supports

Overview of Code Classification

This section covers code classification and assignment of engineering design rules for pressure retaining components and their supports. Such components include pressure vessels, heat exchangers, storage tanks, piping systems, pumps, valves, core support structures, supports and similar items.

Consensus standards provide sets of engineering design rules applied to the construction of pressure retaining components and supports. Construction, as used here, is an all-inclusive term that includes material selection, design, fabrication, installation, examination, testing, overpressure protection, inspection, stamping, and certification. These sets of requirements establish varying levels of component quality and reliability in service. When adopted by a regulatory authority for a specific application, these standards become design codes.

Code classification builds on the safety classification of structures, systems, and components, with components having the highest safety classification often being further divided into two or three code classifications. Typically, component code classes are set commensurate with the importance of the component's function in assuring safety (pressure boundary integrity in this case). Specific criteria that define the code class and indicate the associated code or standard to be applied can be established by the regulatory authority or proposed by an applicant depending on the regulatory approach. The owner or applicant of a nuclear facility seeking a license or certification from the regulatory authority uses code classification of components among the means to demonstrate safe design. The design rules vary depending on the code class selected and the design rules that must be applied to that class.

For pressure retaining components, the code classification is done by the design owner based on safety criteria, regulatory requirements, and characteristics of the specific reactor design. The code classification then determines which part of the code or standard is applied to best assure protection against catastrophic failure, initiation and propagation of cracks, excessive material creep, or fatigue failure. As discussed in the following sections, nuclear power plants in the United States and Canada normally reference consensus standards from the American Society of Mechanical Engineers (ASME) as codes for design of nuclear power plant pressure retaining components and supports.

Section III, "Rules for Construction of Nuclear Facility Components," of the ASME Boiler and Pressure Vessel Code (BPVC) [36] provide rules for the construction of pressure retaining nuclear components and their supports with the greatest importance to safety. Section III is subdivided into Divisions, with Division 1 and Division 5 applicable to pressure retaining components and associated supports. Division 1 includes three classes of design rules for metal components operating at non-elevated temperatures, and Division 5 includes two classes of design rules for components operating at elevated temperatures, including both metal components and nonmetallic core support structures.

ASME Section III Division 1 Nuclear Facility Components

ASME BPVC Section III Division 1 govern the construction of metallic vessels, heat exchangers, storage tanks, piping systems, pumps, valves, supports, core support structures and similar items for use in nuclear facilities. The scope of existing sub-sections does not address creep and stress rupture characteristics of materials permitted by Section III rules. Therefore, the existing sub-sections are limited to operating temperatures where those phenomena are not important to material behavior. The following table provides the organization of the ASME BPVC, Section III, Division 1:

Table 12: ASME BPVC, Section III, Division 1 Organization

Subsection	Title	Description
NCA	General Requirements	Establishes general structure and rules applicable to Division I sub-sections
NB	Class 1 Components	Pressure retaining components with highest safety-significance
NC	Class 2 Components	Pressure-retaining components with intermediate safety-significance
ND	Class 3 Components	Pressure-retaining components with moderate safety-significance
NE	Class MC Components	Metal containment vessel components (addressed with civil structures in this report)
NF	Supports	Structural supports for Class 1, 2, and 3 pressure retaining components
NG	Core Support Structures	Reactor vessel internal support structures

ASME Section III Division 5, "High Temperature Reactors"

ASME BPVC Section III Division 5 is a high temperature reactor (HTR) component code used to ensure structural integrity at high operating temperatures. Division 5 rules coordinate with Division 1 because a HTR will also have components operating at lower temperatures. Division 5 rules govern the construction of vessels, piping, pumps, valves, supports, core support structures and nonmetallic core components for use in HTR systems and their supporting systems. For low temperature service, Division 5 often references Division 1 rules. Division 5 includes Class A and Class B components, which are subject to rules similar to Division 1 Class 1 and Class 2, respectively. Division 5 is organized as shown in the table below:

Table 13: ASME BPVC, Section III, Division 5 Organization

Subsection	Title	Subpart	Description
HA	General Requirements	A	Metallic Materials
		B	Graphite Materials
		C	Composite Materials
HB	Class A – Metallic Pressure Boundary	A	Low Temperature Service
		B	Elevated Temperature Service
HC	Class B – Metallic Pressure Boundary	A	Low Temperature Service
		B	Elevated Temperature Service
HF	Class A and Class B	A	Low Temperature Service

