U.S. NRC – CNSC Memorandum of Cooperation

JOINT REPORT

concerning

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

January 2025

Approved by:

1/24/2025

X Jeremy Bowen

Jeremy S. Bowen
Director, Division of Advanced Reactors and No...
Signed by: JEREMY BOWEN

Jeremy Bowen, Director
Division of Advanced Reactors and Non-Power Production and Utilization Facilities
United States Nuclear Regulatory Commission

1/24/2025

X Michele M. Sampson

Michele M. Sampson
Director, Division of New and Renwed Licenses
Signed by: MICHELE SAMPSON

Michele M. Sampson, Director
Division of New and Renewed Licenses
United States Nuclear Regulatory Commission

2025-01-14

X SJ Eaton

Sarah Eaton
Director General, Directorate of Advanced Re...
Signed by: Eaton, SarahJane

Sarah Eaton, Director General
Directorate of Advanced Reactor Technologies
Canadian Nuclear Safety Commission

2025-01-24

X Melanie Rickard

Melanie Rickard
Director General, Directorate of Assessment a...
Signed by: rickard, melanie

Melanie Rickard, Director General
Directorate of Assessment and Analysis
Canadian Nuclear Safety Commission

Executive Summary

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 small modular reactor (SMR)[1] and advanced reactor[2] technologies. As part of the program of work under the MOC, the CNSC and the NRC issued CNSC-NRC Report: "Technology Inclusive and Risk-Informed Reviews for Advanced Reactors: Comparing the US Licensing Modernization Project [LMP] with the Canadian Regulatory Approach" (CNSC Approach/LMP Comparison). The CNSC Approach/LMP Comparison report concludes that 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 their regulatory findings and their respective independent National decisions required by law.

In recognition of the conclusions of the CNSC Approach/LMP Comparison report, 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 structures, systems, and components (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.

The regulatory frameworks in both Canada and the United States reflect extensive experience with the licensing and oversight of water-cooled reactors. This report addresses licensing reviews of SMRs and advanced reactors using the CNSC approach in Canada and the NRC traditional and LMP approaches in the United States. Both the CNSC and the NRC have regulatory 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 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 initiating events and 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 safety 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.

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[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] For this report, the advanced reactor designation refers to non-water-cooled reactors with design features that provide for enhanced safety relative to currently operating large, water-cooled reactors.

Comparison of Safety Analysis and Classification Approaches

Licensing Approach	CNSC	NRC Traditional	NRC LMP
Use of Probabilistic Analyses	Level 2 PSA - Complementary to Deterministic Analyses	Level 1 PRA - Confirmatory and for Identification of Risk Insights	Level 3 PRA - Foundational; Supported by Deterministic Analyses
Defense-in-Depth	Structured Review of Defence-in-Depth in Addition to Design Criteria	Established by Design Criteria and Special Regulations	Structured Review of Defense-in-Depth, Design Capabilities, and Programs
Anticipated Operational Occurrences	Initiating Event Frequency; Best-Estimate for Control System Mitigation; Conservative for Safety System Mitigation	Initiating Event Frequency; Best-Estimate for Control System Mitigation; Conservative for Safety System Mitigation	Sequence Frequency; Best-Estimate w/Uncertainty considering all Modeled SSCs
Design Basis Accidents	Sequence Frequency; Conservative Analyses or Best-Estimate w/Uncertainty	Guidance for Initiating Event Selection; Conservative Analyses	Sequence Frequency; Best-Estimate w/Uncertainty
Beyond-Design-Basis Events	Sequence Frequency; Best-Estimate for Design Extension Conditions	Special Regulations; Deterministic Requirements	Sequence Frequency; Best-Estimate w/Uncertainty
Classification of Mitigating SSCs	Applicant-Designated Classifications of Important to Safety SSCs; Safety System Classification for Standby Systems Selected to Mitigate Design Basis Accidents	Safety-Related for SSCs Selected to Mitigate Anticipated Operational Occurrences and Design Basis Accidents; Important to Safety for Defense-in-Depth SSCs	Safety-Related for SSCs selected to mitigate Design Basis Accidents and Certain Beyond-Design-Basis Events; Nonsafety-related with Special Treatment for Defense-in-Depth

The three licensing approaches can be viewed on a spectrum regarding the degree that probabilistic insights are incorporated into the safety analysis. The NRC traditional approach was established to evaluate safety using only deterministic evaluations, whereas the NRC LMP approach uses probabilistic methods to identify important initiating events and safety significant functions to prevent or mitigate the escalation of the event. The CNSC approach fits between the two NRC approaches using deterministic evaluations supported by probabilistic methods. Application of existing risk-informed safety classification methods to the NRC traditional approach would produce a safety analysis framework similar to the CNSC approach by incorporating probabilistic insights and a structured review of defense-in-depth.

The target values used for assessment of the radiological consequences of design basis accidents (DBAs) is a significant distinction between the approaches. The CNSC approach uses a fixed target value, whereas the NRC traditional approach uses two somewhat higher target values, with the higher value associated with lower frequency DBAs. The NRC LMP approach uses a target value that increases with decreasing frequency of occurrence and is higher than the CNSC value through most of the event sequence frequencies considered as DBAs. Under the NRC LMP approach, this difference is mitigated by the evaluation of mean consequences at the confidence extremes (5% and 95%) of the frequency uncertainty and the DBA evaluation considering only safety-related SSCs at the upper confidence (95%) bound of consequences. However, meeting the CNSC DBA dose consequence target value may necessitate additional measures beyond those required under the NRC approaches, such as additional systems or barriers to retain radionuclides or larger sites, to reduce the postulated exposure to sensitive populations.

The collaborative review by the CNSC and the NRC determined that the three regulatory approaches would support establishing comparable levels of safety, consistent with the similar overall safety objectives of the two regulatory bodies. The review also determined that the commonalities in the regulatory approaches would support joint reviews by the CNSC and the 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.

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, safety analysis methodologies, and appropriate assignment of engineering design rules in specific technical areas, including seismic qualification, quality assurance, and environmental qualification.

In this report, nine technical areas (as indicated in the following table) have been assessed for the application of engineering design rules and other specifications on the basis of safety classification. The technical areas were categorized into programmatic rules, design standards, and rules for protection from hazards. The collaborative review noted substantial alignment related to the scope of SSCs subject to application of engineering design rules and other specifications. However, safety classification alone rarely defines the complete application of engineering design rules.

Technical Areas Evaluated for Assignment of Engineering Design Rules

Programmatic Rules	Design	Protection from Hazards
Management for Quality	Pressure-Retaining Components	Seismic Design
Operational Reliability	Electrical Distribution	Fire Protection
	Instrumentation and Control	Environmental Qualification and Hazard Barriers
	Civil Structures	

The programmatic rules include quality assurance activities and activities affecting operational reliability, such as availability controls, performance and condition monitoring, maintenance effectiveness, and pre-service and in-service testing and inspection. Both the CNSC and the NRC provide for graded implementation of these programs where the extent of application, which may be in terms of scope, depth, or intensity, increases with safety significance. The NRC quality assurance requirements are prescriptive for activities affecting the safety-related functions of SSCs and, despite the graded approach, are somewhat more restrictive than comparable CNSC provisions governing quality and the facility management system. At lower levels of safety significance, the graded approaches to quality assurance of both the NRC and the CNSC have similar elements and are applied in a similar manner consistent with the safety significance of the activity. Provisions for availability controls, SSC monitoring, and testing/inspection are very similar in terms of scope of applicability and extent, but factors in addition to safety classification define the scope.

The design of SSCs focuses on the application of specific standards and the scope of applicability to SSCs within each classification. The scope of applicability is similar because both regulatory bodies have similar expectations regarding safety functions, determination of postulated initiating events, and hazards that important to safety SSCs must be protected against. The application of engineering design rules and specifications is very similar among pressure-retaining, electrical, and instrumentation and control components because the NRC and the CNSC rely on common design standards with international presence (i.e., the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), the Institute of Electrical and Electronics Engineers (IEEE), and the International Electrotechnical Commission (IEC)) for detailed technical specifications in these areas. The application of engineering design rules to structures other than pressure-retaining containment structures differ, and this may have an impact on risk and safety significance. However, both the CNSC and the NRC have regulatory provisions to demonstrate acceptable levels of safety through alternative means.

The protection against hazards (i.e., seismic, fire, and environmental hazards) is incorporated to a similar degree in both NRC and CNSC expectations for design of SSCs.

Overall, the collaborative review of the CNSC and the NRC licensing approaches and technical design expectations identified many areas of commonality that support the development of substantial technical information that could be shared in applications for SMRs or advanced reactors to both the CNSC and the NRC. These commonalities extend to the engineering design rules applied in the design, fabrication, and construction of components used in SMRs and advanced reactors. Although limited, differences in SSC classification and the resulting application of engineering design rules and specifications are most pronounced with respect to quality assurance measures applied to the most safety-significant SSCs, where NRC regulations are more prescriptive, and classification plays a significant role in the graded application of those measures. Another key area of difference is the effect of varying dose consequence target values under the various regulatory approaches for advanced reactor design basis accidents, which may affect the design of barriers to radionuclide release and siting of reactors, particularly if higher-consequence events are identified among LMP design basis events and higher frequency beyond-design-basis events. However, the CNSC and the NRC regulatory frameworks support the use of risk information to help ameliorate these differences.

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Abbreviations

ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act
ALARA	as low as reasonably achievable
ALWR	advanced light water reactor
AOO	anticipated operational occurrence
ARDC	advanced reactor design criteria
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 conditions
DL	defence level
DEC	design extension condition
EAB	exclusion area boundary
EME	emergency mitigating equipment
EPA	U.S. Environmental Protection Agency
F-C	frequency-consequence
I&C	instrumentation and control
ISI	in-service inspection
IST	in-service testing
IAEA	International Atomic Energy Agency
IDP	integrated decision-making process
ITS	important to safety
LBE	licensing basis event
LERF	large early release frequency
LMP	licensing modernization project
LRF	large release frequency

LUHS	loss of normal access to the ultimate heat sink
LWR	light-water reactor
MOC	Memorandum of Cooperation
NSCA	Nuclear Safety and Control Act
NEI	Nuclear Energy Institute
NITS	not important to safety
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
NSRST	nonsafety-related with special treatment
NST	nonsafety-related with no special treatment
PAG	protective action guide
PDC	principal design criteria
PIE	postulated initiating event
PRA	probabilistic risk assessment
PSA	probabilistic safety assessment
PSI	pre-service inspection
QA	quality assurance
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
RTNSS	Regulatory Treatment of Non-Safety System
SAFDL	specified acceptable fuel design limits
SARRDL	specified acceptable radionuclide retention design limits
SDA	standard design approval
SMA	seismic margin assessment/analysis
SR	safety-related
SRP	Standard Review Plan
SSCs	structures, systems, and components
SSE	safe shutdown earthquake
TEDE	total effective dose equivalent
TI-RIPB	technology-inclusive, risk-informed, and performance-based

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 the NRC 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 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. The activity aims to promote a mutual understanding of each organization's regulatory framework with a focus mainly on safety analysis expectations that are fundamental to the safety case that would support a licence application.

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 US Licensing Modernization Project [LMP] with the Canadian Regulatory Approach" [2] (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 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 that 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 acceptance criteria in mechanical, electrical, structural, and instrumentation and control disciplines.

[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] For this report, the advanced reactor designation refers to non-water-cooled reactors with design features that provide for enhanced safety relative to currently operating large, water-cooled reactors.

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) [3], 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) [4], its associated regulations, and 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 the NRC staffs.

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 applicants seeking approvals of SMR or advanced reactor designs. The reports will also assist the staffs of both the CNSC and the 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 [5] 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 that appropriate codes and standards are being developed, applied, and endorsed.

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 and specifications can be applied to ensure that they are designed, manufactured, constructed, installed, commissioned, operated, tested, inspected, and maintained with sufficient quality to fulfill 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 fulfill 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 NRC processes and regulatory criteria supporting safety classification are provided in Section 3 of this report.

Sections 6, 7, and 8 of this report address engineering design rules and specifications. Engineering design rules and specifications 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 and control of special processes during fabrication

The application of engineering design rules and specifications was sorted into the following three categories: (1) programmatic engineering design rules and specifications, (2) design of SSCs, and (3) 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:

- Programmatic engineering design rules:
    - Management for quality
    - Operational reliability
- Design of specific structures, systems, and components:
    - Pressure-retaining components and supports
    - Electrical distribution
    - Instrumentation and control
    - Civil structures
- Engineering design rules for hazard protection
    - 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 in-service testing and inspection, intended to maintain reliability.

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 Canadian Regulatory Framework

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 describes its regulatory approach and how it applies the NSCA to its regulatory oversight role in REGDOC-3.5.3 [6], "Regulatory Fundamentals." The Commission is ultimately responsible for licensing decisions in Canada.

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 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 is an indication that the regulatory framework provides flexibility for licensees to propose alternative means of achieving the intent of the requirement.

The licensing process consists of submission of a licence application, an assessment of the application by CNSC staff, and a decision by the Commission or a designated officer. The CNSC evaluates if and how well licensees meet regulatory requirements and CNSC expectations for the performance of activities within the scope of the license. Each licensee is required to conduct its activities in accordance with the licensing basis, which is defined as a set of requirements and documents for a regulated activity comprising the following:

1. The regulatory requirements set out in the applicable laws and regulations
2. The conditions and safety and control measures described in the licence, and the documents directly referenced in that licence
3. The safety and control measures described in the licence application and the documents needed to support that licence application

The CNSC's licensing regime includes a licence conditions handbook (LCH) as a companion to each licence. The purpose of the LCH is to clarify the regulatory requirements and other relevant parts of the licensing basis. The LCH includes the facility safe operating envelope, which is the set of limits and conditions that a nuclear power plant must be operated within to ensure compliance with the safety analysis that forms the basis for licensing of reactor operation.

2.1.2 United States' Regulatory Framework

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," [7] provides a 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 [8], "Licenses, Certifications, and Approvals for Nuclear Power Plants," that provides for conditional approval of sites, standardized reactor designs, manufacturing licenses, and construction and operation of a reactor as a combined license (COL). Each standard design approval (partial design), standard design certification (complete design excluding certain site-specific elements), or manufacturing license (authorization to construct a standard reactor design in a factory) could be incorporated by reference in an application for a COL at any number of sites that could accommodate the design.

The regulations in 10 CFR Part 50 and 10 CFR Part 52 reflect extensive experience with large light-water reactors (LWRs). Consequently, a number of regulations apply only to large LWRs, and other regulations have an implicit basis reflecting design characteristics of large LWRs. Therefore, certain regulations may be applicable but not technically relevant to SMRs or advanced reactors. These regulations include reasonable flexibility to address these situations by providing for specific exemptions from individual requirements when special circumstances are met. Special circumstances include conditions where compliance is not necessary to meet the underlying purpose of the regulation or if the underlying purpose of the regulation may be met in another way.

The AEA requires that each applicant for a license provide technical specifications, which include information on the nuclear material to be used, the location, and specific characteristics of the facility. Each license issued will include these technical specifications and any terms or conditions the Commission prescribes by rule or regulation to carry out the provisions of the AEA. The Commission is ultimately responsible for licensing decisions in the U.S.

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 and licensees. 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. These RGs may endorse consensus standards or provide information on methods of analysis or other information that forms part of the basis for a license.

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 RG 1.233 [9], "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 LMP principles and methodology, as described in Nuclear Energy Institute (NEI) 18-04 [10], 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 the 10 CFR Part 52 licensing process, with appropriate supported exemptions, to identify events to be considered in the licensing process, classify SSCs by the importance of the safety functions they perform, and ensure that design capabilities and programmatic requirements establish acceptable levels of defense-in depth.

In addition, NEIMA Section 103(a)(4) directs the NRC to "complete a rulemaking to establish a technology-inclusive, regulatory framework for optional use by commercial advanced nuclear reactor applicants for new reactor license applications" by December 31, 2027. The NRC staff proposed a draft 10 CFR Part 53 rulemaking (RIN-3150-AK31; NRC 2019-0062) [11], "Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants," in response to this direction. The Commission approved, in part, the draft proposed rule in the Staff Requirements Memorandum (SRM) for SECY-23-0021 [12]. In response to SRM SECY-23-0021, the NRC staff developed a proposed rule, designated as 10 CFR Part 53 [13]. The NRC staff anticipates that, under the proposed 10 CFR Part 53, the LMP methodology would be one acceptable way of identifying licensing basis events, classifying SSCs by safety function importance, and ensuring defense-in-depth.

2.1.3 Comparison of Regulatory Frameworks

The regulatory frameworks for the CNSC and the NRC have similar hierarchical structures, with an act by each country's legislative body establishing the highest-level requirements. The next level requirements are regulations developed by each regulatory body under the authority granted by the act. These regulations define the process and requirements that must be met to issue licenses or certificates.

The CNSC regulations are structured to provide high-level requirements related to safety and the licensing process. The CNSC provides expected implementation of those requirements for water-cooled reactors in Regulatory Documents. The expectations from Regulatory Documents that are incorporated into the facility license become obligations that the licensee must meet, but applicants may propose alternative means of meeting the higher-level regulations. This provides flexibility because the acceptability of the alternative is evaluated during the licensing review.

The NRC regulations for reactor operating licenses under 10 CFR Part 50 or 10 CFR Part 52 provide more detailed requirements, and several requirements reflect the NRC's long experience with licensing of large LWRs. Consistent with NEIMA, the NRC has endorsed a technology-inclusive, risk-informed framework that can be implemented with implementation of appropriate supported exemptions from 10 CFR Part 50 or 10 CFR Part 52 requirements, and the proposed 10 CFR Part 53 regulations provide for a future technology-inclusive, risk-informed framework. Alternatively, a traditional licensing approach under the 10 CFR Part 50 or 10 CFR Part 52 framework may be used, with appropriate supported exemptions for power reactors that are not cooled by light water.

Figure 1 provides a comparison of the regulatory hierarchy of documents within the CNSC and the NRC regulatory frameworks. Each framework includes an implementing act of the respective countries legislative body, regulations to implement the act and to provide for licensing of nuclear power reactors, a license for each reactor facility, and guidance documents and consensus codes and standards that support development of the detailed licensing basis supporting safe operation of the reactor facility. The primary distinction between the two frameworks is the less prescriptive regulations and more detailed license under the CNSC framework compared with more prescriptive regulations and less detailed license under the NRC framework, which is indicated by differences in the relative sizes of the regulation and licensing layers in the diagram. However, the legal obligations of an operating reactor license holder with respect to safety of a nuclear power plant are similar.

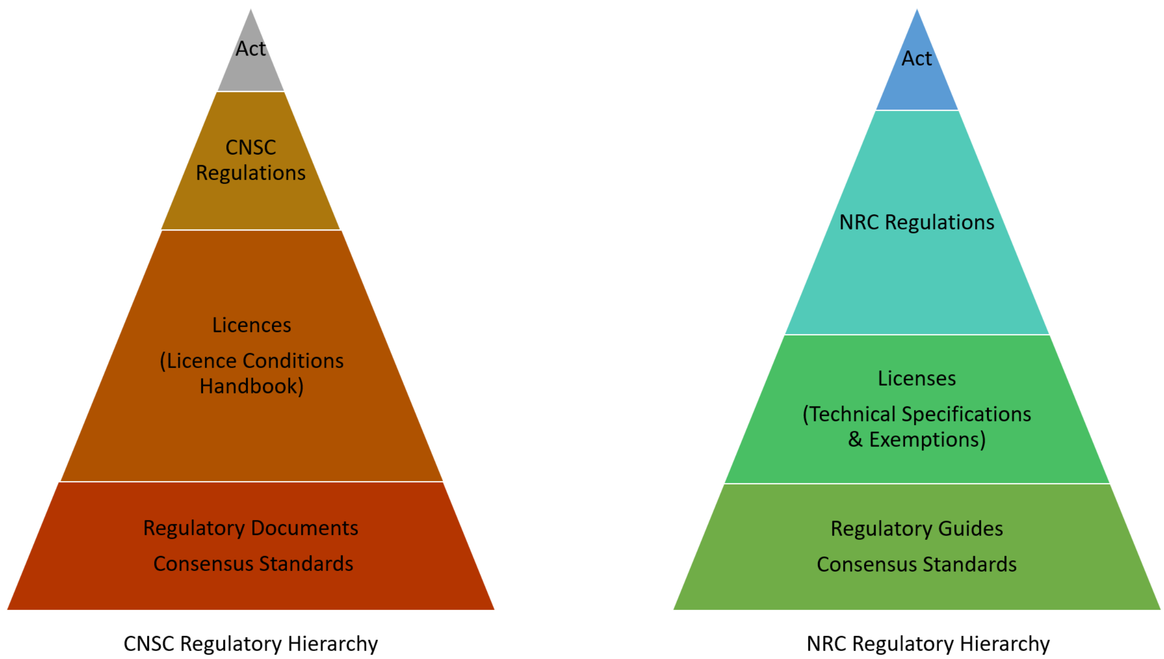

Figure 1: Comparison of Regulatory Frameworks

2.2 Reactor Facility Licensing

The regulatory frameworks for the licensing of new reactors by the CNSC and the NRC have both similarities and differences. The CNSC and the 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) describes the licensing processes in Canada and the United States for new nuclear reactors. The section provides an overview of preapplication interactions, application interactions, and regulatory safety objectives that are applicable to all new reactors.