	Metallic Supports		
HG	Class A Metallic Core Support Structures	A	Low Temperature Service
		B	Elevated Temperature Service
HH	Class A Nonmetallic Core Support Structures	A	Graphite Materials
		B	Composite Materials

Other Applicable Codes and Standards

Other codes and standards apply to pressure retaining components and supports having lower safety-significance. These standards generally reflect commercial levels of quality and reliability. For example, ASME BPVC, Section VIII [37], "Rules for Construction of Pressure Vessels," provides engineering design rules for commercial pressure vessels made of several different materials and operating under a variety of service conditions, and ASME B31.1 [38], "Power Piping," provides engineering design rules for construction of pressure-retaining piping in power plants.

7.1.1. Canada

In Canada, the CNSC uses regulatory documents and licence conditions as the means to establish both requirements and guidance. CNSC REGDOC-1.1.2, *Licence Application Guide: Licence to Construct a Nuclear Reactor Facility* [39] sets out requirements and guidance on applying to the CNSC to obtain a licence to construct a reactor facility in Canada. It states that the application should describe the basis for the design of the pressure- or fluid-retaining SSCs and their supports, in order to meet the expectations of section 5.7 *Pressure-retaining structures, systems and components* of REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*. The application should also describe the pressure boundary codes and standards (and their editions / effective dates) and the overall pressure boundary program implementation processes and procedures.

CNSC REGDOC-2.5.2 elaborates further on requirements by including guidance to licensees and applicants on how to meet requirements. Licensees are expected to review and consider this guidance; if they choose not to follow it, they should explain how their selected approach still meets regulatory requirements.

Establishing which design codes apply for pressure retaining components is currently done following the requirements of CSA N285.0 [40], "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants." The specific objectives of the N285 series are:

a. To establish rules relating to authorization, approval, and acceptance where such rules are different from those specified in the ASME Code;
b. To specify requirements for materials and rules for the design, fabrication, installation, examination, inspection, testing, and repair of pressure-retaining systems and components, when such systems and components are not within the scope of the ASME Code;
c. To establish rules for classifying systems and components based on principles and criteria consistent with the Canadian safety philosophy, as promulgated by the CNSC;
d. To establish rules for the periodic inspection of CANDU nuclear power plants;

e. To make provision for interpretation of the rules contained by the Standards of this series.

For CANDUs, code classification is done following the N285.0 flow chart shown below:

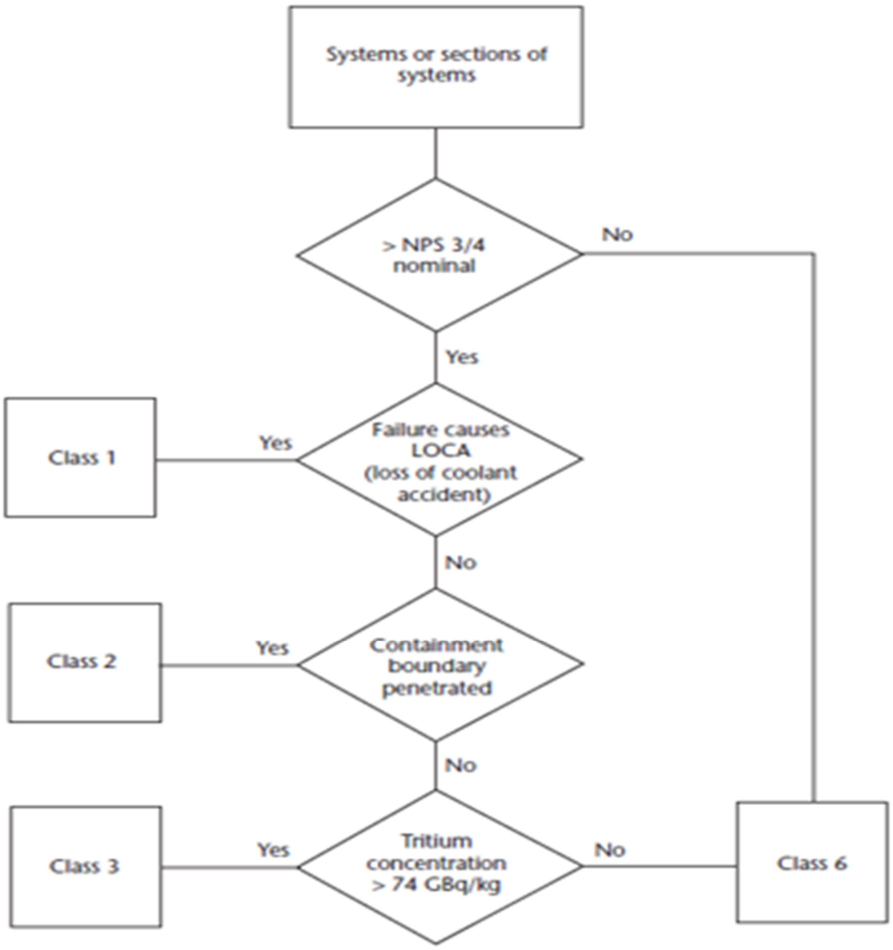


Figure 11: N285.0 Simplified guide to the classification of process systems

Based on the code class, N285.0 specifies technical requirements for the design, procurement, fabrication, installation, modification, repair, replacement, testing, examination, and inspection. Importantly, for this project, it should be noted that CSA N285.0 does not contain any requirements that would prevent it being applied to technologies other than CANDU. Applicants propose the code classification and associated standards but must meet expectations defined in REGDOC 2.5.2. The N285 classes are shown below with the assigned design code:

Table 14: Canadian Standards Association N285 Classification and Standards

N285.0 system class	Design code
Class 1	ASME BPVC, Section III, Division 1, Subsection NB
Class 2	ASME BPVC, Section III, Division 1, Subsection NC
Class 3	ASME BPVC, Section III, Division 1, Subsection ND
Class 1C, 2C, 3C* (CANDU only)	ASME BPVC, Section III, Division 1, NB, NC, ND and CSA N285.0, Annex E, “Requirements for Class 1C, 2C, and 3C components and supports in nuclear power plants”
Class 4 (Metal Containment components – see civil structures)	ASME BPV Code, Section III, Division 1, NE and CSA N285.0, Annex F, “Design rules for containment components”
Class 6	CSA B51 (ASME BPVC, Section VIII, B31.1 or B31.3)
Class 1, 2, 3 Supports	ASME BPVC, Section III, Subsection NF supports

*Components-unique to CANDU design hold the same system classification but with the additional designation of ‘C’

ASME BPVC is aimed at the component level and only covers the technical requirements for the construction of new components and supports after they have been classified, whereas CSA N285.0 includes requirements for classification of systems and components, plant and system documentation and registration. Once the systems have been classified and the classification boundaries have been established, then components and supports adopt the classification of the system. The construction of these items is then based on technical requirements that are either the same or similar to those in ASME BPVC, Section III, Division 1.

For the design of pressure-retaining systems and components, the REGDOC-2.5.2 guidance states that the design authority should ensure the selection of codes and standards is commensurate with the safety class and is adequate to provide confidence that plant failures are minimized. Guidance in REGDOC-2.5.2 references the ASME BPVC for the construction of pressure retaining components and in-vessel support structures. Alternative codes and standards may be used if this would result in an equivalent or superior level of safety; justifications should be provided in such cases.

In addition to ASME requirements, REGDOC 2.5.2 includes general design requirements directly related to pressure-retaining systems, structures, and components as well as in-direct requirements (cross-cutting). Requirements in the document may not be specific to any industry code or standard but have their origins in operating experience or IAEA safety standards such as IAEA SSR-2/1, “Safety of Nuclear Power Plants: Design,” Rev. 1.