2.3 Interrelationship of Design, Safety Analysis, and Regulatory Criteria

This report documents the results of the collaborative activities between the CNSC and the NRC on specific areas of the regulatory review process: the identification of events and conditions considered for the design of a reactor facility, the identification and classification of SSCs necessary to maintain safety through those events and conditions, and the assignment of engineering design rules and other specifications to those SSCs based, in part, on their safety classification in order to manage their reliability and availability. In accomplishing these tasks, the CNSC and the NRC staff considered the interrelationship of reactor design, safety analyses, and regulatory criteria.

Figure 2 shows this relationship as an iterative process beginning with an initial reactor design. The design includes initial information related to the capability, 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 for the identified events and conditions considered for the design. 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(s). 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 to the capabilities, reliability, or availability of select SSCs 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 and specifications to ensure that the reliability and availability of individual SSCs are commensurate with the importance of their safety functions.

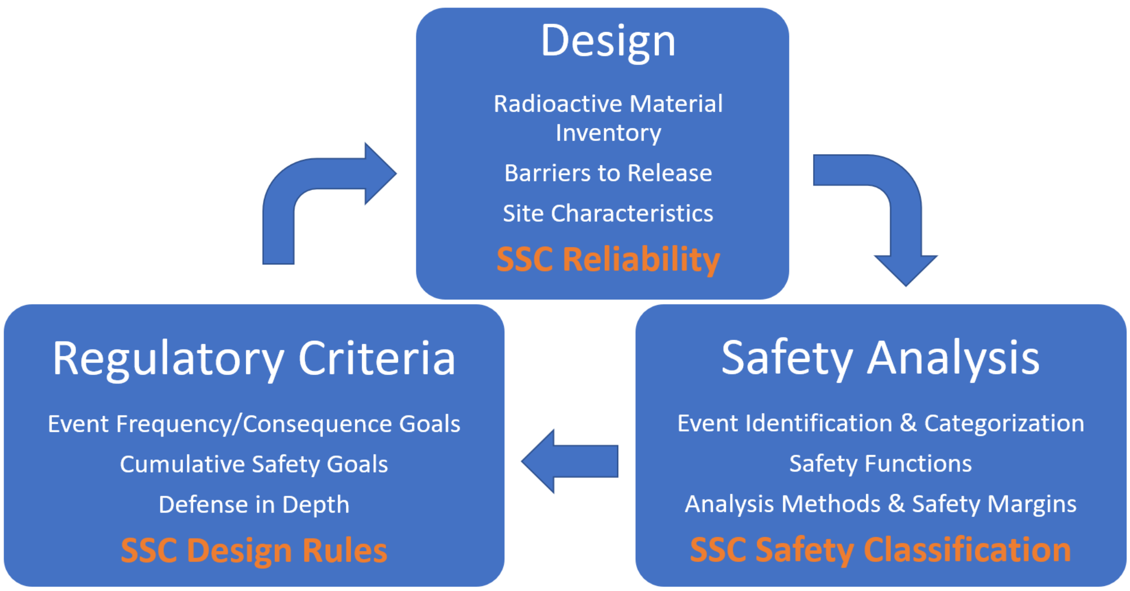

Figure 2: Interrelationship of Design, Safety Analysis, and Regulatory Criteria

2.4 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) [14], 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 begins with the conceptual 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.

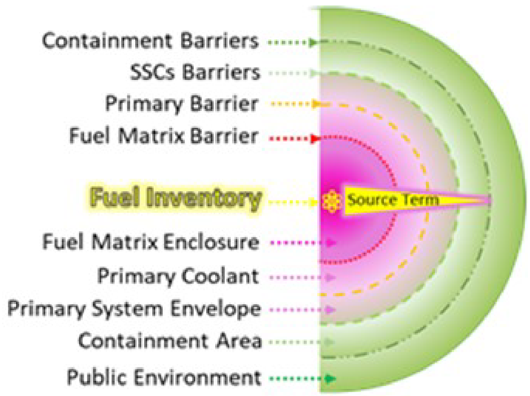

Figure 3: Fission Product Barriers

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 and conditions 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.

2.5 Defense-in-Depth Considerations

According to the NRC glossary [15], 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.

Figure 4, taken from NRC NUREG/KM-0009 [16], "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 [17], "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.

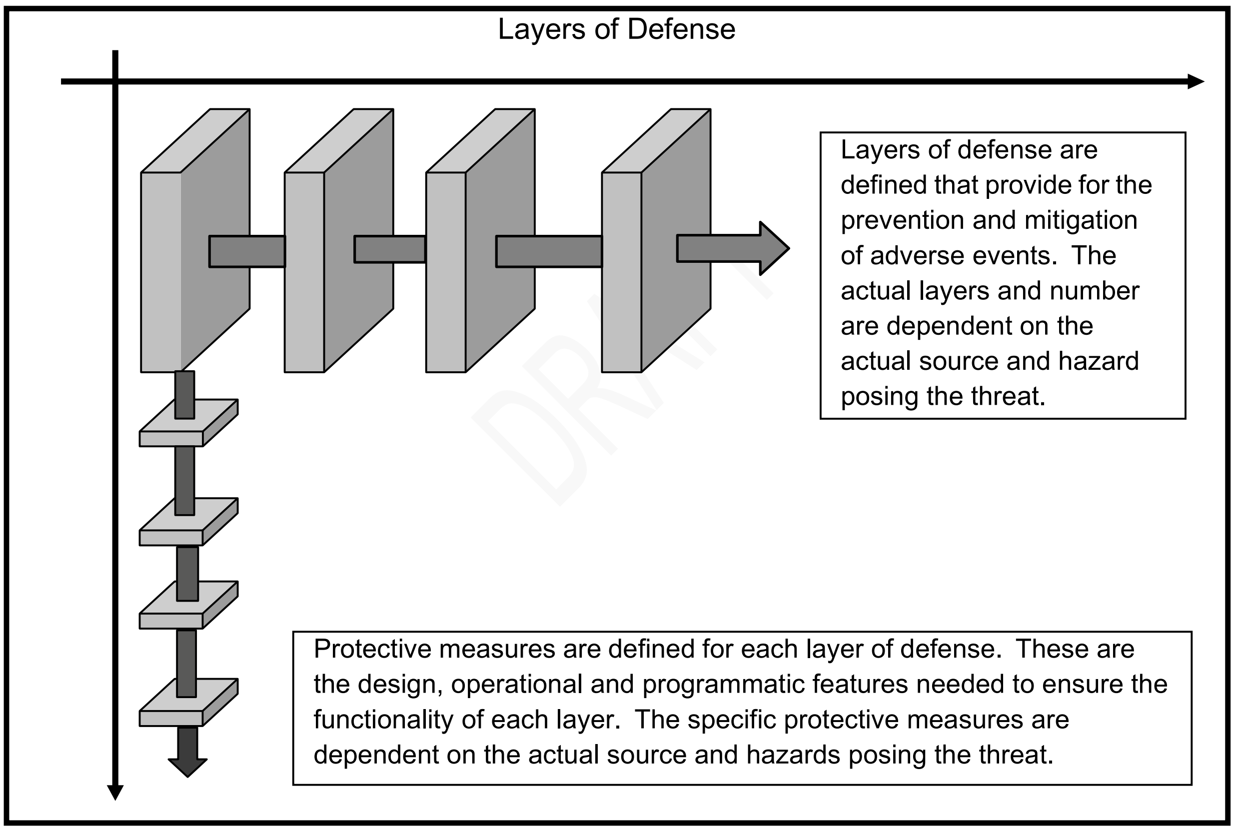

Figure 4: Concept of Defense-in-Depth

3 Approaches to Event Identification; Classification of Structures, Systems, and Components; and Design Rules and Specifications

This section provides a summary of the approaches established under the current regulatory frameworks in Canada and the United States for event identification; classification of SSCs based on the safety analysis; and management of the reliability and availability of SSCs through engineering design rules and specifications.

3.1 CNSC Approach

Section 2.4.1 of the CNSC Approach/LMP Comparison report 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 [18], "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" [19] 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.

As described in REGDOC-3.5.3, 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 9 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.

3.1.1 Safety Analysis

Applicants complete a deterministic safety analysis consistent with REGDOC-2.4.1 [20], "Deterministic Safety Analysis," and a probabilistic safety analysis consistent with REGDOC-2.4.2 [21], "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 NPP'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. *Figure 5* shows how the CNSC event categories relate to the plant operating states and are considered with respect to the nuclear facility design basis.

<table>
<tr><td colspan="2">Operational states</td><td colspan="4">Accident conditions</td></tr>
<tr><td rowspan="3">Normal operation</td><td rowspan="3">Anticipated operational occurrence</td><td rowspan="3">Design-basis accident</td><td colspan="3">Beyond-design-basis accidents</td></tr>
<tr><td colspan="2">Design-extension conditions</td><td>Practically eliminated conditions</td></tr>
<tr><td>No severe fuel degradation</td><td colspan="2">Severe accidents</td></tr>
<tr><td colspan="3">Design basis</td><td colspan="2">Design extension</td><td>Not considered as design extension</td></tr>
<tr><td colspan="6">Reducing frequency of occurrence ➔</td></tr>
</table>

Figure 5: CNSC Plant States

Source: Figure1: Plant States, from REGDOC-2.4.1.

All natural and human-induced external hazards that may be linked to significant radiological risk shall be identified. External hazards that the plant is designed to withstand are selected and classified as DBAs or DECs. External hazards may be evaluated and screened out of design when the hazard evaluation finds extremely low probabilities of occurrence, significant separation of hazard effects from the plant, adequate warning for protective actions, or effects bounded by another hazard included 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 describes the key objectives of the PSA as follows:

1. assess the impact of changes to procedures and/or components on the likelihood of core damage provide a systematic analysis in order to give confidence that the reactor facility's design will align with the fundamental safety objectives as established in IAEA No.SF-1 [22], *Fundamental Safety Principles*, including to protect people and the environment from radiation
2. demonstrate that a balanced design has been achieved; this can be demonstrated as achieved if no particular feature or postulated initiating event makes a disproportionately large or significantly uncertain contribution to the overall risk

3. provide confidence that small changes of conditions that may lead to a catastrophic increase in the severity of consequences (cliff-edge effects) will be prevented
4. provide assessments of the quantitative safety goals (the probabilities of occurrence for severe core damage states, and the assessments of the risks of radioactive releases to the environment) as defined in REGDOC-2.5.2, *Design of Reactor Facilities*, or as established in licensing basis for the facility
5. provide site-specific assessments of the probabilities of occurrence and the consequences of external hazards
6. identify facility vulnerabilities and systems for which design improvements or modifications to operational procedures could reduce the probabilities of severe accidents, or mitigate their consequences

The PSA complements the deterministic safety assessment.

3.1.2 Safety Functions

In Section 4.2 of 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

These functions apply to the reactor as well as fuel storage and handling locations.

3.1.3 Defence-in-Depth

Section 2.3.1, "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" [23] and International Nuclear Safety Advisory Group (INSAG) Series 10, "Defense in Depth in Nuclear Safety" [24]. Section 4.1, "Application of defence-in-depth," of REGDOC-2.5.2 includes additional guidance. *Table 1* 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 and high quality construction (e.g., appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication and plant construction, and consideration of 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 safety features, reliable control and protection systems, 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, and engineered design features that protect against DBAs and mitigate their consequences
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, including: • Robust containment design • Complementary design features to limit accident progression and mitigate consequences
Level 5	To mitigate the radiological consequences of potential releases of radioactive materials that may result from accident conditions	Emergency support facilities

The elements of Levels 2, 3, and 4 should be considered in the 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 are challenges, 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.

3.1.4 Design Basis Dose Assessment

In Section 2.2.1 of REGDOC-2.5.2, the CNSC specifies that 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 for a period of 30 days following the event as part of the deterministic safety analysis. The calculated dose shall be less than 0.5 millisievert (mSv) for AOO initiating events and 20 mSv for DBAs.

3.1.5 Safety Objective Assessment

In Section 2.2.2 of 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.

3.1.6 Design for Reliability

In Section 5.6 of REGDOC-2.5.2, the CNSC addresses considerations in design for SSC 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. 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 postulated initiating events (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.

Section 5.6 of REGDOC-2.5.2 specifies consideration of common-cause failures, separation, diversity, and independence between system groups that perform the same function, incorporation of fail-safe design features, allowance for outages, and limits on sharing of systems. In addition, each safety group shall perform its safety function in the presence of any single component failure, as well as all consequential failures resulting from that single failure and any failures that result in or result from postulated initiating events.

Section 6.4, “Means of shutdown,” of REGDOC-2.5.2 includes provisions specific to the reliability of the reactivity control function. 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 means of shutdown may not be completely independent.

3.1.7 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 for DID Level 2	Single Failure Affecting Safety System Group	No Single Failure (DID Level 4)
Analysis Methods	Best-Estimate - Control System Only for Most AOOs (DID Level 2); Conservative - Safety System Only (DID Level 3)	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 Accident Design Limits; No Unanalyzed Conditions	Evaluate Ability to Restore or Maintain Safety Functions
Dose Acceptance Criteria	0.5 millisievert (mSv)	20 mSv	Safety Goals Specify Cumulative Goals for Frequency and Magnitude of Accident Releases
Consequential Failures	Prevented to the Extent Practicable	Prevented to the Extent Practicable	Avoid Cliff-Edge Effects; Prevent Early Containment Failure

3.1.8 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.

Certain design criteria in REGDOC-2.5.2 are specified for safety systems or safety groups. These terms are used primarily in specifying systems subject to reliability criteria for design and certain instrumentation and control design rules. The CNSC Glossary in REGDOC-3.6 defines these terms in the following manner:

- safety system: A system provided to ensure the safe shutdown of a nuclear reactor or the residual heat removal from the core, or to limit the consequences of anticipated operational occurrences and design-basis accidents.

- safety group: The set of structures, systems and components designated to perform all actions required for a particular postulated initiating event, and to ensure that the specified limits for anticipated operational occurrences and design-basis accidents are not exceeded. The safety group may include certain safety and safety support systems and any interacting process system.

Therefore, safety systems and safety groups are specific sets of SSCs that are selected by the designer to ensure that safe shutdown is attainable and specified limits for AOOs and DBAs are not exceeded.

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 6*, which was drawn from IAEA SSG-30 [25], “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:

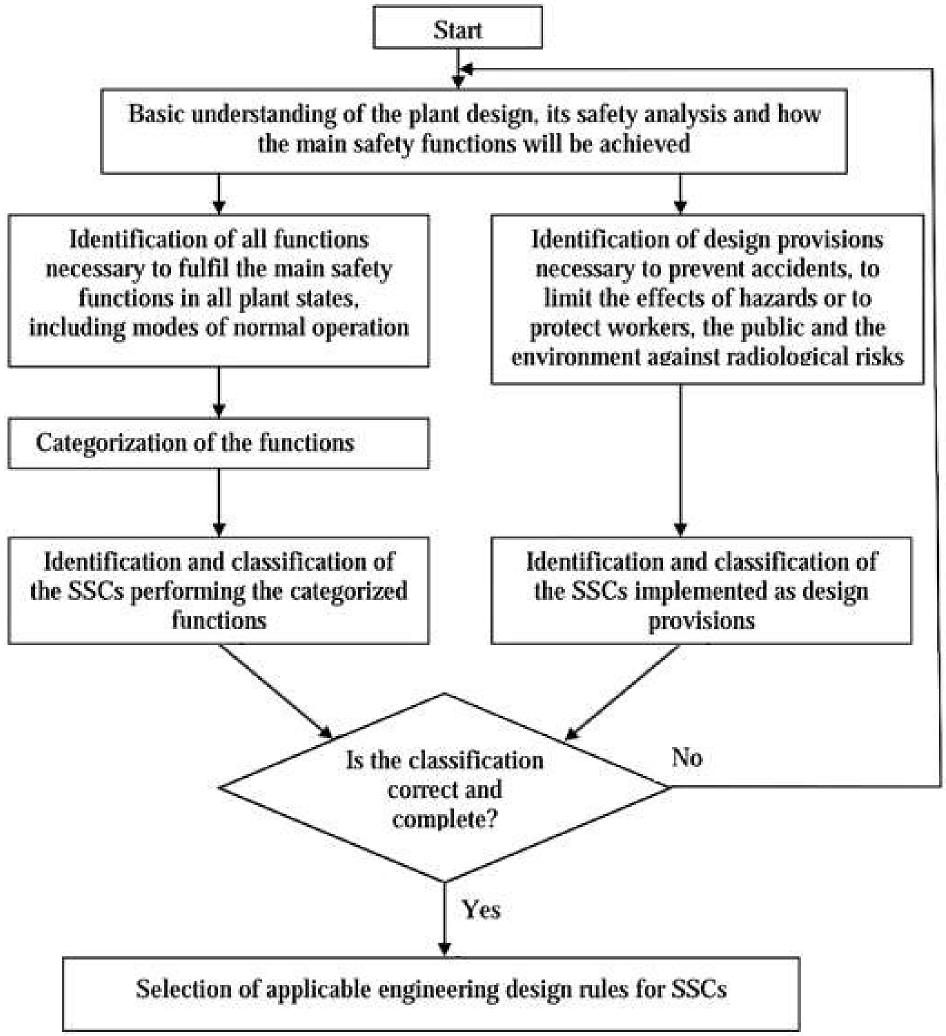

Figure 6: IAEA Safety Classification Flow Diagram

3.1.9 Assignment of Engineering Design Rules

Section 5.5, “Design rules and limits,” 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 and availability
- equipment qualification
- provisions for inspections, testing, and maintenance
- management system application (i.e., organizational quality assurance)

3.1.10 Application of Codes and Standards

Section 5, “Licence to Construct,” of the Class I Nuclear Facilities Regulations requires, in part, that an application for a licence to construct a nuclear power plant contain a description of the structures proposed to be built as part of the nuclear power plant, including their design and their design characteristics; and a description of the systems and equipment proposed to be installed at the nuclear power plant, including their design and their design operating conditions. This information is expected to include codes and standards applied to the design of these structures, systems, and equipment.

Expectations for codes and standards used in the design of nuclear power plant structures, systems, and equipment are specified in REGDOC-2.5.2. Section 4.5.3 of REGDOC-1.1.2 [26], "Licence Application Guide: Licence to Construct a Reactor Facility," provides the following guidance regarding application of codes and standards:

- The application should include declarations of the design's compliance with the codes and standards used.
- The applicant should provide an assessment, such as a gap analysis, if the codes and standards differ from those used in Canada.
- The application should include an assessment of the safety significance of any deviations from applicable codes and standards.

3.2 NRC Licensing Pathways and Generally Applicable Guidance

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 large LWRs. The NRC established regulatory guidance to standardize the format and content of applications with specific applicability to LWRs. Current guidance for applications using 10 CFR Part 50 is contained in RG 1.70 [27], "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," and RG 1.206 [28], "Combined License Applications for Nuclear Power Plants," issued June 2007, supported combined operating license applications under 10 CFR Part 52. The NRC staff issued Revision 1 to RG 1.206 [29] in October 2018, with the new title, "Applications for Nuclear Power Plants," which retained specific applicability to power reactors using LWR technology but clarified that the staff also considers it to be generally applicable to other types of reactors (e.g., non-LWRs).

The format and content guidance in these RGs aligns with the structure of NUREG-0800 [30], "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP). The SRP provides comprehensive guidance for completion of the safety analysis report, including identification of specific groups of initiating events and development of the associated safety analysis. Currently, 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 in effect 6 months prior to docketing of the application. However, proposed rulemaking[3] would remove this requirement and otherwise improve the consistency in requirements between 10 CFR Part 50 or 10 CFR Part 52. Items identified to improve the consistency of regulations in 10 CFR Part 50 or 10 CFR Part 52 include requirements for operating license applicants to submit descriptions of the plant-specific probabilistic risk

[3] 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 10 CFR Part 50 requirements with 10 CFR 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). The NRC staff proposed specific regulatory changes in SECY 22-052 (ADAMS Accession No. ML21159A055), and the Commission approved publication of a revised proposed rule in the associated SRM (ADAMS Accession No. ML24326A003).

assessment for all new reactors and to address conformance with technically relevant Three-Mile Island Action Plan Items listed in 10 CFR 50.34(f).

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 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. The NRC staff developed RG 1.253 [31] associated with the Advanced Reactor Content of Application Project, titled “Guidance for a Technology-Inclusive Content of Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors.” This RG endorses the nuclear-industry-developed content of application guidance in NEI 21-07 [32], “Technology-Inclusive Guidance for Non-Light Water Reactor Safety Analysis Report: for Applicants Utilizing NEI 18-04 Methodology,” Rev. 1, as one acceptable method for developing certain portions of the safety analysis report as part of an application for a non-LWR under 10 CFR Part 50 or a COL or certified design under 10 CFR Part 52.