7.1.2. United States

The NRC requires that applicants for a CP, DC, COL, SDA, or ML include a description and analysis of facility SSCs, including the extent to which generally accepted engineering standards are applied to the design of the reactor[10]. More specifically, 10 CFR 50.55a, “Codes and Standards,” lists codes and standards that have been approved for incorporation by reference, including multiple revisions of the ASME BPVC and specific Code Cases applicable to Section III of the ASME BPVC, and requires that pressure retaining components of LWRs conform to Section III of the ASME Boiler and Pressure Vessel Code (ASME BPV Code). In accordance with 10 CFR 50.55a(c), the reactor coolant pressure boundary components of LWRs must meet the requirements for Class 1 components in Section III of the ASME BPVC. In addition, 10 CFR 50.55a(d) and (e) require that Quality Group B and Quality Group C components must meet the requirements for Class 2 and Class 3 components in Section III of the ASME BPVC, respectively. The regulation references guidance documents[11] for the purpose of defining Quality Groups B and C. Although 10 CFR 50.55a does not impose ASME design rules on any components of non-LWRs, the NRC staff expects non-LWRs to identify the generally accepted engineering codes and standards that will be used for the design of the facility as required in other regulations.

Additionally, NRC regulations require that applicants for CPs, DCs, and COLs describe the PDC of the facility. Appendix A to 10 CFR Part 50 establishes minimum requirements for the PDC for LWRs and provides guidance for other types of nuclear power reactors. GDC 1 of that Appendix, “Quality Standards and Records,” states, in part, that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. This requirement is applicable to both pressure-retaining and non-pressure retaining SSCs that are part of the reactor coolant pressure boundary (RCPB) and other systems important to safety. While GDC 1 directly applies only to LWRs, non-LWR designs must have PDC that fulfill a similar role[12].

Quality Groups

The primary NRC guidance in RGs 1.26 [41], “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” uses quality groups to establish a level of quality for water-cooled reactor pressure retaining components and supports commensurate with their importance to safety. Quality Groups A, B, and C apply to components classified as safety-related, and Quality Group D applies to all other components of pressure-retaining components and associated supports. The following general criteria (more detailed criteria are defined in Section C of RG 1.26) establish the components identified as Quality Group A, B, and C:

---

[10] See 10 CFR 50.34(a)(1)(ii), 10 CFR 52.47(a)(2), 10 CFR 52.79(a)(2), 10 CFR 52.137(a)(2), and 10 CFR 52.157(c), respectively.
[11] Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radiological-Waste-Containing Components of Nuclear Power Plants," and in Section 3.2.2 of NUREG–0800, "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants."
[12] Examples of substitute PDC can be found in Regulatory Guide (RG) 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” which provides guidance for developing PDC for non-LWR designs.

- Quality Group A consists of the reactor coolant pressure boundary components of light water reactors, including supports.
- Quality Group B includes frontline accident mitigation systems, frontline safe shutdown systems, and systems and components whose integrity is important to maintaining containment function.
- Quality Group C applies primarily to safety-related systems supporting frontline accident mitigation or safe shutdown systems.
- Quality Group D applies to pressure retaining components that are not safety-related and their supports, which may include components considered important to safety performing functions such as:
    - Piping and supports designed only to retain structural integrity during seismic events
    - Radioactive waste system piping, pumps, and tanks
    - In LWRs with passive safety systems, SSCs that provide DID by directly acting to prevent unnecessary actuation of the passive safety systems

Components considered important to safety may serve functions to protect safety-related equipment from the effects of nonsafety-related equipment failures, the effects of natural phenomena, or the forces resulting from direct attachment to safety-related components. Other important to safety functions include DID functions or functions necessary to support long-term safe shutdown. Accordingly, additional measures related to QA, design margin, load combinations, material verification, and post-fabrication testing may be specified beyond those specified in the design standard for commercial-grade SSCs.

Table 15: NRC Classification and Standards for LWR Pressure Retaining Components

<table>
<tr><th>Component</th><th>Quality Group A</th><th>Quality Group B</th><th>Quality Group C</th><th>Quality Group D</th></tr>
<tr><td>Pressure Vessels</td><td rowspan="4">ASME BPVC, Section III, Division 1, Subsection NB: Class 1, Nuclear Power Plant Components</td><td rowspan="6">ASME BPVC, Section III, Division 1, Subsection NC : Class 2, Nuclear Power Plant Components</td><td rowspan="6">ASME BPVC, Section III, Division 1, Subsection ND: Class 3, Nuclear Power Plant Components</td><td>ASME Boiler and Pressure Vessel Code, Section VIII, Division 1</td></tr>
<tr><td>Piping</td><td>ANSI B31.1 Power Piping</td></tr>
<tr><td>Pumps</td><td>Manufacturer’s standards</td></tr>
<tr><td>Valves</td><td>ANSI B31.1 Power Piping and ANSI B16.34</td></tr>
<tr><td>Atmospheric Storage Tanks</td><td rowspan="2">Not applicable</td><td>API-650, AWWA D100, or ANSI B96.1</td></tr>
<tr><td>0-15 psig Storage Tanks</td><td>API-620</td></tr>
<tr><td>Supports</td><td>Subsection NF Class 1 supports</td><td>Subsection NF Class 2 supports</td><td>Subsection NF Class 3 supports</td><td>Manufacturers standards</td></tr>
</table>