The NRC staff has developed a set of nine accompanying guidance documents for other portions of these applications under an NRC-led Advanced Reactor Content of Application Project [33]. Among these guidance documents is a roadmap document, Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU) Interim Staff Guidance (ISG) 2022-01 [34], “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap,” dated March 2024. Appendix B to DANU-ISG-2022-01 includes an analysis of the applicability of NRC regulations to advanced reactors. This analysis concludes that 10 CFR Part 50 and Part 52 would support licensing of non-LWR advanced reactors with appropriately supported exemptions. Alternatively, advanced reactor applicants that have engaged in thorough preapplication activities may seek a rule of particular applicability or a Commission order establishing the regulations that apply to the review of the application.

3.2.1 Postulated Initiating Event Identification

The guidance that the NRC staff is developing in Draft Guide (DG) 1413 [35], “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. This table also shows the applicable licensing event categories and risk evaluation expectations for each pathway.

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:
➢ Conditions of normal operation, including 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 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</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[b]
• 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"><u>Notes:</u>
(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 licensing basis event (LBE) selection for non-LWRs licensed under 10 CFR Parts 50 or 52 and includes an expanded role for PRA beyond that currently required.
(b) Design basis events (DBEs) are defined differently for use in the LMP than as defined in 10 CFR 50.49; the LMP use of DBEs includes only event sequences with mean frequencies of 1×10^{-4}/plant-year to 1×10^{-2}/plant-year, taking into account the expected response of all SSCs within the plant regardless of safety classification and considering frequency uncertainty.
(c) Chapter 19 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," addresses the use of PRA and evaluation of severe accidents in LWRs. In the Staff Requirements Memorandum (ADAMS Accession No. ML24326A003) associated with SECY 2022-0052, "Proposed Rule: Alignment of Licensing Processes and Lessons Learned from New Reactor Licensing (RIN 3150-AI66)," the Commission directed the staff to publish a revised proposed rule in the Federal Register that would remove the requirement for applicants to include an evaluation of conformance with the SRP and add an item to require 10 CFR Part 50 operating license applicants to submit a description of the plant-specific PRA and its results.</td></tr>
</table>

Although a similar methodology can be used for postulated initiating event identification, 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 AOOs, DBAs, external events, and natural phenomena. An AOO is an initiating 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 event sequences are sorted into AOOs, DBEs, and BDBEs based on the mean frequency of the event sequence. The LMP DBAs are derived from the DBEs by evaluating the event sequence assuming only safety-related SSCs are available for mitigation.

External events and natural phenomena are treated differently than AOOs and DBAs (Traditional) or DBEs (LMP), which are typically initiated by internal events or hazards. For both the Traditional and LMP frameworks, a specific severity or intensity of the external hazard or natural phenomena is considered in the design of SSCs, often in the design of structures such that SSCs inside the structure are protected from the effects of these external events. The result is that external events considered within the design do not challenge the capability of SSCs necessary to perform essential safety functions.

3.2.2 Application of Codes and Standards

Regardless of safety analysis approach, NRC regulations require description of the codes and standards used in the design of the reactor. Specifically, 10 CFR 50.34(a)(1)(ii)(B) requires that stationary power reactor applicants for a CP describe the extent to which generally accepted engineering standards are applied to the design of the reactor. Comparable content of application requirements in 10 CFR Part 52 similarly requires applicants for DCs, SDAs, COLs, and MLs to describe the extent to which generally accepted engineering standards are used in the design of the reactor.

Furthermore, the content of application requirements of 10 CFR Part 50 and 10 CFR 52 require identification of the principal design criteria (PDC) used in the design of the facility and the relationship of the facility design bases to the PDC. 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 [36], “Guidance for Developing Principal Design Criteria for Non-Light Water Reactors,” to provide specific guidance for development of PDCs for non-LWRs, including providing a set of Advanced Reactor Design Criteria (ARDC) in Appendix A, “Advanced Reactor Design Criteria,” to RG 1.232. For all reactor types, GDC 1 is applicable and specifies that, where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. The corresponding criterion for non-LWRs, ARDC 1, is identical to GDC 1.

NRC regulations require application of specific codes and standards for design of certain reactor systems and components. For nuclear power reactors of all types with applications for construction permits, design approvals, design certifications, or combined licenses filed on or after May 13, 1999, 10 CFR 50.55a, “Codes and Standards,” requires that protection systems meet the requirements for safety systems in IEEE Std. 603–1991. In addition, 10 CFR 50.55a requires application of portions of the ASME Boiler and Pressure Vessel Code (BPVC) to the design, repair, inspection, and testing of pressure-retaining components in boiling and pressurized water-cooled reactors. Specific engineering design rules applicable to pressure-retaining components and instrumentation and control safety systems are discussed in Sections 7.1 and 7.3 of this report, respectively.

For other applications, the NRC has endorsed standards in RGs addressing specific applications. Use of those endorsed standards is voluntary, but use of alternate standards or design-specific approaches may decrease review efficiency. To reduce this impact, applicants should consider development of analyses identifying deviations from the endorsed standard and evaluating the significance of the deviation.

3.3 Traditional NRC Safety Analysis and SSC Classification

3.3.1 Safety Analysis

Deterministic 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 to be engineered into the facility and those barriers that must be breeched as a result of an accident before a release radioactive material to the environment could occur
  - an analysis of a postulated fission product release to evaluate the offsite radiological consequences
- An assessment of the design of the facility, including:
  - the 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 internal 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. These events are evaluated assuming only safety-related systems and components provide mitigation, and the analysis must consider 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 considered to assist in mitigation.

The PDC provide deterministic criteria for evaluating the overall design of the facility and the performance of SSCs. As described in Section 3.2.2 of this report, the GDC 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 to provide specific guidance for development of PDCs for non-LWRs. Deterministic design criteria included in the PDC help provide reasonable assurance that safety functions are adequately maintained following postulated events.

The GDC include acceptance criteria relevant to safety analyses for both AOOs and DBAs. For example, GDC 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, GDC 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 AOOs. The ARDC presented in Appendix A to RG 1.232 corresponding to GDC 20 and GDC 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 Risk Assessment

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" [37]. 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. 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.

Applicants for design certifications (DCs), combined licenses (COLs), standard design approvals (SDAs), and manufacturing licenses (MLs) under 10 CFR Part 52 must develop PRAs to support the applications. Consistent with the Commission's policy, applicants under 10 CFR

Part 50 are also expected to develop PRAs, and, as discussed in Section 3.2 of this report, 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 that the standard of adequate protection of public health and safety has been met. The design-specific PRA can be used to identify risk insights, identify severe accident vulnerabilities, and ensure that the quantitative health objective (QHOs) are met.

3.3.2 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 safety functions identified in the GDC include protection from hazards and natural phenomena; design of piping systems, structures, electrical systems, and instrumentation and control systems; design functions of protection and reactivity control systems; design of water systems for heat removal; design of containment and filtering systems to mitigate accidents; and design of radioactive waste systems to manage normal sources of radioactivity. The PDC described in RG 1.232 identify similar functions for advanced reactors and additional technology-dependent functions.

The NRC addressed the concept of functional containment and necessary performance in SECY 18-0096 [38], “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 control of leakage from physical building, the acceptance criteria would be established for each event classification to meet specified frequency-consequence targets (F-C) that evaluate a combination of barriers to radionuclide release, specified acceptable fuel design limits (SAFDL) focused on fuel cladding integrity for reactor centered events, or specified acceptable radionuclide release design limits (SARRDL) that ensure other SSCs perform as expected in retaining radionuclides. For AOOs, the barriers of concern are the fuel cladding and other integrated radionuclide retention features. For other DBEs, the barriers of concern may include additional physical barriers as well as other SSCs, such as contained fluids, that can reliably retain nuclides. This correlation of event classification to functional containment performance criteria is depicted in *Figure 7*.

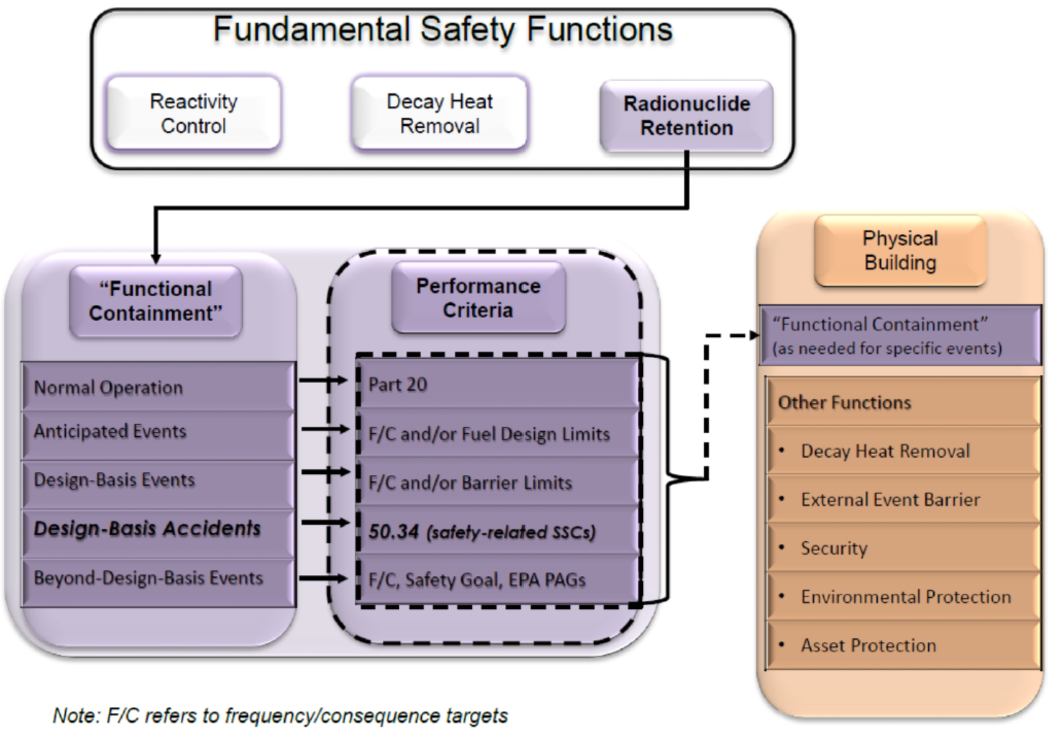

Figure 7: Functional Containment Performance Criteria

3.3.3 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[4];
- 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 mitigation of radionuclide release if the accident is severe;
- 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)[5] that manage specific conditions beyond the facility design basis; and

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[4] Several NRC design criteria define conservative performance capabilities for SSCs performing fundamental safety functions during:

- normal operations (i.e., GDC/ARDC 10, 13, 14, 17, 33, and 44)
- anticipated operational occurrences (i.e., GDC/ARDC 13, 15, 17, 20, 26, 33, and 34)
- design basis accidents (i.e., GDC/ARDC 16, 17, 20, 27, 34, 35, 38, 41,and 50)

[5] 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 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".

- 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.

Although a PRA is not an explicit requirement for applicants under 10 CFR Part 50, it plays a complementary role in ensuring adequate DID. It serves this role through identification of risk insights and severe accident vulnerabilities.

3.3.4 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. Potential applicants for such designs may consider preapplication engagement to identify appropriate mechanistic source terms to support the dose assessment and whether a request for exemption from regulatory requirements would be appropriate. Other DBEs with postulated releases are evaluated as DBAs to dose criteria that are a fraction (10 or 25 percent, based on a qualitative assessment of the DBE frequency) of the above dose criteria. Normal operational releases and AOOs are evaluated against the criteria of 10 CFR Part 20 [39], "Standards for Protection Against Radiation."

3.3.5 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 Early Site Permit, 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
- reasonable assurance that the health and safety of the public will not be endangered

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 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 NRC traditional approach has produced acceptable results, as confirmed by probabilistic analyses, demonstrating that the NRC safety objectives have been met. 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 PRA.

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 [40]. The NRC staff has established the following radiation exposure guideline values to meet its 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 risk assessments to confirm the NRC QHOs have been met.

3.3.6 Design for Reliability

The GDC and guidance for conduct of safety analysis ensure consideration of redundancy and diversity in design. The GDC include provisions for separation of protection systems from control systems and diversity in reactivity control systems. In addition, the 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.

The introduction to the GDC in Appendix A to 10 CFR Part 50 states that some specific design requirements relating to SSC reliability have not been developed, but their omission does not relieve applicants from considering these matters. The listed matters include:

1. Consideration of the need to design against single failures of passive components in fluid systems important to safety.
2. Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem, and the required interconnection and independence of the subsystems have not yet been developed or defined. (See Criteria 34, 35, 38, 41, and 44.)
3. Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents.
4. Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Criteria 22, 24, 26, and 29.)

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.

3.3.7 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, consistency with NRC guidance documents, 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 for LWRs

Technical Item	AOO Acceptance Criteria	DBA Acceptance Criteria
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 normal design limit	Within acceptable accident design limits
Fuel	Within Specified Acceptable Fuel Design Limits (SAFDLs)	Partial cladding oxidation and loss of cladding pressure integrity
Dose	10 CFR Part 20	Small fraction (10 or 25 percent) of Reference Value (25 Rem TEDE) for a Hypothetical Major Accident
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

3.3.8 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.

3.3.8.1 Traditional Safety Classification

For applicants for power reactor licenses under 10 CFR Part 50 or 10 CFR Part 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 [10 CFR] 50.34(a)(1) or [10 CFR] 100.11…, as applicable.

Several NRC regulations that establish engineering design rules or specifications for SSCs that rely on the classification of an SSC as safety-related or criteria similar to the definition of a safety-related SSC to establish the scope of applicability. Section 3.3.9 of this report lists regulations included among those that rely on the criteria used in the definition of safety-related SSCs to establish scope.

Certain terms in the definition of safety-related SSCs may require additional clarification. The NRC has defined the term ”design basis event” in 10 CFR 50.49(b)(1)(ii) as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure the functions specified in the definition of safety-related SSCs are accomplished. Therefore, safety-related SSCs should be considered to encompass the SSCs relied upon to shut down the plant from normal operating conditions or to mitigate the full scope of initiating events within the facility licensing basis, not only those SSCs relied upon to mitigate design basis accidents. The term “safe shutdown condition” has not been explicitly defined for use in determining the scope of safety-related SSCs. However, maintenance of acceptable levels of risk, which may be considered the product of the frequency and consequences for particular event sequences, for all initiating events necessitates that event sequences with high frequencies of occurrence have minimal consequences and events with significant consequences have very low frequencies of occurrence. Accordingly, as specified in the GDC of Appendix A to 10 CFR Part 50, AOOs are evaluated against criteria consistent with continued operation and DBAs are evaluated against less conservative criteria.

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 the 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 establish the appropriate scope of applicability for regulations that rely, in part, on the definition of “reactor coolant pressure boundary” for that purpose.

Although several NRC regulations use the term SSCs important to safety, that term has not been formally defined in regulations[6]. The introduction to the GDC in Appendix A to 10 CFR Part 50 describes important to safety in the following manner:

> The [PDC] 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

[6] 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).

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 use of the terms “nonsafety-related” or “not safety-related” mean only that the associated SSCs are not relied upon to remain functional during and following DBEs, as evaluated in the safety analyses, to perform the functions identified under the definition for safety-related. Risk- or safety significant equipment that does not meet the definition of safety-related would be considered important to safety.

3.3.8.2 Reliability Assurance Program and Regulatory Treatment of Nonsafety Systems

The NRC proposed the use of a reliability assurance program (RAP) as part of the licensing of LWRs relying on passive safety systems under 10 CFR Part 52. The purpose of the RAP is, in part, to ensure that the reactor is designed, constructed, and operated in a manner consistent with the reliability assumptions and capabilities modeled in the PRA supporting design certification. The application of the RAP is a two-step process, with the first stage applying during the design phase and being referred to as the design reliability assurance program (D-RAP) and the second stage involving application of existing operational programs to ensure reliability. The 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.

In the staff requirements memorandum (SRM) for SECY-94-084 [41], “Policy and Technical Issues Associated with the Regulatory Treatment of Nonsafety Systems in Passive Plant Designs,” the Commission approved the concept of the D-RAP and directed that the operational aspects of the reliability program be incorporated into existing programs. In the SRM associated with NRC SECY-95-132 [42], “Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems (RTNSS) in Passive Plant Designs (SECY 94-084),” the Commission approved the design reliability assurance program. The staff guidance for implementation of a D-RAP was primarily derived from Item E, “Reliability Assurance Program,” of Attachment 2 to SECY-95-132. Guidance for its implementation is presented in SRP Section 17.4 [43], “Reliability Assurance Program.”

The RAP applies to those SSCs 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. These methods include:

- quantitative risk evaluations based on fault trees and event trees
- other forms of risk evaluation, which may be quantitative or qualitative (e.g., fire-induced vulnerability evaluation or seismic margins analysis)
- severe accident evaluations
- industry wide operating experience
- expert panel(s)

The NRC has identified certain functions important to safety for advanced, passive LWRs that are not inherently functions performed by safety-related SSCs. These functions are applicable to SMRs and may be considered by applicants for advanced reactors in identifying safety-significant functions. In SRP Section 19.3, “Regulatory Treatment of Nonsafety Systems for Passive Advanced Light Water Reactors,’, Revision 0, June 2014 (ADAMS Accession No. ML14035A149), the NRC staff provides guidance on identifying nonsafety-related SSCs that perform risk-significant functions in a passive LWR plant design and are candidates for regulatory oversight. Consistent with SRP Section 19.3, the scope of the RAP for passive advanced LWRs should include components that perform the following risk-significant functions:

A. SSC functions relied on to meet beyond-design-basis deterministic NRC performance requirements such as those set forth in 10 CFR 50.62 for mitigating anticipated transients without scram (ATWS) and in 10 CFR 50.63 for station blackout (SBO).
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 core damage frequency (CDF) of less than 1×10^{-4} each reactor year and a large release frequency (LRF) of less than 1×10^{-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 above functions are not technology-inclusive, but the program could be voluntarily applied to SMRs and advanced reactors with appropriate modifications to the list of risk-significant functions for technological applicability. The cited regulations in Item A (i.e., ATWS and SBO regulations) apply to LWRs, and the CDF and LRF values in Item C are derived from the Commission Safety Goal Policy Statement criteria considering severe accident characteristics specific to LWRs. Advanced reactor applicants should consider other beyond-design-basis requirements, such as 10 CFR 50.155 for mitigation of BDBEs, that may apply to other reactor types licensed under 10 CFR Part 50 or 10 CFR Part 52. Specific safety goals for non-LWR technologies may be derived from the QHOs to define numeric criteria for Item C appropriate for the technology.

3.3.8.3 Risk-Informed Safety Classification

The NRC regulations provide for a voluntary risk-informed classification process 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 [44], “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance.” This RG endorses, with clarifications, NEI 00-04 [45], “10 CFR 50.69 SSC Categorization Guideline,” Revision 0.

The 10 CFR 50.69 regulation and associated guidance provide a risk-informed process to define those engineering design rules and special treatments that would be applied to SSCs considering both the safety classification under the traditional deterministic classification process and the safety significance of their functions. Implementation of this risk-informed classification process requires development of a PRA and an integrated decision-making process that incorporates DID concepts to characterize SSC safety significance.

An applicant for a CP or an OL under 10 CFR Part 50 or an applicant for an SDA, COL, or ML under 10 CFR Part 52[7] may implement this classification method provided that the applicant includes the required information in its application and the NRC approves implementation. The safety classifications are defined in the following manner:

- *Risk-Informed Safety Class (RISC)–1* structures, systems, and components (SSCs) means safety-related SSCs that perform safety-significant functions.
- *Risk-Informed Safety Class (RISC)–2* structures, systems and components (SSCs) means nonsafety-related SSCs that perform safety-significant functions.
- *Risk-Informed Safety Class (RISC)–3* structures, systems and components (SSCs) means safety-related SSCs that perform low safety-significant functions.
- *Risk-Informed Safety Class (RISC)–4* structures, systems and components (SSCs) means nonsafety-related SSCs that perform low safety-significant functions.
- *Safety-significant function* means a function whose degradation or loss could result in a significant adverse effect on defense-in-depth, safety margin, or risk.

Figure 8 shows the relative arrangement of these classifications.

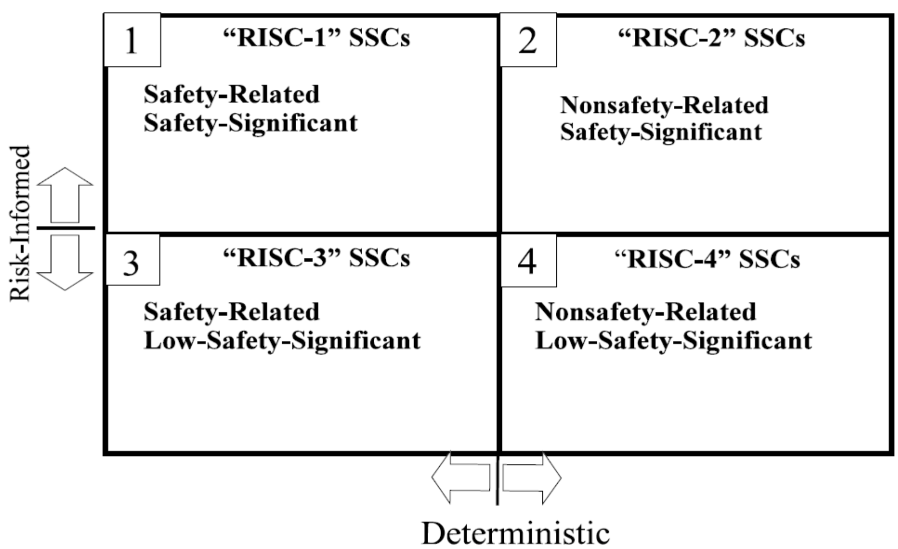

Figure 8: NRC Risk-Informed Classification Groups

Applicants approved to implement this classification structure may apply less rigorous requirements listed in 10 CFR 50.69 for RISC-3 SSCs than would otherwise be required for qualification, testing, and reporting for safety-related SSCs. However, implementation of this regulation includes alternate treatment requirements to ensure that RISC-1,RISC-2, and RISC-3 systems and structures perform their functions consistent with the categorization process assumptions. For RISC-3 SSCs, the regulation addresses inspection, testing, and corrective

[7] In the Staff Requirements Memorandum (ADAMS Accession No. ML24326A003) associated with SECY 2022-0052, "Proposed Rule: Alignment of Licensing Processes and Lessons Learned from New Reactor Licensing (RIN 3150-AI66)," the Commission directed the staff to publish a revised proposed rule in the *Federal Register* that would include an item to expand the applicability of 10 CFR 50.69 to include design certification applicants, and CP and COL holders.

actions as required elements of the alternative treatments replacing treatments normally applied to safety-related SSCs.

3.3.9 Assignment of Engineering Design Rules and Special Treatments

The GDC presented in Appendix A to 10 CFR Part 50 contain the minimum requirements for the PDC 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 GDC 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 and special treatments based on the significance of the safety function performed by the SSC.

Several regulations in 10 CFR Part 50 use the term "safety-related" or criteria identical to those used in the definition of *safety-related SSC* in 10 CFR 50.2 to establish, in full or in part, the scope of application for specific engineering design rules and special treatments. These requirements include:

- environmental qualification of electrical equipment, as required by 10 CFR 50.49
- in-service 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 that could cause undue risk to the health and safety of the public
- earthquake engineering design, as required by 10 CFR 50.34 and Appendix S to10 CFR Part 50

Certain regulations also apply engineering design rules to important to safety SSCs that do not meet the deterministic criteria for classification as safety-related. Requirements that apply to equipment described in regulations as nonsafety-related include:

- environmental qualification per 10 CFR 50.49 of nonsafety-related electrical equipment whose failure under postulated environmental conditions could affect satisfactory accomplishment of safety functions by safety-related equipment and certain post-accident monitoring equipment;
- performance or condition monitoring per 10 CFR 50.65 of nonsafety-related SSCs:
  - relied on to mitigate accidents or transients;
  - used in emergency operating procedures;
  - whose failure could prevent safety-related SSCs from performing their safety function; or
  - whose failure could cause a reactor scram.

- seismic design of important to safety SSCs whose failure under design earthquake conditions could affect satisfactory accomplishment of safety functions per GDC 2
- means to limit fire damage to SSCs important to safety so that the capability to shut down the plant safely is ensured per 10 CFR 50.48.

As discussed above, the NRC developed a policy for nonsafety-related SSCs that perform risk-significant functions in LWRs with passive safety systems in NRC SECY-95-132.
The RTNSS policy includes establishment of appropriate levels of reliability and availability through the RAP. Guidance for establishment of appropriate quality assurance controls for the design activities related to nonsafety-related RAP SSCs is included in Part II.U of SRP Section 17.5, "Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants." This guidance is technology-neutral and appropriate for important to safety SSCs that are not classified as safety-related.

The second stage of the RAP comprises the reliability assurance activities conducted during the operations phase of the plant. These activities are those necessary to ensure that the reliability and availability of RAP SSCs are maintained commensurate with their risk significance.

Administrative controls should be established for RAP SSCs to ensure availability of SSCs consistent with assumptions in the quantitative risk evaluations used as input to the RAP. Technical specification limiting conditions for operation that conform with 10 CFR 50.36(c)(2) provide availability controls for safety-related SSCs within the RAP and those nonsafety-related SSCs that contribute significantly to meeting criteria derived from the Commission Safety Goal Policy Statement. Other availability controls are appropriate for other nonsafety-related SSCs within the scope of the RAP. The reliability of RAP SSCs should be controlled using existing programs.

During the operations phase, regulatory requirements affecting the reliability of RAP SSCs include (1) 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants", (2) Quality Assurance Program requirements applicable to activities affecting the safety-related functions of SSCs, (3) quality controls for nonsafety-related RAP SSCs, and (4) in-service inspection, in-service testing, and surveillance testing programs. These measures are implemented to provide reasonable assurance that RAP SSCs do not degrade to an unacceptable level of reliability, availability, or condition. Similar considerations apply to alternate treatments applied pursuant to 10 CFR 50.69(d) to SSCs categorized as RISC-2 or RISC-3 for applicants and licensees approved to use that voluntary risk-informed categorization process.

3.4 NRC Risk-Informed, Technology-Inclusive Approach (LMP)

Section 2.4.2 of the CNSC Approach/LMP Comparison report 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 approach endorsed by this RG (referred to as the LMP), development of a PRA that includes evaluation of dose consequences to identified populations[8] is necessary. The LMP methodology includes the following processes:

- Systematic definition, categorization, and evaluation of event sequences for selection of licensing basis events (LBEs)[9], which include AOOs (mean frequency ≥10-2 per plant year), DBEs (mean frequency ≥10-4 and <10-2 per plant year), DBAs, and BDBEs (mean frequency ≥5 x 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

As described in Section 3.2 of this report, non-LWR applicants using the LMP with 10 CFR Part 50 or 10 CFR Part 52 regulations may seek regulatory flexibility through exemptions or other regulatory approaches. Appendix B to DANU-ISG-2022-01 includes an analysis of the applicability of NRC regulations to advanced reactors, considerations for demonstrating compliance with applicable regulations in new or unforeseen ways, and considerations regarding requests for specific exemptions from applicable regulations. Alternatively, advanced reactor applicants that have engaged in thorough preapplication activities may seek a rule of particular applicability or a Commission order establishing the regulations that apply to the review of the application.

3.4.1 Safety Analysis

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 9* (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.[10]

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 help inform development of design-appropriate PDC, considering the ARDC presented in Appendix A of RG 1.232. The LMP also includes a risk-informed evaluation of DID, which may identify additional SSCs that perform safety-significant

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[8] The specific evaluations of dose consequences to identified populations prescribed as part of the LMP approach are described in Section 3.4.5 of this report.

[9] 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 and other significant radionuclide sources associated with the licensed facility. LBEs include AOOs, DBEs, BDBEs, and DBAs.

[10] *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.

functions. Because the LMP for advanced reactors considers combinations of inherent, passive, and 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[11] as specified in the active ARDC.

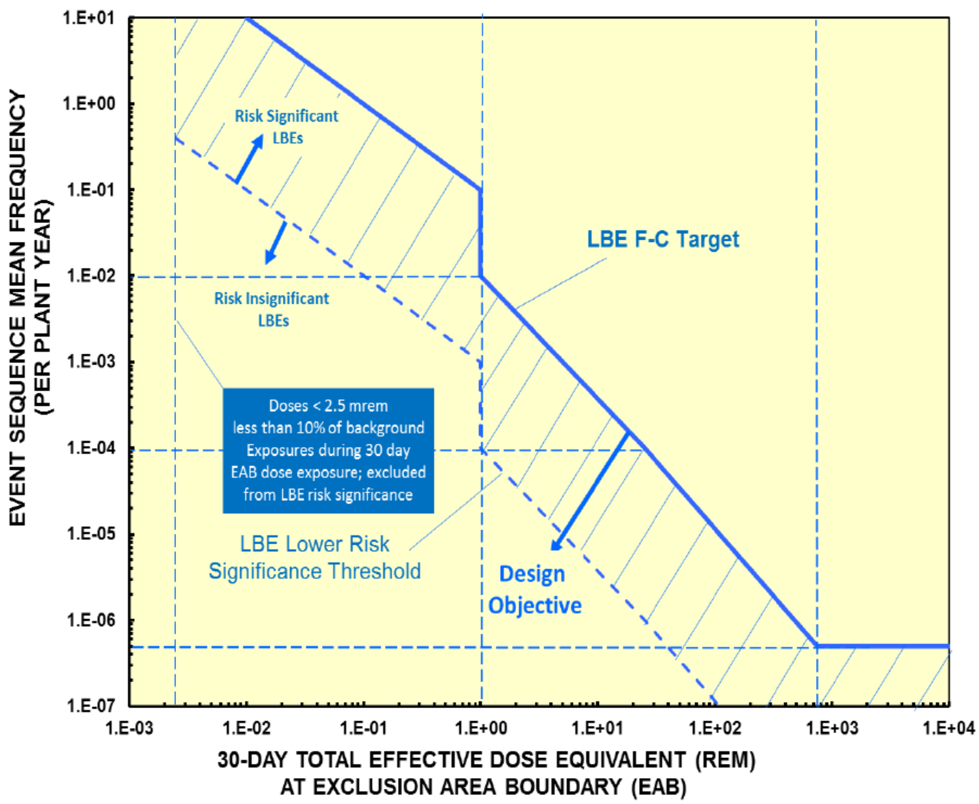


Figure 9: Licensing Basis Event F-C Target Curve

3.4.2 Safety Functions

The LMP process begins with the following technology-inclusive fundamental safety functions defined by an IAEA Technical Report [46]:

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

[11] In SECY 2003-0047 (ML030160002), "Policy Issues Related to Licensing Non-Light-Water Reactor Designs," the staff recommended that the single failure criterion be replaced with a probabilistic (reliability) criterion as part of incorporating a probabilistic approach in the establishment of a facility's licensing basis. The Commission approved of this recommendation in the SRM for SECY 2003-0047 (ML031770124). The Commission re-affirmed this position when it approved the LMP in the SRM for SECY 19-0117 (ML20147A504), "Staff Requirements – SECY-19-0117 – 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."

From these functions, the designer considers the unique characteristics of the reactor design to develop the necessary SSC functional capabilities to perform these functions. As described in Section 2.3 of this report, the design process is iterative, and the LMP approach includes steps to analyze the functional capabilities of SSCs against performance goals (i.e., the F-C target curve, cumulative risk, and DID) to identify changes and refine the design.

The LMP guidance states that the ARDC from RG 1.232 should be used as one input to initially establish PDC for a facility. As the design matures, the RSFs affirmed through the design process support identification of the design criteria that may be used to supplement or modify the ARDC. These PDC define the important safety functions considered in the facility design.

3.4.3 Defense-in-Depth

The LMP includes a specific process for consideration of DID principles in the classification of SSCs. *Figure 10* is from NEI 18-04 and shows this process:

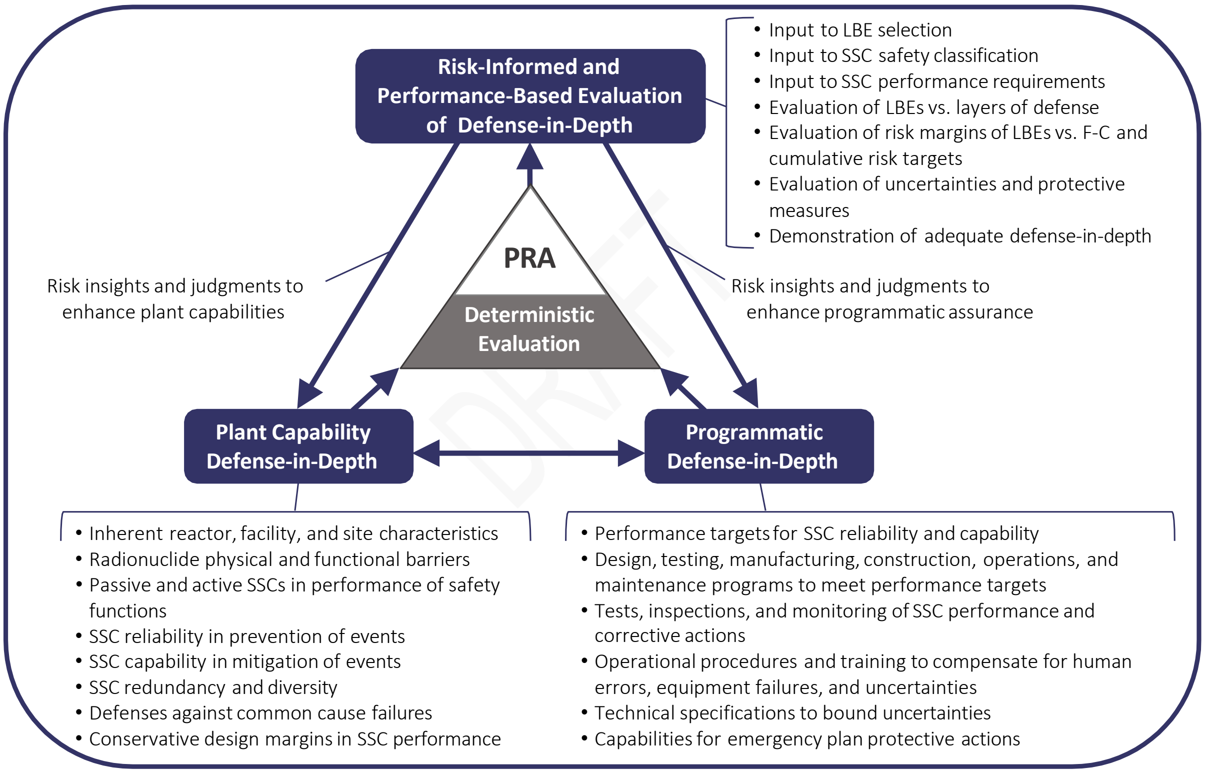


Figure 10: LMP Framework for Establishing DID Adequacy

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.

3.4.4 Design Basis Accident Dose Assessment

The LMP includes other elements that address conformance with NRC regulatory requirements, including requirements for evaluating the plant design features intended to mitigate the radiological consequences of accidents. These design features have been evaluated by assuming a fission product release from a “major accident,” which has generally been assumed to involve substantial meltdown of an LWR core with appreciable quantities of fission products released to containment.

For the evaluations of advance reactors 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. The applicant deterministically evaluates a set of DBAs derived from the DBEs to ensure that the dose reference values for a major hypothetical fission product release under 10 CFR Part 50 or 10 CFR Part 52 would be met using only the SSCs classified as safety-related and assuming that all other SSCs are not available. As noted in Section A-3.1, “Regulatory Guide 1.233 Approach (non-LWRs),” of Appendix A, “Alternative Approaches to Address Population-Related Siting Considerations,” to RG 4.7 [47], “General Site Suitability Criteria for Nuclear Power Stations,” applicants using the LMP may need to request an exemption from the content of application regulations if the most severe DBA does not involve the equivalent of significant core damage because these regulations require an assumed “major accident” to confirm that the calculated doses to individuals are below applicable reference values. Applicants that have significant preapplication engagement may alternatively seek a rule of particular applicability or a case-specific order to define application of the regulations to the technology considered in the application.

3.4.5 Safety Objective Assessment

The LMP provides a process for classifying identified LBEs into AOO, DBE, and BDBE categories based on the mean values of the event sequence frequency and evaluating the consequences against the F-C targets, which provide reference values to assess whether the design establishes adequate safety margins and DID. The LMP also provides for a comparison of the cumulative risk of all LBEs against the following three cumulative risk targets:

- The total frequency of exceeding a site boundary dose of 100 mrem from all LBEs should not exceed 1/plant-year. This metric is introduced to ensure that the consequences from the entire range of LBEs from higher frequency, lower consequences to lower frequency, higher consequences are considered. The value of 100 mrem is selected from the annual exposure limits in 10 CFR Part 20.
- The average individual risk of early fatality within 1 mile of the exclusion area boundary (EAB) shall not exceed 5×10-7/plant-year to ensure that the NRC safety goal QHO for early fatality risk is met.
- The average individual risk of latent cancer fatalities within 10 miles of the EAB shall not exceed 2×10-6/plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met.

3.4.6 Design for Reliability

Advanced reactors are anticipated to have a combination of inherent, passive, and active design features to perform essential safety functions. Under the LMP approach, these features are evaluated for reliability through both a PSA and a detailed evaluation of DID adequacy, which results in the identification of reliability performance targets that need to be maintained for the life of the plant.

3.4.7 Summary of Safety Analysis Acceptance Criteria

Table 5 summarizes SSC availability assumptions, analysis methods, and consequence target values for AOOs, DBEs, BDBEs, DBAs, and cumulative consequence assessments.

Table 5: NRC LMP Accident Analysis Summary

Initiating Event Category	SSC Availability	Analysis Methods	Consequence Target Note
AOO	SSCs as Modeled in PRA	Mechanistic source term; mean values of frequency and consequences derived from uncertainty analysis; and evaluate risk significance in adjacent event categories if frequency uncertainty band crosses boundary	Iso-risk line equivalent to 1 mSv (0.1 rem) per year and no more than 10 mSv (1.0 rem) per event
DBE			Line between 10 mSv (1 rem) at $1x10^{-2}$ per plant year and 250 mSv (25 rem) at $1x10^{-4}$ per plant year
BDBE			Continuation of DBE line slope to frequency of $5x10^{-7}$ per plant year
DBA	Safety-Related SSCs only	Mechanistic source term with upper bound consequences (i.e., 95th percentile of uncertainty distribution)	< 250 mSv (25 rem)
Cumulative	SSCs as Modeled in PRA	Integration of LBE mean frequencies and consequences	• Mean frequency of exceeding site boundary dose of 0.1 rem does not exceed 1 per year. • Average individual early fatality risk within 1 mile of EAB does not exceed $5x10^{-7}$ per plant year. • Average individual latent cancer fatality risk within 10 miles of EAB does not exceed $2x10^{-6}$ per plant year

Note: Dose consequences refer to 30-day exposure to an individual at the EAB unless otherwise specified.

3.4.8 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. The RSFs are a subset of safety-significant functions and are identified from the safety analysis as those functions modeled in the 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; or
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 11 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.