Alternative Code Classification for Light Water Reactor Components

Appendix A, “Alternative Classification for Components in Light Water-Cooled Nuclear Power Plants,” to RG 1.26 and Section B, “Discussion,” of RG 1.26, describe potential alternative methods of code classification for LWR components. Appendix A to RG 1.26 states that an applicant may propose the classification method discussed in ANSI/ANS-58.14-2011 [42] as an alternative means to comply with NRC regulations, including GDC 1 and 10 CFR 50.55a. Separately, Section B of RG 1.26 describes that, although the NRC does not endorse IAEA Specific Safety Guide (SSG)-30, “Safety Classification of Structures, Systems and Components in Nuclear Power Plants,” applicants may propose use of this qualitative risk-informed method of classification and assignment of engineering design rules when supported by sufficient information to establish that the proposed alternative complies with GDC 1 and 10 CFR 50.55a. These alternative methods provide for classification within four categories (Class 1 through Class 4 under ANSI/ANS 58.14 and Safety Category 1 through Safety Category 3 and Not Categorized under IAEA SSG-30), similar to RG 1.26 guidance.

Code Classification for Advanced Reactors

For non-LWR advanced reactors, NRC guidance in Appendix A of Revision 2 to RG 1.87 [43], “Acceptability of ASME Code, Section III, Division 5, ’High Temperature Reactors’,” describes acceptable methods for quality group classification of components of high temperature reactors for traditional (i.e., classification based on function) and LMP safety classification processes. The LMP safety classification process defines SR, NSRST, and NSR safety classes. These classification methods are described in Section 2.6.2 of this report. The guidance in Appendix A to RG 1.87 specifies application of the ASME Code, Section III, Division 5 engineering design rules to components in the highest classification (i.e., safety-related under the traditional classification method, RISC-1 under the risk-informed categorization process defined in 10 CFR 50.69, and safety-related under the LMP process), with the distinction between Class A and Class B determined by the safety-significance of the component. The guidance states that components important to safety (i.e., important to DID or meeting the principle design criteria under the traditional classification process, RISC-2 or RISC-3 under the risk-informed categorization process of 10 CFR 50.69, and NSRST under the LMP) may be designed to appropriate commercial design codes encompassing high temperature applications, such as Section VIII of the ASME Code or ASME B31.1, with justification on a case-by-case basis. The following table provides guidance defining the quality groups applicable to the various safety classifications and the codes and standards considered acceptable for application to each group:

Table 16: NRC Classification and Standards Applicable to Advanced Reactors

<table>
<tr><th>Classification Method</th><th colspan="3">Component Classification</th></tr>
<tr><td>Traditional</td><td>Quality Group A</td><td>Quality Group B</td><td>Quality Group C</td></tr>
<tr><td>Risk-Informed (RG 1.233)</td><td>SR</td><td>SR</td><td>NSRST</td></tr>
<tr><td></td><td colspan="2">SR Quality Design Standards</td><td>Important to Safety Design Standards</td></tr>
<tr><td>Components</td><td colspan="2"></td><td></td></tr>
<tr><td>Pressure Vessels</td><td rowspan="6">ASME Code, Section III, Division 5, Class A</td><td rowspan="6">ASME Code, Section III, Division 5, Class B</td><td>ASME Code, Section VIII, Division 1 or Division 2[Note]</td></tr>
<tr><td>Piping</td><td>ASME B31.1/B31.3 [Note]</td></tr>
<tr><td>Pumps</td><td></td></tr>
<tr><td>Valves</td><td>ASME B31.1/B31.3 [Note]</td></tr>
<tr><td>Atmospheric Storage Tanks</td><td></td></tr>
<tr><td>Storage Tanks (0-15 pounds per square inch gauge)</td><td>ASME Code, Section VIII, Division 1 or Division 2 [Note]</td></tr>
<tr><td>Metallic Core Support Structures</td><td>ASME Code, Section III, Division 5, Subsection HG</td><td>N/A</td><td></td></tr>
<tr><td>Nonmetallic Core Support Structures</td><td>ASME Code, Section III, Division 5, Subsection HH</td><td>N/A</td><td></td></tr>
</table>

Note: Application of these standards should be justified on a case-by-case basis.