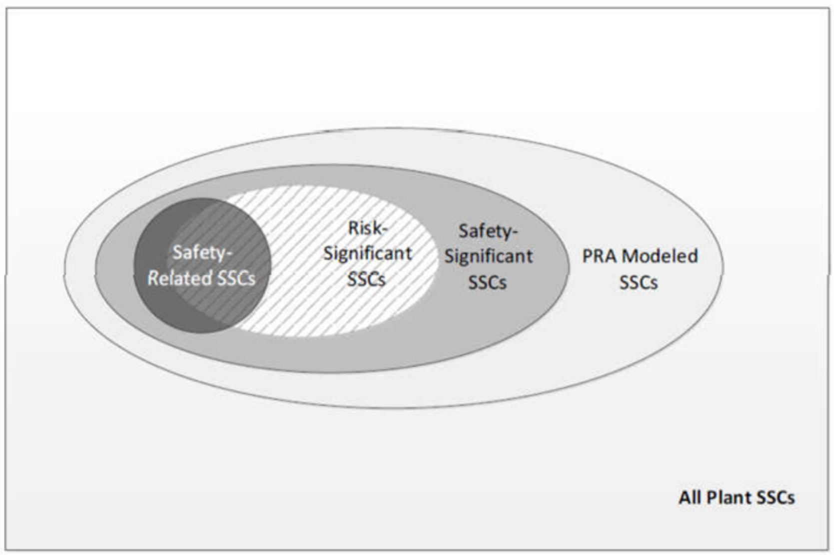
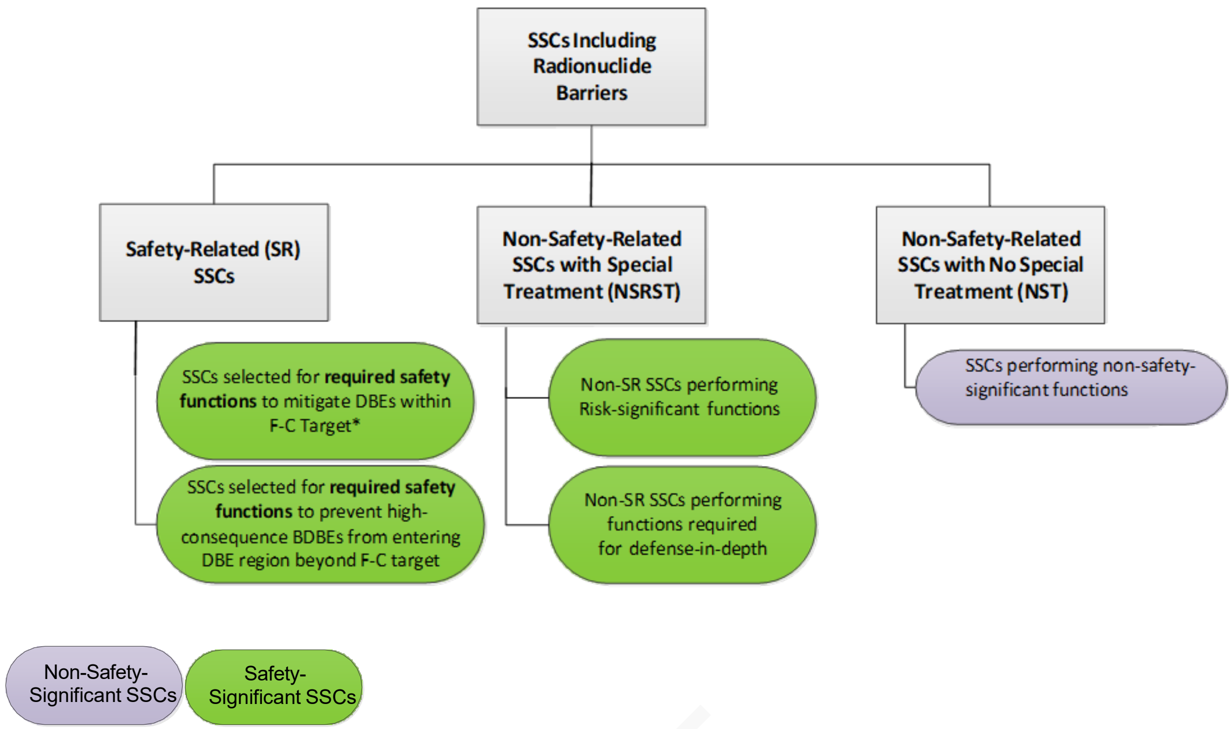


Figure 11: Relationship Between LMP Categories of SSCs



* SR SSCs are also relied on during DBAs to meet 10 CFR 50.34 dose limits using conservative assumptions

Figure 12: Relationship Between LMP SSC Classifications

3.4.9 Assignment of Engineering Design Rules

The LMP includes a process to assign engineering design rules, which NEI 18-04 describes as special treatments beyond normal industrial practices, to SSCs. The assignment is determined using the LMP safety classification and, for certain special treatments, the safety function of the SSC. Consistent with the traditional safety classification process, SSCs designated as safety-related would be subject to the engineering design rules for safety-related SSCs prescribed in 10 CFR Part 50 for quality assurance under Appendix B to 10 CFR Part 50, seismic qualification per Appendix S to 10 CFR Part 50, environmental qualification per 10 CFR 50.49, and availability and reliability monitoring per 10 CFR 50.65. The SSCs designated as NSRST are assigned special treatments considering the significance of the safety function using a design-specific RAP, which is similar to the RAP described in Section 3.3.8 of this report. *Table 6* is an abbreviated version of Table 4-1, “Summary of Special Treatments for SR and NSRST SSCs,” from NEI-18-04, Rev. 1.

Table 6: 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	10 CFR 50.65	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 (Applies to NST SSCs also)
Protection Against Design Basis External Events	In Scope	Not Specified
Equipment Qualification	10 CFR 50.49	Not Specified
Pre-Service and In-service 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 postulated initiating events, (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 (including how DID is considered in design), (2) the boundary values and specific acceptance criteria applied in the dose consequence analyses and safety assessments, and (3) the process for assigning safety classifications to SSCs.

4.1 Deterministic 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 report 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. Under the traditional approach, the GDC enhance the reliability of specified functions through deterministic criteria, such as the assumption of a single failure for many safety functions, provision of diverse methods of accomplishing certain functions (e.g., reactivity control and electric power sources), and separation of control and protection systems. The ARDC provide similar criteria for application to non-LWR designs under the NRC traditional approach. The CNSC provides similar design criteria in REGDOC-2.5.2. Appendix A to this report provides a comparison between the NRC GDC and the content of CNSC REGDOCs, in particular REGDOC-2.5.2.

In addition to design criteria, the CNSC provides a quantitative failure on demand criterion for systems performing important to safety functions. The LMP uses a fully risk-informed methodology to identify technology appropriate design criteria and 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 end. 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 PDC 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, deterministic functional classification criteria remain part of the classification process.

The CNSC approach, as described in REGDOC-2.4.1 and REGDOC-2.4.2, is more risk-informed than NRC's traditional approach to SSC classification in that PSA 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.

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

4.3 Safety Functions

Table 7 provides a comparison between the safety-significant functions as referenced in CNSC REGDOC-2.5.2, the NRC traditional approach as reflected by the GDC of Appendix A to 10 CFR Part 50, and NRC LMP as reflected by RG 1.233.

Table 7: 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 (GDC 26)	Reactivity and Power Control
Heat removal	Removal of heat from fuel	Residual heat removal (GDC 34) Emergency core cooling (GDC 35) Containment heat removal (GDC 38)	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) Fuel storage and handling and radioactivity control (GDC 60-64)	Radioactive material retention
Support and monitoring systems	Monitoring of safety-critical parameters to guide operator actions	Electric power (GDC 17) Equipment cooling water (GDC 44)	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 process of selecting PIEs and determining the functions that must be accomplished to satisfy the fundamental safety functions are expected 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.

The CNSC functions of shielding against radiation, control of operational discharges, and post-event monitoring are capabilities that do not directly align with the fundamental safety functions involving control of reactivity, control of heat removal, and containment or confinement of radioactive material. However, these functions align with NRC requirements described in the GDC of Appendix A to 10 CFR Part 50 (i.e., GDC-19, Control room; GDC 60, Control of releases of radioactive materials to the environment; and GDC 61, “Fuel storage and handling and radioactivity control”), the additional TMI-related requirements in 10 CFR 50.34(f), and 10 CFR 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents–nuclear power reactors.” These functions related to shielding and effluent control are important to safety but generally of lower safety significance because either failure of the function has relatively low consequences or the likelihood of needing the function is very low.

4.4 Defense-in-Depth

All safety analysis methods include measures for consideration of DID. Section 3.6 of the CNSC Approach/LMP Comparison report 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 DLs. Several NRC GDC define conservative performance capabilities for SSCs performing 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 DL 1 components. The components performing CNSC DL 2 functions generally correspond to control systems addressed within the NRC GDC. Specifically, NRC GDC 13 addresses control system functions and specifies that appropriate controls be provided to maintain important variables and systems within prescribed operating ranges. Several NRC GDC 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 anticipated operational occurrences”, specifies 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 framework specifies a significant degree of independence for the DL 2 components from safety systems that primarily perform DL 3 functions, which is comparable to NRC GDC 24, “Separation of protection and control systems,” regarding independence of control systems from those highly-reliable safety systems relied upon to mitigate both AOOs and DBAs. A key difference between the CNSC and the NRC approach is that the control systems that normally would respond to maintain plant parameters within bounds allowing prompt return to normal operation following anticipated upsets are not required to meet engineering design rules and specifications under the NRC framework unless a risk-informed SSC classification process is applied. The CNSC framework includes provisions to evaluate these control systems for importance and associate engineering design rules or other specifications with these control systems.

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. The NRC GDC include criteria to separate protection system functions, which are comparable to DL 3 functions, from control functions, which are comparable to DL 2 functions. In addition, several NRC GDC and certain regulations ensure that 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”). Several NRC GDC and certain regulations provide for the capability to mitigate radioactive material releases and limit further damage under design basis accident conditions (e.g., GDC 16, “Containment design”; GDC 38, “Containment heat removal”; GDC 41, “Containment atmosphere cleanup”; and GDC 50, “Containment design basis”).

The fourth level of defence includes 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. The NRC regulations at 10 CFR 50.44, “Combustible gas control for nuclear power reactors,” and 10 CFR 50.155, “Mitigation of beyond-design-basis events,” require these types of capabilities. The NRC traditional approach specifies design criteria for beyond-design-basis accident mitigation capability that is reasonably 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 the 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. 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 the 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.

Under traditional NRC licensing approaches, acceptable functional reliability has been achieved through the 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 that specific functions can be accomplished assuming a single failure and using either onsite or offsite power alone. An important distinction from the CNSC and the LMP approaches in the conduct of the evaluations and analyses is that the SRP guidance specifies reliance on safety-related SSCs to ensure that the acceptance criteria are satisfied following postulated transients and accidents. The CNSC and the LMP approaches generally consider availability of SSCs in a more risk-informed way, although the CNSC approach includes consideration of deterministic safety analysis that assumes a single failure affecting a safety group. In order to demonstrate compliance with NRC regulations, the LMP approach specifies evaluation of LMP DBAs assuming only safety-related SSCs are available to mitigate the DBAs and satisfy the dose assessment requirements of 10 CFR Part 50 or 10 CFR Part 52.

The three licensing approaches can be viewed on a spectrum regarding the degree that probabilistic insights are incorporated into the safety analysis. The NRC traditional approach was established to evaluate safety using only deterministic evaluations, whereas the NRC LMP approach uses probabilistic methods to identify important initiating events and safety-significant functions to prevent or mitigate the escalation of the event. The CNSC approach fits between the two NRC approaches using deterministic evaluations supported by probabilistic methods. Application of risk-informed safety classification methods to the NRC traditional approach would produce a safety analysis framework similar to the CNSC approach by incorporating probabilistic insights and a structured review of DID.

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 report 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.

NRC 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.

The target values used for assessment of the radiological consequences of DBAs is a significant distinction between the approaches. The CNSC approach uses a fixed target value, whereas the NRC traditional approach uses two somewhat higher target values, with the higher value associated with lower frequency DBAs. The NRC LMP approach uses a target value that increases with decreasing frequency of occurrence and is higher than the CNSC value through most of the event sequence frequencies considered as DBAs. Under the NRC LMP approach, this difference is mitigated by the evaluation of mean consequences at the confidence extremes (5% and 95%) of the frequency uncertainty and the DBA evaluation considering only safety-related SSCs at the upper confidence (95%) bound of consequences. However, satisfaction of the CNSC DBA dose consequence target value may necessitate additional measures beyond those required under the NRC approaches, such as additional systems or barriers to retain radionuclides or larger sites, to reduce the postulated exposure to sensitive populations.

Figure 13 provides a stylistic comparison of the safety analysis domains with respect to frequency of occurrence and consequences for the following event categories:

- AOO Baseline: All three analysis frameworks evaluate the best-estimate capability of normal control systems in responding to an AOO to maintain plant parameters within design limits for normal operation and to prevent escalation to an accident condition (CNSC Baseline DL2 analysis; NRC Traditional GDC-13 control system capability evaluation; and LMP AOO evaluation).
- AOO Conservative: The CNSC and the NRC Traditional approaches include a conservative AOO evaluation that assumes the failure of one or more control systems and mitigation by safety systems (CNSC) or safety-related SSCs (NRC traditional), assuming a single failure (CNSC Conservative DL3 AOO evaluation and NRC Traditional AOO analysis). This conservative analysis appears under the LMP approach as lower frequency event sequences beginning with an AOO PIE that would typically be evaluated against DBE F-C targets but could include evaluation against AOO F-C targets considering frequency uncertainty.
- DBA Analysis: The CNSC and the NRC Traditional approaches include a conservative DBA evaluation that assumes mitigation by safety systems (CNSC) or safety-related SSCs (NRC Traditional), assuming a single failure (CNSC conservative DL3 DBA evaluation and NRC traditional DBA analysis). This conservative analysis appears under the LMP approach as DBE event sequence that would typically be evaluated against DBE F-C targets but could include evaluation against AOO or BDBE F-C targets considering frequency uncertainty. The minimum set of SSCs necessary to satisfy the F-C targets for these DBEs constitute the majority of safety-related SSCs under the LMP.
- NRC Traditional Evaluations and Capabilities Mandated by Regulation: NRC regulations include requirements to evaluate specific LWR events as DBAs (i.e., GDC-28 LWR reactivity insertion accidents and LWR loss-of-coolant accidents to support evaluation of the emergency core cooling system per the requirements of 10 CFR 50.46).
- BDBE Evaluation: NRC regulations specify minimum capabilities for specific LWR BDBEs (e.g., ATWS per 10 CFR 50.62 and loss of all alternating current power per 10 CFR 50.63) and other capabilities applicable to all reactors (e.g., mitigation of beyond design-basis events per 10 CFR 50.155). The CNSC identifies complementary design features for BDBEs where necessary to prevent additional fuel damage, avoid cliff-edge effects, and prevent containment failure. The LMP considers BDBEs with sequence frequencies, including uncertainty, as low as 5×10^{-7} per plant year. With consideration of frequency uncertainty, SSCs necessary to ensure BDBE consequences satisfy the F-C Target in the DBE frequency range are also classified as safety-related.
- NRC Maximum Hypothetical Accident (MHA) Siting Evaluation per 10 CFR 50.34(a)(1): This regulation requires evaluation of a major accident involving a fission product release hypothesized for purposes of site analysis. This evaluation under the NRC traditional approach generally assumes a substantial meltdown of an LWR core with release into containment for LWRs and may include comparable fission product release scenarios for advanced reactors. The LMP approach addresses this requirement through evaluation of DBA dose consequences considering only safety-related SSCs to act to prevent the progression of the accident or mitigate its consequences. As noted in

Section 3.4.4 of this report, advanced reactor applicants may need to request an exemption from the NRC content of application regulations if the most severe DBA does not involve the equivalent of significant core damage because these regulations require an assumed “major accident” to confirm the calculated doses to individuals are below applicable reference values.

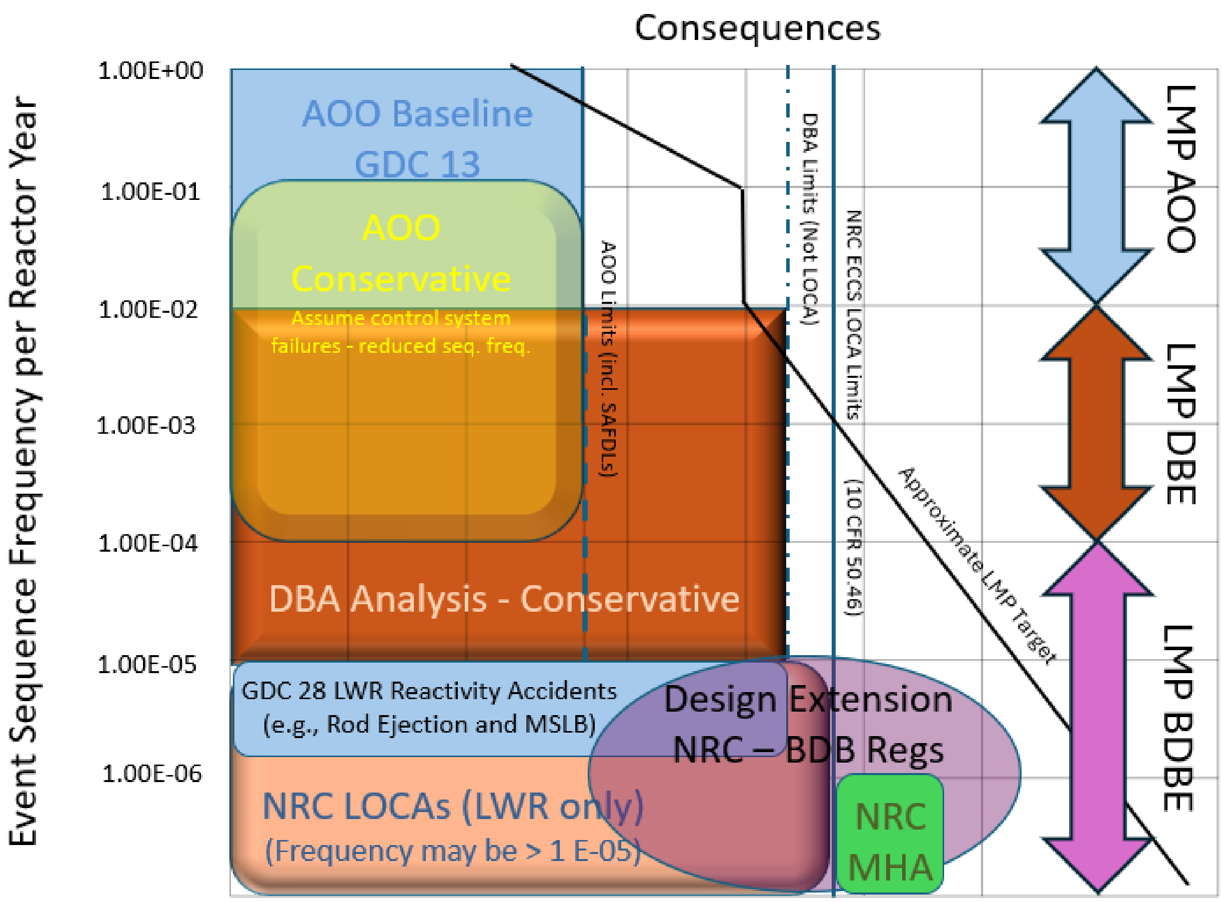


Figure 13: Comparison of Safety Analysis Domains

Table 8 provides a summary of the safety analysis approaches, regulatory considerations, and more detailed acceptance criteria associated with reactor safety analysis approaches under the CNSC, NRC Traditional, and the LMP frameworks.

Table 8: Comparison of Licensing Paths and Safety Analysis Criteria

Safety Analyses and Event Groups	CNSC Regulatory Approach	NRC Traditional Licensing (10 CFR Part 50 or 52)	NRC Risk-Informed, Technology-Inclusive Licensing (LMP)
Deterministic Safety Analysis	Required by regulation; REGDOC-2.5.2 includes deterministic design criteria	Required by regulation; includes development of principal 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	Initiating Event $\geq 10^{-2}$ 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	Best-estimate with control system operation (DL-2) and conservative using only safety systems w/single failure (DL-3) - SSCs within design, safety functions met, and no escalation	GDC 13 control system review; conservative using safety-related systems w/single failure - fuel design limits met, safety functions met, and no escalation	Meet SSC functional design criteria and prevent escalation – best-estimate with consideration of uncertainty in frequency
AOO Dose Criteria	≤0.5 mSv (0.05 rem) per event	Part 20 dose limit - ≤1 mSv (0.1 rem) per event	F-C Target – Annualized dose below Part 20 limits
DBA / DBE Frequency	Event sequence ≥10-5 and <10-2 per reactor year	Event/accident selection guidance	Event sequence ≥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 w/single failure; LWRs - 10 CFR 50.46	F-C Target met for each DBE w/all SSCs; 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]; DBA dose limits set at 25 - 62.5 mSv	DBAs mitigated by SR SSCs only ≤250 mSv (25 rem) TEDE[Note B]
BDBE Frequency	Below 10^{-5} occurrences per reactor year – best-estimate	Deterministic design criteria for certain events	From 5×10^{-7} to 10^{-4} per plant year – w/uncertainty
DEC or BDBE Constraints	Complementary design features enhance mitigation	QHOs considering PRA uncertainty	F-C Target per event
Cumulative Constraints	CNSC Safety Goals[Note C]	Targets derived from QHOs	Targets derived from public dose limit in 10 CFR Part 20 and QHOs

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 design basis accident.

B Per 10 CFR 50.34(a)(1): A deterministic evaluation of a hypothetical or possible event involving substantial core damage and release of fission products to containment 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 [determines exclusion area boundary per 10 CFR 100.11]; 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 frequency and small release frequency $<10^{-5}$ per reactor year; large release frequency $<10^{-6}$ per reactor year

4.6 Safety Classification

Newer 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 the NRC approaches determined that similar outcomes in the classification of SSCs could be expected [see Section 3.5 of CNSC Approach/LMP Comparison report]. The resulting classification groups would be similar because the licensing processes have shared characteristics that support accurate ranking of SSCs by safety importance.

The NRC Traditional 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. Under the NRC traditional approach, the applicant determines SSC safety classification solely based on the functions performed by the SSC unless the risk-informed classification process per 10 CFR 50.69 is approved.

The CNSC approach is informed by both the deterministic safety analysis and probabilistic safety analysis. The CNSC also provides for an applicant-defined classification scheme based on risk for all SSCs considered important to safety. However, REGDOC-2.5.2 defines SSCs important to safety as including the following SSC categories:

1. safety systems
2. complementary design features
3. safety support systems
4. other SSCs whose failure may lead to safety concerns (e.g., process and control systems)

The CNSC defines the first three terms in REGDOC-3.6, “Glossary of CNSC Terminology,” in the following manner:

> *safety system* - A system provided to ensure the safe shutdown of a nuclear reactor or the residual heat removal from the core, or to limit the consequences of anticipated operational occurrences and design-basis accidents.
>
> *safety support system* - A system designed to support the operation of one or more safety systems.
>
> *complementary design feature* - A design feature added to the design as a standalone structure, system or component (SSC) or added capability to an existing SSC to cope with design extension conditions.

These definitions of *safety system* and *safety support system* closely match the NRC definition of a safety-related system, but these terms represent only part of the important to safety SSCs related to licensed activities. The NRC treats SSCs performing functions similar to those performed by a CNSC *complementary design feature* as important to safety but not safety-related.

An important distinction in the classification of SSCs under CNSC requirements and guidance compared to the NRC classification systems (Traditional or LMP) 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 the LMP are those functions both necessary and sufficient to (1) maintain the consequences of a postulated DBE or a high-consequence BDBE that enters the DBE range 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 the 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 Safety 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 assumptions identified in *Table 9* were used.

Table 9: 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.

Topic	Assumption	Basis
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 Section 3.3.4, "Design Basis Accident Dose Assessment," 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 10*, 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 4.2 of REGDOC-2.5.2 specifies that fundamental safety functions (i.e., those listed in *Table 7* 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 that 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 DLs 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 SSCs responding to design basis accidents having the highest classification among important to safety SSCs. These functions include, for example, maintaining reactor core geometry or maintaining reactor heat removal during and following an accident, where failure could result in unacceptable consequences because fundamental safety functions would not be accomplished.

Considering the assumptions listed in *Table 9* 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 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 the 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. Appendix A to this report provides a comparison of the NRC GDC to the CNSC design criteria present in REGDOCs. 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 3.3.3 of this report, 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 SSCs, 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 3.3.8 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 the NRC and the 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 3.1.7 of this report), 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 nonsafety-related with special treatment (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 would result 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 the NRC staffs 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 the 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 10* outlines expected actions, given the assumptions outlined in *Table 9*, to align safety classifications when either regulatory framework is used for the initial SSC safety classification process:

Table 10: 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 support DID confirmation.]
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 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	Conformance expected based on assumptions in *Table 9*. Justify any deviations from REGDOC-2.5.2 requirements, leveraging information from risk informed processes (for example, NRC RISC or LMP evaluation processes) in accordance with section 9 of REGDOC-2.5.2.

Process	Leveraging of CNSC Classification Framework Outcome for an NRC Application	Leveraging of NRC Classification Framework Outcome for a CNSC Application
	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.	For the traditional approach, demonstrate conformance with defense levels by leveraging information from application of the principal design criteria and any risk-informed classification processes. For the LMP approach, the DID evaluation for safety function capability should provide the necessary information to demonstrate defense levels are met.
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 approaches. 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. CNSC safety significance classification process may inform assignment of design rules to SSCs classified as NSRST under the LMP.	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.

Table 11 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 11*). 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 the assignment of engineering design rules commensurate with the safety significance of the SSC.

Table 11: 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 for Defense-in-Depth
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 primary function of safety classification is to map certain engineering design rules and specifications to individual SSCs based on the safety classification. The CNSC expects safety systems to meet certain reliability targets. Within the broad important to safety classification, the CNSC permits applicants to propose safety classifications and how those classifications translate into graded application of engineering design rules and other specifications for approval. Conversely, both NRC approaches recognize certain prescriptive requirements linked to classification as a safety-related SSC. Similar to the CNSC approach, NRC guidance under both the Traditional and LMP approaches permit more flexibility in defining the extent certain engineering design rules and specifications apply to other important to safety SSCs. Risk-informed structures have been established as part of the risk-informed safety classification process defined in 10 CFR 50.69, as part of the Reliability Assurance Program in SRP Section 17.4, and as part of the LMP, which references SRP Section 17.4. *Figure 14* provides a perspective of various safety classification terms used under each of the three regulatory approaches arranged relative to safety significance.

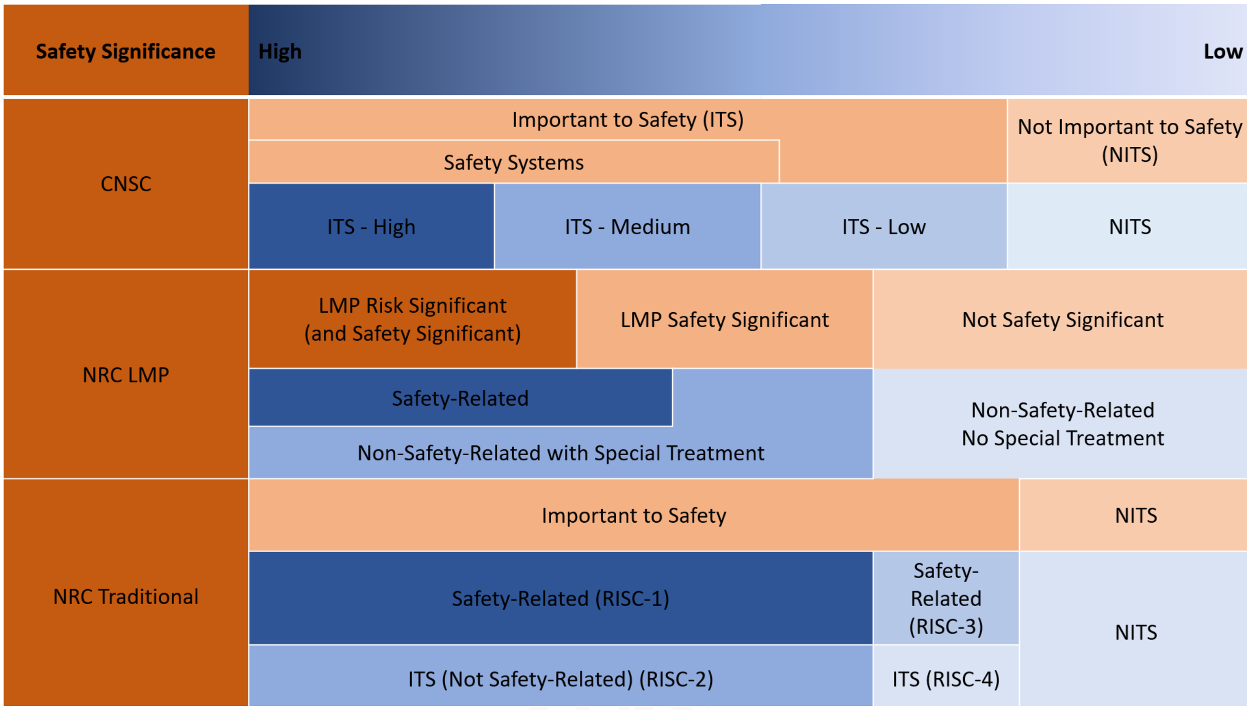


Figure 14: Safety Classification Overview (Illustrative Purposes Only)

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. The 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. The risk-informed safety classification process defined in 10 CFR 50.69 and the LMP are expected to minimize the scope of SSCs subject to the most comprehensive engineering design rules and specifications.

6 Programmatic Engineering Design Rules

Each regulatory body requires implementation of programmatic rules to ensure the quality and reliability of SSCs performing safety functions. Each regulatory body has different approaches to the grouping of these programmatic rules. For this report, programmatic rules will be compared in the following categories:

- Management for quality, which addresses those characteristics and functions of the organization that generally assure quality and reliability in areas important to the safety of the facility.
- Operational reliability, which addresses availability controls, maintenance practices, and periodic inspections and tests that help ensure the reliability of SSCs during operation.

6.1 Management for Quality

6.1.1 Overview

Quality assurance comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements. Quality assurance requirements are implemented under a management system that defines the organizational structure and establishes responsibilities for the conduct of certain functions related to the safe operation of the licensed facility, including implementation of procedures, control of changes to documents, assessments, and identification and resolution of problems.

A management system is the framework of processes, procedures and practices used to ensure that an organization can fulfill all tasks required to achieve its objectives safely and consistently. All objectives are managed under a management system, including quality. Management system requirements provide overall direction to the organization for developing and implementing sound management practices and controls for the organization.

Management system and quality assurance requirements and guidance are generally considered technology-neutral. Therefore, although some regulatory documents were developed for water-cooled reactor applications, the requirements generally apply to all reactor types, and guidance can be adopted for advanced reactor applications.

An effective and well-implemented management system provides the assurance that the design organization's design process has the necessary rigor and controls to ensure that the design outputs will:

- result in provisions meeting regulatory requirements for design and safety analysis
- minimize potential for latent safety issues
- provide the basis for design decisions.

6.1.2 CNSC Regulations, Guidance, and Standards

Quality Assurance Programs for Design

In Canada, the Class I Nuclear Facilities Regulations require that an applicant for any licence shall submit a proposed management system for the activity to be licensed (Class I Nuclear Regulations 3(d)). In addition, the Class I Nuclear Facilities Regulations require that an applicant to prepare a site or construct a reactor shall submit a proposed quality assurance program for the design of the nuclear facility (Class I Nuclear Regulations 4(d), 5(g)). In REGDOC-1.1.1 [48], “Site Evaluation and Site Preparation for New Reactor Facilities,” and related regulatory documents, the CNSC specifies that management system arrangements shall demonstrate adherence to Canadian Standard Association (CSA) N286 [49], “Management system requirements for nuclear facilities,” or equivalent standard established in the licensing basis, as applicable to the relative project phase. Management system and quality assurance requirements are embedded within CSA N286-12.

The CNSC regulatory framework does not segregate regulatory requirements based on classification of the safety functions involved. CNSC regulatory requirements are segregated between licensed and non-licensed activities, as defined by the NSCA. Management system and associated quality assurance requirements are applicable to all equipment and components used in support of licensed activities.

- In Canada, an applicant needs to demonstrate conformance to CNSC requirements for the licensed activities via its management system process document submissions. Related to design, these processes include, but are not limited to:

- Design Control
- Configuration Management
- Design Inputs & Assumptions
- Design Review/Verification
- Design Records
- Design Completion Assurance
- Design Authority
- Safety Analysis
- Use of External Design Agencies
- Graded Approach
- Document Control
- Software
- Research & Development

To meet the regulations, these processes should be described in the submitted design quality assurance program.

Relevant standards and REGDOCs include:

- CSA N286-12, Management system requirements for nuclear facilities, which integrates requirements from other management system standards for quality, health and safety, environment, economics, and security. This standard provides for graded application commensurate with risk and is based, in part, on the following management principles:
    - Safety is the paramount consideration guiding decisions and actions

    - The business is defined, planned, and controlled
    - The organization is defined and understood
    - Resources and managed
    - Communication is effective
    - Information is managed
    - Work is managed
    - Problems are identified and resolved
    - Changes are controlled
    - Assessments are performed
    - Experience is sought, shared, and used
    - The management system is continually improved
- REGDOC-2.1.1 [50], "Management System," references CSA N286-12 and highlights specific regulatory topics, including:
    - Leadership
    - Safety culture
    - Supply chain
        - Counterfeit, fraudulent, and suspect items
        - Management of contractors
    - Configuration Management
    - Software quality assurance

Control of Design Process

CNSC REGDOC-2.5.2 establishes a set of comprehensive design requirements and guidance that are risk-informed and align with accepted international codes and practices throughout all licensing phases. Section 3, Safety Management in Design, states that:

> *[T]he applicant or licensee shall be ultimately responsible for the design of the NPP and shall establish a management system for ensuring the continuing safety of the plant design throughout the lifetime of the NPP.*

An applicant must specify the processes to ensure that the design of structures, systems and components meets regulatory expectations, listed above. In the application, these processes must be documented as a part of the design quality assurance program submission.

REGDOC-2.5.2 specifies that design management be established for ensuring the continuing safety of the plant design throughout the lifetime of the NPP; design control measures (i.e. processes, procedures and practices) be established as part of the overall management system so as to achieve the design objectives; and design configuration be controlled. SSCs used in support of licensed activities include both basic component/safety- related and non-basic component/nonsafety-related and need to be designed to the requirements of CSA N286-12. REGDOC-2.5.2 states: "When the nations are not using the Canadian regulatory documents and standards and other control measures, those control measures should be mapped to the requisite CSA N286 clauses to demonstrate that they satisfy Canadian requirements."

The industry standard series CSA N299-19 [51], "Quality assurance program requirements for the supply of items and services for nuclear power plants," provides a quality standard that may be incorporated into procurement contracts. The first standard in standard series, CSA N299.1 [52], "Quality assurance program requirements for the supply of items and services for nuclear power plants, Category 1," is a standard that the CNSC has found to satisfy standard CSA N286-12 requirements for Category 1 items, which include those with a significant impact to safety, significant complexity, or first-of-a-kind design. Additional standards in that series are available for components and services with progressively lower safety significance in CSA Standards N299.2, N299.3, and N299.4 for Categories 2, 3, and 4, respectively. Appendix A, “Category Selection,” of each of the CSA N299 standards provides a systematic approach to selecting the appropriate category based on safety significance, complexity, and other characteristics of the affected service or component. This CSA N299-19 standard series is aimed at preventing nonconforming conditions by controlling design and production processes, conducting inspections and test verification activities, and developing corrective actions. Standards CSA N299.1, N299.2, and N299.3 include additional provisions in Section 8 describing a dedication process for instances where items or services could not be procured that were entirely controlled under the appropriate CSA N299 program category.

The CNSC staff verifies that the design process effectively controls the design, ensures that the design requirements are met, and the design configuration is controlled. Technical design rules may be specified in various regulatory documents, and the CNSC staff verifies that the applicant’s management system ensures the effective implementation of these design rules.

Quality Assurance for Commissioning Activities and Operation

For commissioning activities, an applicant for a license to operate is required to provide proposed commissioning program for the systems and equipment that will be used at the nuclear facility (Class I Nuclear Regulations 6(g)), in addition to the management system for operation. Both CSA N286-12 and REGDOC-2.3.1 [53], “Conduct of Licensed Activities: Construction and Commissioning Programs,” state construction and commissioning requirements.

Consistent with CSA N286-12, licensees must meet construction prerequisites, such as design completion, development of construction plans, and selection of materials and equipment, prior to construction. During construction activities, the licensee is required to ensure that drawings, specification and work documentation have controls in place. Commissioning activities must ensure that SSCs meet their design and safety analysis requirements prior to be considered in service. In addition, design, procurement, construction, and commissioning activities shall ensure that completion assurance elements are satisfied and turnover of the SSCs from one organization to another is controlled.

REGDOC-2.3.1 establishes the requirements and guidance for the construction and commissioning of facilities in Canada that use nuclear reactors. Part A, the construction of reactor facilities, and Part B, the commissioning of reactor facilities, identify the quality assurance requirements. In Part A, Section 3.2.3, Oversight of contractors, the licensee is required to have measures to ensure that contractors and subcontractors meet their respective contractual obligations and maintain records of oversight activities including the quality of construction and future operational safety. For example, monitoring the contractors’ and subcontractors’ effectiveness of controls for their management systems/quality assurance programs. In Part B, Section 8.1(1) requires the licensee to have primary responsibility for the safety and security for overseeing the organization, planning, execution, and assessment of the commissioning program including the construction organization’s assurance that SSCs are constructed as per design, and that quality assurance requirements have been satisfied.

6.1.3 NRC Regulations, Guidance, and Standards

Requirements and Guidance

Regulatory Requirements

1. 10 CFR Part 50 establishes quality assurance program requirements for the design, construction, and operation of nuclear power plants.
   - Appendix A, General Design Criterion 1 (GDC 1), “Quality standards and records,” to 10 CFR Part 50 requires, in part, that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed and that a quality assurance program be established and implemented to provide adequate assurance that these SSCs will satisfactorily perform their safety functions.
   - 10 CFR 50.34(a)(7) requires that an application for a CP include a description of the quality assurance program to be applied to the design, fabrication, construction, and testing of SSCs of the facility. The description must include a discussion of how the applicable requirements of Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50 will be satisfied.
   - 10 CFR 50.34(b)(6) requires, in part, that an application for an operating license include the following information related to facility operation:
     - The applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements.
     - Managerial and administrative controls to be used to assure safe operation. Appendix B to 10 CFR Part 50 sets forth the requirements for such controls for nuclear power plants and fuel reprocessing plants.

2. Appendix B to 10 CFR part 50 sets forth the requirements for establishing and implementing a quality assurance program for nuclear power plants and fuel reprocessing plants. This appendix establishes quality assurance requirements for the design, manufacture, construction, and operation of SSCs relied upon to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those SSCs; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

3. 10 CFR Part 52 references 10 CFR Part 50, Appendix B, for quality assurance programs associated with 10 CFR Part 52 licensees. The content of application requirements for SDAs, SDCs, MLs, and COLs require inclusion of a description of the quality assurance applied to the SSCs of the facility that satisfies applicable portions of Appendix B to 10 CFR Part 50, as well as a discussion of how the applicable requirements of Appendix B to 10 CFR Part 50 have been applied to design and will be satisfied during fabrication, construction, and testing, as appropriate.

4. An applicant for a CP or operating license under 10 CFR Part 50 or an applicant for a design approval, combined license, or manufacturing license under 10 CFR Part 52 may voluntarily choose to use 10 CFR 50.69 for risk-informed categorization and treatment of SSCs. For RISC-3 [safety-related with low safety significance] systems and structures, an applicant may choose to voluntarily comply with the requirements of 10 CFR 50.69 (e.g., corrective action and periodic inspection and testing for RISC-3 SSCs) in lieu of the requirements of

Appendix B to 10 CFR Part 50. For applicants that voluntarily adopt the risk-informed categorization process of 10 CFR 50.69, RISC-1 [safety-related with high safety significance] systems and structures must still comply with the requirements of Appendix B to 10 CFR Part 50, and alternative treatment requirements must be applied to RISC-2 [nonsafety-related with high safety significance] systems and structures to ensure performance of their functions consistent with the categorization process assumptions.

5. Per SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY 94-084)," and the associated SRM, applicants should specify the quality controls for nonsafety-related SSCs that are identified as being significant contributors to plant safety. Nonsafety-related SSC quality controls are detailed in SRP Acceptance Criteria "U. Nonsafety-related SSC Quality Controls," of SRP Section 17.5. Regulations that may involve the installation of safety-significant nonsafety-related SSCs include:

    - 10 CFR 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"
    - 10 CFR 50.48, "Fire Protection"

6. 10 CFR 50.55a, "Codes and standards," requires that systems and components of boiling and pressurized water-cooled reactors must meet the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III [54], "Rules for Construction of Nuclear Facility Components," as specified in paragraphs (b)-(g) of this section. The referenced standards invoke ASME NQA-1 [55], "Quality Assurance Requirements for Nuclear Facility Applications," for the design and fabrication of safety-related LWR pressure boundary components and supports.

<u>Regulatory Guidance</u>

1. RG 1.28 [56], "Quality Assurance Program Criteria (Design and Construction)," Revision 6, describes methods that the NRC staff considers acceptable for complying with the provisions of 10 CFR Part 50 and 10 CFR Part 52 for establishing and implementing a quality assurance program that is compliant with the requirements of Appendix B to 10 CFR Part 50 for the design and construction of nuclear power plants and fuel reprocessing plants. RG 1.28 endorses, with certain clarifications and regulatory positions, the Part I and Part II requirements included in various versions of, ASME NQA-1, including NQA-1-2017, NQA-1-2019, NQA-1-2022, and previously endorsed revisions NQA-1b-2011, Addenda to NQA-1-2008, NQA-1-2012, and NQA-1-2015.

2. RG 1.164 [57], "Dedication of Commercial-Grade Items for Use in Nuclear Power Plants," Revision 1, provides guidance for dedication of commercial-grade items and services used in nuclear power plants and additional clarification on the NRC's definition of counterfeit, fraudulent, and suspect items. This RG endorses, in part, the Electric Power Research Institute (EPRI) 3002002982, Revision 1 to EPRI NP-5652 and TR-102260, "Plant Engineering: Guideline for the Acceptance of Commercial-Grade Items in Nuclear Safety-Related Applications," with respect to acceptance of commercial-grade dedication of items and services to be used as basic components for nuclear power plants.

3. RG 1.231 [58], "Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Safety-Related Applications for Nuclear Power Plants," Revision 0, describes a

method that is acceptable to the NRC in meeting regulatory requirements for acceptance and dedication of commercial-grade design and analysis computer programs used in safety-related applications for nuclear power plants. RG 1.231 endorses EPRI Technical Report 1025243, “Plant Engineering: Guideline for the Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Nuclear Safety-Related Applications.”

4. SRP Section 17.5, “Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants,” provides guidance to the NRC staff in reviewing quality assurance program descriptions submitted by applicants for a design certification, combined license, early site permit, CP, and operating license.