7.1.3. Similarities and Differences

An approximate correlation of CSA N285.0, NRC RG 1.26, and ANSI/ANS 58.14 code classifications is shown in Table 17.

Table 17: Correlation of Canadian and U.S. Water-Cooled Reactor Code Classifications

CSA N285.0	NRC RG 1.26	ANSI/ANS-58.14	Applicable ASME BPVC Subsection
Class 1	Quality Group A	Class 1	Section III, Division 1, Class 1
Class 2	Quality Group B	Class 2	Section III, Division 1, Class 2
Class 3	Quality Group C	Class 3	Section III, Division 1, Class 3
Class 6	Quality Group D	Class 4	Section VIII for pressure vessels; ASME B31.1 for piping

As indicated by Table 17, the number and identification of water-cooled reactor code classes and associated engineering design rules is similar between CNSC and NRC guidance, particularly the alternate classification guidance in Appendix A to RG 1.26, which describes the

ANSI/ANS-58.14 methodology. The components in CSA N285.0 Class 1, RG 1.26 Quality Group A, and ANSI/ANS 58.14 Class 1 would be essentially identical because the methods all rely on functional classification and the criteria defining components in this code classification relate to reactor coolant pressure boundary components of sufficient size that pressure boundary failure would have significant safety consequences. For lower code classifications, the functional classification processes would produce minor differences in sets of components because the functional criteria defining each code classification are not identical. CNSC N285.0 existing guidance for code classification (i.e., ASME Class 1, Class 2, and Class 3 criteria) are written for CANDU reactors, with Class 2 and Class 3 criteria considering tritium release potential. Therefore, the existing CANDU guidance differs from NRC functional criteria for code Class 2 and Class 3. However, because applicants may propose the code classification scheme under the CNSC regulatory approach, the code classifications are likely to be nearly identical.

For advanced non-water-cooled reactors, the safety classification of pressure retaining components would be similar to that of LWRs, with the exception that the risk-informed LMP process applies to advanced reactors. Each safety classification category translates into one or more code classifications. The highest code classification should be aligned to the most safety-significant components. For high temperature applications, the highest code classification is Class A under the ASME BPVC, Section III, Division 5. The next highest code classification within Division 5 is Class B. Lower classifications for high temperature reactors would use commercial-grade standards, such as ASME BPVC Section VIII. The regulatory frameworks for both the NRC and the CNSC rely on graded risk-informed processes to complete the safety classification for components of high temperature reactors. The NRC has developed guidance for code classification of non-LWR advanced reactors with the highest safety classification under each classification method aligned to ASME BPVC, Section III, Division 5; the CNSC would evaluate proposed design standards for acceptability and would be likely to find common CNSC/NRC applications referencing ASME BPVC, Section III, Division 5 appropriate and acceptable for similar components.

7.1.4. Assessment of Impacts

The engineering rules applied to LWR pressure retaining components and supports are essentially identical for each identified code classification because both the NRC and the CNSC reference Section III of the ASME BPVC in guidance documents for the construction of pressure-retaining components and in-vessel support structures. In addition, the LWR components identified as ASME Class 1 are essentially identical because the NRC and CNSC have very similar criteria for Class 1 components (i.e., reactor coolant pressure boundary excluding small bore piping for NRC vs. component failure causes a loss of coolant accident). In addition, LWR piping segments penetrating primary containment that are identified as Class 2 through application of NRC guidance from RG 1.26 would also be identified as Class 2 through application of the Canadian N285.0 standard. The staff expects several other systems would also be identified as Class 2 under each regulatory organization's guidance based on the safety-significance of the function with significant overlap. Likewise, the identification of Class 3 systems and components would likely overlap because the NRC treats essentially all safety-related pressure retaining components as ASME Class 3 that have not been identified as Class 1 or 2, and the CNSC code classification process would align systems and components classified as moderately important to safety as Class 3. For the remaining systems and components classified as important to safety under both regulatory frameworks, both the CNSC regulatory guidance and the NRC guidance specify application of appropriate commercial industrial standards for the component type. Therefore, the overall LWR system and component code classification is expected to be essentially identical, with only minor deviation at lower safety-significance code classifications.