Specific codes and standards

1. NQA-1: The requirements and guidance within NQA-1 is organized into four parts:
   - Part I contains requirements for a quality assurance program for a nuclear facility application. This part includes 18 sections that map to each criterion within Appendix B to 10 CFR Part 50.
   - Part II contains additional quality assurance requirements for the planning and conduct of specific work activities performed under a quality assurance program developed in accordance with requirements of Part I. For design and construction activities, this part includes quality assurance requirements for:
     - installation, inspection, and testing requirements of structural concrete, structural steel, soils, and foundations for nuclear facilities
     - computer software for nuclear facility applications
     - installation, inspection, and testing requirements of mechanical items for nuclear facilities
     - commercial-grade items and services
     - subsurface investigations for nuclear facilities
   - Part III contains guidance for implementing the requirements of Part I and II.
   - Part IV contains guidance for the application of NQA-1 and comparisons of NQA-1 with other quality requirements.

2. ASME BPV Code, Section III, “Rules for Construction of Nuclear Facility Components,” which is required by 10 CFR 50.55a to apply to the design and fabrication of safety-related pressure-retaining components and supports.

6.1.4 Similarities and Differences

The regulatory frameworks that establish quality assurance requirements for each regulatory body are similar. NRC regulations require that power reactor applicants include a description of the quality assurance program that will be applied to design, construction, and operation of the facility, including addressing the “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” provided in Appendix B to 10 CFR Part 50. Similarly, the Class I Nuclear Facilities Regulations under the *NSCA* specify the requirements for the proposed management system for the activity to be licensed, which encompasses quality assurance requirements. These documents provide regulations governing the quality assurance program by which each regulatory body will assess the applicant’s submission during construction, which includes design, and operation, which includes commissioning.

Although the frameworks are similar, differences do exist in the scope of the application of the quality assurance program requirement, in particular, for the quality standards specified by the licensee for the vendors. The NRC regulations require each license applicant to implement a quality assurance program that meets the criteria of Appendix B of 10 CFR Part 50, and the NRC has endorsed the ASME NQA-1 standard as providing methods acceptable to the NRC to comply with those requirements. These programs apply to all activities affecting the ability of the facility to safely shutdown and mitigate postulated DBEs using selected SSCs shown to meet acceptable performance standards in completing those functions. These activities are defined as "safety-related" activities. NRC policy and guidance establish a less rigorous level of quality assurance that should be applied to activities affecting SSCs that perform risk-significant important to safety functions identified through the RAP (see Section 3.3.8 of this report). Alternatively, a voluntary, risk-informed classification process established by regulation in 10 CFR 50.69 may be used to classify those SSCs where activities affecting their function must be covered by a quality assurance program meeting Appendix B to 10 CFR Part 50 and those SSCs subject to a less rigorous quality assurance program. Guidance for implementation of a less rigorous level of quality assurance is provided in SRP Section 17.5.

The CNSC's REGDOCs and CSA standards provide elements by which the regulatory body will assess the applicant's submission during the license to construct (design, construction, and some commissioning activities) and license to operate (including the remainder of commissioning activities) phases. The CNSC's licensing approach references CSA standard N286-12, which includes quality assurance as part of a broader management system framework. This standard specifies application of its requirements in a graded manner based on safety significance, which the applicant must define. Under this graded approach, all requirements continue to apply, but to varying degrees depending upon the safety significance and complexity of the work being performed (i.e. the attributes specified in the standard are expected to be met, but the measures used to meet those attributes can be less rigorous for less significant or less complex activities). The CNSC regulatory framework applies universally across all licensed activities. A licensed activity means an activity described in paragraph 26(e) of the *NSCA* (i.e. prepare a site for, construct, operate, modify, decommission, or abandon a nuclear facility) that a licence authorizes the licensee to carry on in relation to a Class I nuclear facility.

The CNSC requirements, guidance, and the referenced CSA N286-12 standard encompass all expected management system and quality assurance program elements, and these requirements and guidance are reasonably well aligned with Appendix B to 10 CFR Part 50 and the more detailed guidance in the NQA-1 standard endorsed by the NRC. Table B-1 in Appendix B to this report provides a comparison of NRC requirements and Appendix B quality assurance attributes with CNSC requirements and the scope of the CSA N286-12 standard and a brief assessment of the differences. Noted differences relate to:

- the independence of the quality assurance management and staff from the remainder of the facility organization specified in Criterion I of Appendix B to 10 CFR Part 50, which is not as clearly specified in CSA N286-12;
- the requirement for U.S. suppliers of basic components (safety-related SSCs), when the supplier is the dedicating authority, to report failures or nonconformances significantly affecting safety, which differs from CNSC regulations and expectations for only license holders to report such failures or nonconformances; and
- the applicability of CNSC management system regulations and CSA N286-12 to all licensed activities, as opposed to NRC regulations requiring application of the quality assurance criteria specified in 10 CFR Part 50, Appendix B, to activities affecting the

safety-related functions of SSCs that prevent or mitigate the consequences of postulated accidents.[12] The scope of NRC required quality assurance measures for SSCs is expanded by GDC 1, which states, in part, that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed and that a quality assurance program be established and implemented to provide adequate assurance that these SSCs will satisfactorily perform their safety functions.

The specified independence of the quality assurance management in Criterion I of 10 CFR Part 50, Appendix B, is intended to contribute to separation of safety concerns from cost and schedule pressures. However, the criterion notes that, irrespective of organizational structure, the individuals responsible for the effectiveness of the quality assurance program must have direct access to the levels of management necessary to assure its effectiveness. This necessary access to appropriate levels of management is mirrored in Clause 11, “Assessment,” of CSA N286-12, which mitigates the significance of the difference in specified characteristics of the organization structure.

In both the USA and Canada, components and services subject to quality controls are procured from vendors and suppliers, but the scope of regulation of vendor or supplier quality assurance activities vary. In the U.S., vendors and suppliers may be the dedicating authority certifying that the quality requirements of 10 CFR Part 50, Appendix B, have been satisfied, and the vendors and suppliers performing these activities are required to report to the NRC the manufacture of a safety-related item that contains a defect or if the manufacturing process of a safety-related component experienced a breakdown of quality assurance that could have produced a defective component. On the basis of this requirement, the NRC inspects a sample of vendors and suppliers that have a contractual obligation with an NRC licensee to implement a quality assurance program complying with Appendix B to 10 CFR Part 50 and dedicate the product or service as meeting those requirements. In Canada there are suppliers that also perform commercial dedicating services for U.S. and Canadian licensees. However, the manufacturing of items is not a licensed activity under the *NSCA* paragraph 26(e). License holder programs are required by the CNSC regulatory framework to ensure that all SSCs are fabricated and manufactured with the necessary level of quality to ensure that they are available to perform their intended design functions when called upon to do so. The standard series CSA N299-19 provides a quality standard that may be incorporated into procurement contracts, which the CNSC has found to meet its expectations. Although nuclear suppliers are not subject to direct regulation by the CNSC, under both the CNSC and NRC regulatory frameworks, the license holder is ultimately responsible to ensure the proper functioning of installed equipment important to safety.

With the exception of pressure boundary components of water-cooled reactors, neither the CNSC nor the NRC prescribe specific quality requirements or standards for fabrication and manufacturing. Instead, licensees and applicants are expected to establish suitable quality requirements, with justification, proportionate to the applicable performance requirements of the equipment necessary to satisfy its intended design function when called upon to do so. The exception is with pressure boundary items. The fabrication and manufacturing of water-cooled reactor pressure boundary items are governed by CSA standards specified by the CNSC and by

[12] NRC regulations include separate requirements regarding management systems and/or quality controls related to other licensed activities beyond the scope of this report, such as Appendix E to 10 CFR Part 50, “Emergency Planning and Preparedness for Production and Utilization Facilities”;

10 CFR 50.55a for NRC licensees (see Section 7.1, "Pressure-Retaining Components and Supports," of this report for additional details).

Although the scope of CSA N286-12 is well aligned with Appendix B attributes, CSA N286-12 applies to the licensed activity, regardless of the safety classification of an affected SSC, because the purpose and scope of CSA N286-12 is applicable to all licensed activities. Alternatively, the NRC requires the application of quality assurance measures listed in Appendix B to 10 CFR Part 50 only to activities affecting the functions of safety-related SSCs. Consistent with GDC 1 and Commission policy to manage the treatment of safety-significant SSCs that are not classified as safety-related, the NRC expects applicants to use a risk-informed RAP (described in Section 3.3.8 of this report) for the identification of nonsafety-related SSCs and activities associated with those SSCs that would be subject to additional design rules and specifications, such as quality assurance, in addition to normal commercial standards. Similarly, for the LMP framework, NEI 18-04 guidance specifies application of a quality assurance program satisfying Appendix B to 10 CFR Part 50 for activities affecting the safety-related functions of SSCs, while nonsafety-related with special treatment (NSRST) SSCs are specified to have a 'User provided Quality Assurance (QA) Program' consistent with the RAP described in SRP Section 17.4 and quality assurance guidance for nonsafety-related SSCs described in SRP Section 17.5. For applicants that receive NRC approval to voluntarily implement the risk-informed classification process described in 10 CFR 50.69, the applicant applies appropriate treatment consistent with the classification methodology for risk-significant nonsafety-related SSCs (RISC-2) and safety-related SSCs with low safety significance (RISC-3).

Some attributes of the quality assurance program and supporting management system are reasonably independent from the safety significance of the individual activities to which they are applied and reflect instead the overall safety significance of activities related to the nuclear facility. Examples include the organizational structure and the overall quality assurance program. These attributes are common between the CNSC and the NRC regulatory frameworks and are expected to influence the conduct of all licensed activities at the facility to some degree. However, under both regulatory frameworks the majority of quality assurance activities, and certainly those that are quality control activities, may be adapted in the extent and timing of their conduct to be commensurate with the safety significance of the activities to which they are applied.

While the CNSC framework provides for a spectrum of quality assurance levels, actual practice is likely to be a limited number of discrete levels of rigor for each attribute. The applicable CNSC guidance specifies approaches graded by variations in depth and/or timing, which could be reflected in a number of different variations in verification and auditing activities. For example:

1. audits could be varied by frequency, depth or scope of review, or independence (i.e., internal or external to the organization);
2. inspections could be varied in time (e.g., contemporaneous evaluation of the process or evaluation of records after completion) or depth (e.g., evaluation of entire process or evaluation of only the end product);
3. corrective actions could be designed to identify and correct root causes of nonconformances to preclude recurrence or simply correct the identified nonconformance; and
4. control of design changes may be graded by changing the degree of oversight.

Similarly, the NRC approach provides for two levels of quality assurance programs, consisting of a program applied to activities affecting the functions of safety-related SSCs that is mandated by regulation and a less rigorous program applied by policy to activities affecting important to safety functions. For each level, gradation permitted within each classification is based on safety significance. Therefore, the implementation of each attribute of the quality assurance program under both regulatory frameworks is consistent with gradations in the depth and/or timing of individual attributes based on safety significance of the activity.

As an example of the potential application of graded quality assurance, Table B-2 in Appendix B to this report provides a list of NRC and CNSC guidance regarding individual quality assurance attributes intended for activities affecting equipment with lower safety significance. The NRC guidance is drawn from SRP 17.5 for nonsafety-related SSCs that are significant contributors to plant safety (i.e., items contributing to DID that are not within the scope of the QA program described in Appendix B to 10 CFR Part 50). The CNSC guidance primarily references CSA Standard N299-19 for the design, procurement, and production of items under a supplier's internal quality assurance program with license holder oversight. Appendix A to CSA Standard N299-19 provides methods of grading the importance of items and services based on the complexity of manufacturing processes, the maturity of the design, or the risk to safety. Table B-2 lists guidance that could be applied under the NRC and the CNSC frameworks for license holder oversight and use of components manufactured by a supplier or vendor and assesses the degree of similarity in guidance. The comparison indicated that quality assurance attributes would be applied with similar levels of rigor (i.e., applied in a similar graded manner) under comparable guidance for SSCs with low to moderate safety significance.

6.1.5 Assessment of Impact

The overall application of management and quality assurance requirements to activities under the regulatory frameworks of Canada and the U.S. are similar. However, the NRC framework is more prescriptive and fully based on regulatory requirements; whereas the CNSC framework describes higher-level principles that are expected to be incorporated in the applicant's proposed management system and associated quality assurance measures. Therefore, an applicant seeking joint approval of a management system and associated quality assurance measures may use the framework of Appendix B to 10 CFR Part 50 to define the program and justify its conformance with CNSC requirements. At a minimum, the scope should encompass activities affecting the safety functions of SSCs in the highest safety classification. Management system criteria are essentially the same and could be evaluated under a joint review process. Alternative approaches that deviate from the quality assurance criteria established in Appendix B to 10 CFR Part 50 and the applicable NRC classification scheme (i.e., traditional deterministic classification, LMP classification, or risk-informed classification per 10 CFR 50.69) should evaluate the need for an exemption from NRC regulations related to application to activities affecting the safety-related functions of facility SSCs.

Each regulatory framework permits a graded approach to quality assurance in its application to specific SSCs, and a graded approach is expected to be used in determining the extent, depth, and timing of quality assurance measures applied to activities that could affect the functions of SSCs with safety classifications below the highest classification. However, applicants should consider the potentially broader scope of the overall management system specified in CNSC regulations, which specify that the management system encompass all licensed activities. With that consideration in mind, applicants should propose and justify a graded application of quality assurance measures that apply to activities affecting the important to safety functions of SSCs in lower safety classifications as well as management system measures addressing the complete scope of licensed activities.

An overall management structure is required under each framework that applies to any activity that could significantly affect safety. *Table 12* summarizes the application of quality assurance measures to individual SSCs based on the applicable safety classification.

Table 12: Quality Assurance Based on SSC Safety Function

SSC Safety Function	NRC	CNSC
SSCs relied on to ensure performance of fundamental safety functions during or following design basis events	These SSCs are classified as safety-related in the NRC Traditional and LMP (see Note) approaches. Appendix B to 10 CFR Part 50 criteria apply to all activities affecting the safety-related function of these SSCs.	These SSCs are generally part of “Safety System Groups” and are classified as Important to Safety. The application of CSA N286-12 is commensurate with safety significance.
SSCs that contribute to defense-in depth	These SSCs are classified as Important to Safety (Traditional) or NSRST (LMP). Quality assurance measures are applied in a graded manner, consistent with GDC-1 using guidance in SRP Section 17.4 and Section 17.5. Regulatory Guides provide guidance on extent of quality assurance for specific regulations related to fire protection, anticipated transients without scram, and loss of all alternating current power.	Management system and quality assurance implemented under CSA N286-12 applies to all licensed activities. A graded approach may be defined per Section 4 of CSA N286-12 to change the extent, intensity, or timing of management and quality control measures applied to Important to Safety SSCs commensurate with safety significance. Similar provisions in CSA N299-19 apply to license holder oversight of suppliers.
SSCs not important to safety	No special treatment of SSCs not important to safety. However, elements of the Appendix B to 10 CFR Part 50 program apply to all activities that could affect safety-related functions. Other managerial and quality assurance requirements may apply to other licensed activities (e.g., security and emergency planning).	No special quality controls are applied to these SSCs based on safety classification. However, elements of the Management System implemented under CSA N286-12 apply to all licensed activities on site, and, thus, may apply to the SSC on another basis.
Note: Under the LMP approach, DBEs are event sequences with mean frequencies between 10^{-4} and 10^{-2} per facility year, which excludes anticipated operational occurrences that are uncomplicated (i.e., no failures of relevant PRA modeled SSCs).		

6.2 Operational Reliability

6.2.1 Overview

Operational reliability programs play an important role in ensuring facility safety by maintaining the availability and capability of SSCs to perform their safety functions. Operational reliability programs include administrative controls on the availability of SSCs to perform their safety functions, maintenance controls that ensure the capability of SSCs to perform their safety functions, and pre-service and in-service testing and inspection that confirms the ability of SSCs to perform their safety functions. The reliability programs operate in an iterative fashion with the licensing PSA because the availability and reliability of SSCs is an input to the PSA and changes that affect SSC reliability, such as changes in the period of time equipment is unavailable, changes in maintenance practices that affect likelihood of operation at full capability, and changes in the interval between tests and inspections confirming full capability, should be reflected in the PRA.

This section covers the scope of SSCs subject to monitoring for availability, maintenance effectiveness, condition monitoring, and pre-service testing (PST)/in-service testing (IST) and pre-service inspection (PSI)/in-service inspection (ISI) of SSCs important to safety to provide reasonable assurance that these SSCs will be available and reliable when called upon to perform their safety functions. When generally referring to IST and ISI programs, it is understood that PST and PSI activities are included as part of those programs. The implementation of IST and ISI programs begins during the design and construction phases by incorporating adequate provisions to enable testing and examination in accordance with the applicable codes, standards, and regulatory guidance.

6.2.2 CNSC Regulations, Guidance, and Standards

The CNSC regulates the operational reliability of SSCs important to safety. Subsection 6(d) of the *Class I Nuclear Facilities Regulations* stipulates that an application for a licence to operate a Class I nuclear facility shall contain, in addition to other information, “the proposed measures, policies, methods and procedures for operating and maintaining the nuclear facility.” These measures are generally applied in a manner that considers the safety classification of the subject SSCs.

6.2.2.1 CNSC – Operating Limits

Section 4.3.3, “Safe Operating Envelope,” of REGDOC-1.1.3 [59], Licence Application Guide: Licence to Operate a Nuclear Power Plant,” includes provisions for defining the facility Safe Operating Envelope (SOE). The applicant should provide information related to the plant’s safe operating envelope, which is the set of limits and conditions within which a nuclear power plant must be operated to ensure compliance with the safety analysis upon which the reactor operation is licensed and that can be monitored and controlled by the operator. The application should include a description of how the corresponding requirements for surveillance, maintenance and repair are specified, to ensure that these parameters remain within acceptable limits and that systems and components are operable. Where appropriate, this information should be supported by means of a deterministic safety analysis and a PSA. Appendix C, Regulatory Documents and Standards,” indicates that applicants are expected to address as a requirement CSA Standard N290.15 [60], “Requirements for the Safe Operating Envelope of Nuclear Power Plants.”

Clause 4.2 of CSA N290.15: 2019, “Requirements for the Safe Operating Envelope of Nuclear Power Plants,” states that the SOE should include safety analysis limits, safe operating limits, conditions of operability, actions and action times, and surveillances. The guidance in Appendix A to the standard addresses operating Canada Deuterium Uranium (CANDU) nuclear power plants and specifies that the criteria for inclusion in the SOE includes instrumentation used to detect and indicate a significant degradation of the reactor coolant pressure boundary, any SSC providing a credited mitigation function to limit the consequences of an accident evaluated by deterministic safety analysis, initial conditions of SSCs or parameters limited to a specific range to demonstrate acceptable consequences, and other conditions found to be risk-significant that are not encompassed by the other criteria.

6.2.2.2 CNSC – Reliability and Maintenance Effectiveness

Requirements in the design phase help ensure that SSCs are capable of performing their safety function throughout the lifetime of the plant. For example, Section 3.1, “Design authority,” of REGDOC-2.5.2 specifies that the design authority must ensure the availability of the design information that is needed for safe plant operation and maintenance. Similarly, Section 3.2, “Design management,” of REGDOC-2.5.2 specifies that safety design information necessary for safe operation and maintenance of the plant and for subsequent modifications be preserved. It also specifies a design objective that the plant design facilitate maintenance and aging management. Section 5.14, “In-service Testing, Maintenance, Repair, Inspection, and Monitoring,” of REGDOC-2.5.2 states that the facility design shall be such that the SSCs important to safety can be calibrated, tested, maintained, and repaired (or replaced), inspected, and monitored over the lifetime of the plant. Guidance in Section 5.14 promotes the development of strategies and programs to address maintenance as a necessary aspect of the plant design phase.

During operation phase of the nuclear power plants, the CNSC addresses the fitness for service of SSCs important to safety through three related program areas: reliability monitoring, maintenance effectiveness, and aging management. These programs are applied to SSCs based in part on their safety classification.

The CNSC specifies reliability monitoring requirements in REGDOC-2.6.1 [61], “Reliability Programs for Nuclear Power Plants.” This document specifies establishment of reliability targets and monitoring against these targets for systems considered important to safety. The systems and components considered for reliability monitoring are selected from SSCs classified as important to safety using PSA importance measures to assess the relative contribution of systems to plant risk. The specified importance measures supporting this selection are the risk-increase ratio or risk achievement worth and the Fussell-Vesely importance. Among other considerations, an integrated expert panel that considers the results of deterministic safety analyses and DID principles may be used to complement the selection process using PSA importance measures. The program includes specification of reliability targets for the selected SSCs, which could include probability of failure per demand for systems not normally in operation and availability fractions for SSCs that consider periods of unavailability due to maintenance and component failures. The reliability monitoring program should consider design reliability, surveillance test results, condition monitoring, and system modeling in probabilistic analyses to recommend modifications to the maintenance program.

The maintenance program prescribed in REGDOC-2.6.2 [62], "Maintenance Programs for Nuclear Power Plants," applies to SSCs within the bounds of the nuclear power plant. The type and frequency of maintenance activities applied to each SSC are established commensurate with the SSC's importance to safety, design function, and required performance. The goal of the maintenance program is to optimize effective maintenance to support the safe operation of the nuclear power plant. The range of maintenance activities includes monitoring, surveillance, inspection, testing, assessment, calibration, service, overhaul, repair, and replacement.