For advanced reactors, each regulator has guidance referencing the ASME BPVC for code classification. The NRC has developed draft guidance translating the outcome of the traditional classification process and the LMP safety classification methodology to code classifications. The code classification methods result in identification of safety-related core support structures and pressure-retaining components with the greatest safety-significance as Class A under ASME BPVC, Section III, Division 5 and the remaining safety-related SSCs (except RISC-3 under the 10 CFR 50.69 classification process) as Class B. Other important to safety SSCs with less safety-significance may be designed to commercial design standards with appropriate justification and additional special treatment requirements. The NRC and CNSC expect similar outcomes from each regulator's risk-informed safety classification process in determining the most safety-significant supports and pressure-retaining components, which would likely be subject to the Class A engineering design rules of the ASME BPVC, Section III, Division 5.

7.2. Electrical Distribution *(Future)*

7.2.1. Canada

7.2.2. United States

7.2.3. Similarities and Differences

7.2.4. Assessment of Impact

7.3. Instrumentation and Control *(Future)*

7.3.1. Canada

7.3.2. United States

7.3.3. Similarities and Differences

7.3.4. Assessment of Impact

7.4. Civil Structures *(Future)*

7.4.1. Canada

7.4.2. United States

7.4.3. Similarities and Differences

7.4.4. Assessment of Impact

8. Engineering Design Rules for Hazard Protection *(Future)*

8.1. Seismic Design Rules *(Future)*

8.1.1. Canada

8.1.2. United States

8.1.3. Similarities and Differences

8.1.4. Assessment of Impact

8.2. Fire Protection Design Rules *(Future)*

8.2.1. Canada

8.2.2. United States

8.2.3. Similarities and Differences

8.2.4. Assessment of Impact

8.3. Environmental Qualification and Hazard Barriers *(Future)*

8.3.1. Canada

8.3.2. United States

8.3.3. Similarities and Differences

8.3.4. Assessment of Impact

9. Interface with Standards Development Organizations (SDOs) *(Future)*

9.1. Canadian Interface with SDOs *(Future)*

9.2. United States Interface with SDOs *(Future)*

10. Conclusions *(Future)*

References

1. Memorandum of Cooperation on Advanced Reactor and Small Reactor Technologies between the Canadian Nuclear Safety Commission and the United States Nuclear Regulatory Commission, 15 August 2019, e-Doc 5976869.

2 Canadian Nuclear Safety and Control Act (S.C. 1997, c. 9). 1997. Available at https://laws-lois.justice.gc.ca/eng/acts/n-28.3/page-1.html.

3 U.S. Atomic Energy Act (AEA) of 1954, as amended. Available at https://www.govinfo.gov/content/pkg/COMPS-1630/pdf/COMPS-1630.pdf.

4 Work Plan for Joint Report on Classification of Structures, Systems and Components, Assignment of Engineering Design Rules, and Interactions with Standards Development Organizations, April 2022, e-Doc 6757940.

5 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities".

6 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants".

7 NRC RG 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors."

8 Nuclear Energy Institute (NEI) 18-04, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision 1, August 2019. (ADAMS Accession No. ML19241A472).

9 NRC SECY-23-0021, "Proposed Rule: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors," March 1, 2023 (ADAMS Accession No. ML21162A093).

10 Technology Inclusive and Risk-Informed Reviews for Advanced Reactors: Comparing the US Licensing Modernization Project with the Canadian Regulatory Approach, August 2021, (ADAMS Accession No. ML21225A101; CNSC e-Doc 6757940).

11 CNSC REGDOC-3.6, Glossary of CNSC Terminology, May 2022

12 NRC, NUREG/KM-0009, "Historical Review and Observations of Defense-in-Depth," April 2016.

13 International Atomic Energy Agency (IAEA) Safety Report Series No. 46, "Assessment of Defence In Depth for Nuclear Power Plants," February 2005.

14 CNSC REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants, Version 1.0, May 2014.

15 CNSC Strategy for Readiness to Regulate Advanced Reactor Technologies, December 2019.

16 CNSC REGDOC-2.4.1, "Deterministic Safety Analysis," Version 1.0, May 2014.

17 CNSC REGDOC-2.4.2, "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants," Version 1, May 2014.

18 IAEA SSR-2/1, "Safety of Nuclear Power Plants: Design," Revision 1, 2016. Available at: https://www.iaea.org/publications/10885/safety-ofnuclear-power-plants-design.

19 IAEA Publication, Report by the International Nuclear Safety Advisory Group, INSAG Series No. 10, "Defence in Depth in Nuclear Safety," 1996.

20 IAEA SSG-30, "Safety Classification of Structures, Systems and Components in Nuclear Power Plants," 2014.

21 NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition."

22 NRC, "Updated NRC Draft White Paper – Analysis of Applicability of NRC Regulations for Non-Light Water Reactors," (ADAMS Accession No.ML21175A287), July 2021

23 NRC, Draft Interim Staff Guidance: "Guidance for Performing the Review of a Technology-Inclusive Advanced Reactor Application - Review Roadmap," July 2021 (ADAMS Accession No. ML21190A012)

24 NRC Preliminary Draft Regulatory Guide DG-1413 (publicly released to support interactions with the Advisory Committee on Reactor Safeguards), "Technology-Inclusive Identification of Licensing Events for Commercial Nuclear Plants," (Proposed new Regulatory Guide 1.254, Revision 0) (ADAMS Accession No. ML22272A042).

25 NRC SRM SECY-22-0062 – Final Rule: NuScale Small Modular Reactor Design Certification (RIN 3150-AJ98; NRC-2017-0029), July 29, 2022 (ADAMS Accession No. ML22210A158)

26 NRC Regulatory Guide (RG) 1.232, "Developing Principal Design Criteria for Non-Light Water Reactors."

27 NRC, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," Federal Register, Vol. 60, No. 158: pp. 42622 (60 FR 42622), Washington, DC, August 16, 1995.

28 NRC SECY-86-0096, "Functional Containment Performance Criteria for Non-Light-Water-Reactors," (ADAMS Accession No.ML18115A157), September 28, 2018.

29 10 CFR Part 20, "Standards for Protection Against Radiation".

30 Nuclear Regulatory Commission (NRC) Policy Statement: Safety Goals for the Operation of Nuclear Power Plants, August 4, 1986 (51 FR 30028).

31 NRC RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance."

32 NRC SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems (RTNSS) in Passive Plant Designs," (ADAMS Accession No.ML003708005), Washington, DC, May 22, 1995.

33 IAEA, Technical Report IAEA-TECDOC-1570, “Proposal for a Technology-Neutral Safety Approach for New Reactor Designs,” September 2007.

34 CNSC REGDOC 2.6.1, “Reliability Programs for Nuclear Power Plants,” August 2017.

35 NRC SECY-95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems (RTNSS) in Passive Plant Designs,” (ADAMS Accession No.ML003708005), Washington, DC, May 22, 1995.

36 American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code (BPVC), Section III, “Rules for Construction of Nuclear Facility Components,” New York, NY.

37 ASME BPVC, Section VIII, “Rules for Construction of Pressure Vessels,” New York, NY.

38 ASME B31.1, “Power Piping,” New York, NY.

39 CNSC REGDOC-1.1.2, Licence Application Guide: Licence to Construct a Nuclear Reactor Facility

40 Canadian Standards Association, CSA N285.0, “General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants,” CSA Group, Toronto, Canada.

41 NRC RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” Revision 6.

42 American National Standards Institute (ANSI) and American Nuclear Society (ANS), ANSI/ANS-58.14-2011, “Safety and Pressure Integrity Classification Criteria for Light Water Reactors,” Washington, DC.

43 NRC Regulatory Guide 1.87, Revision 2, “Acceptability of ASME Code, Section III, Division 5, “High Temperature Reactors”,”(ADAMS Accession No.ML22101A263).