The aging management program prescribed by REGDOC-2.6.3 [63], "Aging Management," applies to SSCs that can directly or indirectly affect safe operation. This includes SSCs important to safety and nonsafety-related SSCs whose failure could prevent accomplishment of safety functions. The aging management program includes condition monitoring activities.

6.2.2.3 CNSC – Inspection and Testing

In Canada, the CNSC uses regulatory documents and licence conditions as the means to establish inspection requirements and guidance. Section 5.14 of REGDOC-2.5.2 provides the design requirements related to in-service inspection, testing, maintenance, and monitoring. 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.

REGDOC-2.5.2 notes that design of the SSCs important to safety shall ensure that the SSCs can be periodically inspected (inspectability) and tested (testability) to maintain the SSC and the reactor facility to be within the boundaries of the design. The design shall identify the standards for performing the inspections (pre-service and in-service) commensurate with the importance of the respective safety functions of the SSCs, with no significant reduction in system availability or undue exposure of the site personnel to radiation.

REGDOC-2.5.2 requires that the SSCs important to safety shall be designed to enable periodic testing and inspection of the SSC to monitor the condition over the lifetime of the plant. It notes that the design of the SSCs should allow effective implementation of pre-service and in-service inspections and testing during commissioning and operation phases of the plant's lifecycle. In particular, the reactor core should be designed to permit the implementation of a material surveillance program to monitor the effects of service conditions on material properties throughout the operating life of the reactor. Hence, when finalizing the design rules for an SSC, an applicant needs to consider the SSC's importance to safety and the design rules, should include operational considerations, such as testability, inspectability, maintainability and aging management of the respective SSC. Due to the complexity of the design, if an SSC cannot be designed to ensure periodic inspection and/or testing, the design need to ensure that other alternative methods are available to monitor the component, such as surveillance of reference items, or validated calculation methods following international best practices. These alternative approaches shall be part of the design documentation of the SSC.

REGDOC-2.5.2 notes that the establishment of clear strategies and programs to address pre-service and in-service inspection and testing of the SSCs as a necessary and important component of the design phase of a plant. During the design phase itself, an applicant should develop clear and structured strategies and programs for pre-service and in-service inspection and testing of SSCs, that need to be implemented during the commissioning and operational phase of the plant to ensure that plant SSCs remain capable and available to perform their safety functions. The strategies and programs developed by the applicant should identify the relevant inspection and testing activities that needs to be completed during the construction and commissioning of the SSC and the plant, to enable effective trending of the performance and aging and also to provide a meaningful baseline data of the SSC and the plant, at the outset of its operating life. The strategies should also include effective programs for evaluating and trending SSC performance and aging, coupled with an optimized preventive maintenance program. The strategies and programs should consider:

- SSCs safety significance, intended design life, design loading conditions, and operational requirements.
- Requirements of regulations and applicable codes and standards.
- Interdependence of SSCs important to safety and possible effects of failures of SSCs of lower safety significance on SSCs of higher safety significance.
- Design, layout, and the accessibility of SSCs during construction, commissioning, and operational phase of the plant's lifecycle.
- Operational experience from reactor facilities of similar or identical design and layout.
- Technologies and methodologies inspection, monitoring and testing during construction and commissioning phase and operational phase of the plant's lifecycle.

In addition to the requirements of REGDOC-2.5.2, REGDOC-2.6.3, "Aging Management," requires development of effective programs and plans for inspection and testing of the SSCs to document the condition of the SSC and to have effective trending of the performance of the component. REGDOC-2.6.3 also stresses the importance of timely detection and characterization of degradation through inspection, testing and monitoring of an SSC, and the associated evaluation of the inspection and testing data to determine the type and timing of corrective actions required.

6.2.2.4 Pre-service and In-service Inspection and Testing Standards

In Canada, CSA N285.0 [64], "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants," provides the requirements for the applicable design codes that apply for the pressure-retaining components. This includes establishment of rules for classifying SSCs based on importance to safety and principles and criteria consistent with the Canadian safety philosophy. The design requirements prescribed by CSA N285.0 are addressed in Section 7.1.2 of this report.

The Canadian approach as documented in REGDOC-2.5.2 allows the use of alternative approaches when the applicant demonstrates an equivalent or superior level of safety of the SSC for its intended function. It is also noted that CSA N285.0 does not contain any requirements that would prevent it being applied to water-cooled reactor technologies other than CANDU that use Section III, Division 1 of the ASME BPVC for the design of pressure-retaining components and associated supports.

The CSA N285 series, among other requirements, establishes the pre-service and in-service inspection and testing requirements for the metal pressure-retaining SSCs and their supports over the plant's entire lifecycle, starting from the design phase. In general, for the pre-service requirements for non-CANDU specific components, CSA N285.0 relies on the relevant requirements of ASME BPVC, Section III Division 1 for all component classes. However, for in-service requirements, CSA N285.0 notes that the licensee shall comply with the requirements of CSA N285 series of Standards. This includes any inaugural or baseline inspection of a pressure boundary SSC that is repaired, replaced, or changed during service. The CSA N285 Standard series assumes that the safety-significant SSCs are fabricated and installed in accordance with the quality requirements of CSA N285.0, do not exhibit or have any significant defect population, and the SSCs have successfully passed all of the pre-service inspection and testing requirements. In this scenario, periodic inspection can logically be based on a sampling approach provided that the components or samples for inspections are chosen to include extreme operating conditions.

The CSA N287 Standard series establishes rules for the concrete containment structures of nuclear power plants, including pre-service and in-service inspection and testing requirements. The standard CSA N287.7 [65], “In-service Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants,” establishes the scope and frequency of inspections and the periodicity of integrated leak rate tests for concrete containments.

6.2.3 NRC Regulations, Guidance, and Standards

6.2.3.1 Technical Specifications (10 CFR 50.36)

Section 182 of the Atomic Energy Act of 1954, as amended, requires, in part, that applicants for a license to operate a utilization facility include technical specifications (TSs) regarding the specific characteristics of a facility and other information that the Commission deems necessary to find that the facility will operate in accordance with the common defense and security and will provide adequate protection to the health and safety of the public. In 10 CFR 50.36, the Commission specified the scope of TSs for nuclear power reactors. The regulation requires, in part, that nuclear power plants have TSs in the following categories: safety limits and limiting safety system settings; limiting conditions for operation (LCOs); surveillance requirements; design features; and administrative controls.

An LCO is defined as the lowest functional capability or performance level of equipment required for safe operation of the facility. When an LCO is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met. Per 10 CFR 50.36, a TS LCO must be established for each item meeting one or more of the following criteria:

- *Criterion 1*. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

- *Criterion 2*. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- *Criterion 3*. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- *Criterion 4*. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Surveillance requirements are requirements related to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

The TSs LCOs and surveillance requirements apply principally to systems and components classified as safety-related under the NRC traditional approach or the LMP, but also include certain other important to safety SSCs with high safety significance. This outcome is a result of the linkage of Criterion 2 and 3 with those conditions and functions that are relied upon to mitigate a DBA or transient. Criterion 1 applies to important to safety instrumentation in water-cooled reactors, and it may be applied to advanced reactors where the defence-in-depth provided by leakage detection is particularly important to safety. Criterion 4 applies to risk-significant SSCs, and results in the assignment of LCOs to certain nonsafety-related SSCs that are particularly important to safety. The TS LCOs and surveillance requirements provide a means of ensuring the availability of safety-related and risk-significant important to safety SSCs, consistent with assumptions in probabilistic assessments.

6.2.3.2 Maintenance Rule (10 CFR 50.65)

Overview

The NRC Maintenance Rule regulation, 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," was issued in 1991. The regulation requires that licensees monitor the performance or condition of specified SSCs against licensee-established goals in a manner sufficient to provide reasonable assurance that these SSCs are capable of fulfilling their intended functions. The requirements in 10 CFR 50.65(b) relate to the criteria used to determine which SSCs are to be subject to reliability monitoring. While the Maintenance Rule is a performance-based regulation and not generally prescriptive, 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.

The maintenance program is supported by design and management requirements. The GDC of Appendix A to 10 CFR Part 50 and the ARDC listed in RG 1.232 include provisions to design SSCs important to the performance of fundamental safety functions support periodic inspection and testing, which would also support maintenance activities related to these SSCs. The design control and document control provisions of Appendix B to 10 CFR Part 50 support maintenance and retention of design, operational, and maintenance records necessary to support effective maintenance activities that could affect the safety-related functions of related SSCs.

Requirements and Guidance

The regulation at 10 CFR 50.65 specifies criteria for the selection of SSCs 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 DBEs 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.

Monitoring under the Maintenance Rule is also required for nonsafety-related SSCs:

- that are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs);
- 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 rule requires that the identified SSCs are monitored for performance and condition against licensee-established goals. The performance goals may include on-demand failures and availability monitoring.

The risk-informed SSC categorization process of 10 CFR 50.69 allows 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). In cases where an alternative treatment has been established for RISC-3 SSCs, licensees are not required to scope RISC-3 and RISC-4 SSCs into the Maintenance Rule per 10 CFR 50.69(b) for monitoring. 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 the provisions of 10 CFR 50.69.

NRC RG 1.160 [66], "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 4, endorses, with exemptions and clarifications, the industry guidance contained in Nuclear Management and Resources Council (NUMARC) 93-01 [67], "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 4F. 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.

Section 8.0, "Methodology to Select Plant Structures, Systems, and Components," of NUMARC 93-01 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 standard, IEEE 933-2013 [68], "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 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 [69], "Monitoring the Effectiveness of Nuclear Power Plant Maintenance," Revision 0, was issued in July 2019. Industry has not requested NRC endorsement of NEI 18-10. However, industry describes the guidance it contains 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.

6.2.3.3 Pre-Service and In-Service Testing and Inspection

In GDC 1[13], the NRC establishes the need to design, fabricate, erect, and test LWR SSCs to quality standards commensurate with the importance of the safety functions to be performed. The applicability and regulatory guidance related to the use of GDC in developing PDC is discussed in Section 3.2.2 of this report. Therefore, the PDC developed by applicants of SMRs or advanced reactors must include design features and programmatic activities that enable examination and testing methods of important to safety SSCs as required by the applicable codes and standards.

The NRC requires that applicants for an OL or COL include plans for preoperational testing and initial operations, as well as plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components. Commission Paper SECY-05-0197 [70] "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," summarizes the NRC position regarding the full description of operational programs to be provided by COL applicants.

Regulatory Framework for PST/IST and PSI/ISI of LWR SSCs

In 10 CFR 50.55a, the NRC incorporates by reference codes and standards that are to be implemented in design, construction, and operations of LWR power plants. Section 3.2.2**Error! Reference source not found.** of this report describes the regulatory framework that specifies the codes and standards for safety-significant SSCs. In particular, Section 7.1.3 of this report**Error! Reference source not found.** provides a summary of the quality group designations for pressure-retaining components, according to their importance to safety.

The scope of PST/IST and PSI/ISI for LWRs is driven by the SSC code classification under designated quality groups (QG-A, -B, -C). The ASME Code classification is then used to determine the applicable IST and ISI requirements in accordance with the ASME *Operation and*

[13] The requirements of GDC 1 apply to all SSC important to safety. There are other GDC not mentioned in this section that also require the design capability to test specific SSCs important to safety. The regulatory guidance for application of such design criteria for SMRs or advanced reactors is discussed in NRC RG 1.232 and Section 3.2.2 of this report.

Maintenance of Nuclear Power Plants, Division 1, OM Code: Section IST (ASME OM Code) and ASME BPVC, Section XI, respectively. In addition, 10 CFR 50.55a(b)(3)(iii)(D) requires that licensees assess the operational readiness of pumps, valves, and dynamic restraints within the scope of the Regulatory Treatment of Non-Safety Systems for applicable reactor designs.

Regulatory Background for IST and PSI/ISI of non-LWRs

For non-LWR advanced reactors, RG 1.87 describes acceptable methods for quality group classification of components of high temperature reactors under traditional (i.e., classification based on function), risk-informed, and LMP safety classification processes. *Table 15* provides a summary of the applicable code classification for non-LWR high temperature reactors that are constructed to ASME BPV Code, Section III, Division 5. However, it should be noted that the Code classification for non-LWRs does not play a role in determining the requirements for PST/IST and PSI/ISI, as described in the following paragraphs.

The NRC issued DANU-ISG-2022-07 [71], "Risk-Informed In-service Inspection/In-service Testing Programs for Non-LWRs," for two reasons: (1) to provide guidance on the contents of applications to an applicant submitting a CP, OL or COL, and (2) to provide guidance to the NRC staff on how to review such applications.

Framework for PSI/ISI of non-LWR

The regulations in 10 CFR 50.55a do not contain requirements for non-LWR ISI programs. However, the GDC in 10 CFR Part 50, Appendix A, are generally applicable to and provide guidance in establishing the PDC for types of reactors other than water-cooled reactors governed by the GDC. The ability of the design to allow for in-service inspection of SSCs important to safety is addressed in various GDC related to the protection of reactor coolant pressure boundary, emergency core and containment heat removal, containment atmosphere cleanup systems, and essential cooling systems. Similarly, the importance of PSI/ISI is reflected in advanced reactors design criteria defined in RG 1.232.

As identified in DANU-ISG-2022-07, the purpose of a risk-informed ISI program is to periodically monitor and track degradation (defects, corrosion, erosion) in welds and base metal of components and component supports within the program's scope to determine their suitability for continued operation, consistent with the plant-specific PRA. The scope of a risk-informed ISI program includes all piping, pressure-retaining components, and component supports that perform safety-significant functions, as well as piping or other components whose failure could prevent SSCs from performing their safety functions.

Applicants pursuing a risk-informed PSI/ISI program should describe how risk information is used to guide (1) the selection of the inspection locations during each inspection interval, (2) the inspection frequency for each location, (3) the inspection technique to be used, and (4) how the selection process varies from one inspection interval to the next to cover all components of interest. Although intended for LWRs, the ASME BPV Code, Section XI, Division 1, might provide useful information on inspection techniques and frequencies within the conditions for which the ASME Code specifies their use.

PSI/ISI using a Reliability and Integrity Management (RIM) Program

The NRC staff issued RG 1.246 [72], "Acceptability of ASME Code, Section XI, Division 2, 'Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power

Plants,' for Non-Light Water Reactors," to provide applicants and licensees of non-LWRs an acceptable method for developing and implementing a PSI and ISI program. RG 1.246 endorses with conditions ASME BPV Code, Section XI, Division 2, for use by non-LWR applicants and licensees.

ASME BPV Code, Section XI, Division 2, provides a process for developing a RIM program similar to a traditional PSI and ISI program under ASME BPV Code, Section XI, Division 1, for all types of nuclear power plants. The RIM program contains provisions beyond a traditional program, such as significant use of PRA to develop reliability targets for SSCs within the scope of the program.

ASME BPV Code, Section XI, Division 2, also provides a process for the identification of the scope, degradation mechanisms, and reliability targets for the in-scope SSCs; identification and evaluation of RIM strategies and uncertainties; program implementation; performance monitoring; and program updates to be applied for passive components to give assurance that the reliability will meet preestablished targets (developed from the PRA information for the facility).

The use of ASME BPV Code, Section XI, Division 2 and RG 1.246 is not mandatory. Rather, each applicant is ultimately responsible for proposing a risk-informed ISI program appropriate for its own design and technology. This may or may not include a RIM program.

Applicants and licensees for non-LWRs that choose to follow the guidance in RG 1.246 are expected to implement the processes described in ASME BPV Code, Section XI, Division 2, to identify and select SSCs within the scope of the RIM program, consistent with the regulatory positions in RG 1.246. ASME BPV Code, Section XI, Division 2, subparagraph RIM 2.2-2, defines the scope as those SSCs whose failure could adversely affect plant safety and reliability. Article RIM-2 further defines acceptable risk-informed methods for establishing reliability targets for in-scope SSCs and then develop RIM strategies or combinations of strategies that are necessary and sufficient to achieve and maintain SSC reliability consistent with SSC reliability targets. The RIM strategies consider various factors, such as selection of materials, fabrication procedures, pre-service and in-service examinations and testing, as well as maintenance, repair, and replacement practices. Those SSCs whose selected RIM strategy includes PSI/ISI activities will then be subject to such programs in accordance with the rules defined in the applicable Division 2 articles and appendices, in accordance with the regulatory positions specified in RG 1.246.

Framework for PST/IST of non-LWRs

Traditional IST programs executed in accordance with the ASME OM Code are only required for SSCs classified in accordance with the OM Code under either traditional or TI-RIPB approaches, as discussed in Section 3.2 of this report. For example, the scope of the ASME OM Code includes safety-related components as defined in 10 CFR 50.2. Applicants and licensees of non-LWRs must follow the provisions specified in GDC 1 as described in RG 1.232 or justify an alternate method during the NRC licensing process for testing and examining SSCs within the scope of the PST/IST program for the specific non-LWR design. ASME is currently preparing a new code (designated as the OM-2 Code) to provide PST and IST provisions for components in new and advanced reactors, including non-LWR designs. Rather than specifying pumps, valves, and dynamic restraints, the scope of the draft OM-2 Code is stated to be components whose operational readiness will be assessed by demonstrating that they are capable of performing their specified functions, including generate, allow, throttle, or isolate fluid flow; provide pressure relief capability; and establish dynamic restraint to ensure the structural integrity of piping systems and their components. The NRC staff is considering the preparation of a new regulatory guide to accept, with appropriate regulatory positions, the new OM-2 Code when issued by ASME. The NRC staff plans to use RG 1.246, which provides guidance for PSI/ISI programs to be developed for non-LWRs, as a template for preparation of the new regulatory guide that will address the PST/IST provisions to be specified in the ASME OM-2 Code for new and advanced reactors, including non-LWR designs.

6.2.4 Similarities and Differences

Both the CNSC and the NRC have established programs as part of the facility licensing review to ensure engineering design rules and other specifications applied to an SSC provide predicted availability and reliability commensurate with its importance to safety. Programs for availability apply to important to safety SSCs (i.e., for the NRC, the scope includes safety-related as well as important to safety classifications, which includes RTNSS considerations under traditional approaches and NSRST under the LMP approach).

The CNSC reliability goals are more specific than the associated NRC Reliability Assurance Program guidance in that the goals include minimum system failure on demand probabilities 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 PRA results, and (2), for the LMP approach, through the PRA of the design 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 program attributes relate to testing, inspection, availability management, and maintenance. As a design focused review, this assessment focuses on the 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 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 under the LMP approach (i.e., SSCs classified as NSRST), they are expected to align with the results of the risk-informed classification process pursuant to 10 CFR 50.69. This review includes assurance that the nonsafety-related SSCs included in the RAP are also within the scope of 10 CFR 50.65.

6.2.5 Assessment of Impact

The CNSC Safe Operating Envelope and Fitness for Service programs and the NRC operational reliability programs under the TSs and Maintenance Rule have significant overlap such that many programs are very similar, and some are interchangeable. Although the CNSC approach offers additional flexibility with respect to pre- and in- service testing of pressure-retaining components and supports, 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 approach specifies quantitative risk metrics for selection of SSCs subject to reliability monitoring that may not align with the more deterministic scope specified in the NRC regulation at 10 CFR 50.65. However, applicants could minimize the difference in scope by using risk-informed approaches identified in 10 CFR 50.69 to risk-inform the scope of SSCs subject to reliability monitoring under 10 CFR 50.65. Therefore, this impact could be mitigated.

Table 13 lists the respective elements of operational programs, including the scope of applicability for each regulatory body, and assesses the similarity of the programs and alignment of SSCs within the program scope.

Table 13: Comparison of Operational Reliability Measures

Reliability Area	NRC Measures	CNSC Measures	Assessment
Availability - Operating Limits	Technical Specifications (10 CFR 50.36); Applies primarily to safety-related and high safety significance SSCs.	Safe Operating Envelope incorporated in license; Applies primarily to safety group SSCs.	Essentially the same program with the same criteria to establish applicability to SSCs.
Availability – Performance Monitoring	Maintenance Rule (10 CFR 50.65); Applies to deterministic set of SSCs; risk-informed through 10 CFR 50.69 or LMP process.	Reliability Programs (REGDOC-2.6.1); applies to set of SSCs that meet PSA safety significance metrics; may be risk-informed.	Similar programs; risk-informed approaches likely to increase the alignment of SSCs considered to be within scope of the programs.
Availability – Maintenance Effectiveness	Maintenance Rule (10 CFR 50.65); Applies to deterministic set of SSCs; risk-informed through 10 CFR 50.69 or LMP process.	Maintenance Programs (REGDOC-2.6.2); Activities prioritized based on SSC importance and safety classification.	Similar programs intended to optimize maintenance of safety-significant SSCs. Flexibility to align SSCs within scope.
Availability – Condition Monitoring	Maintenance Rule (10 CFR 50.65); Applies to deterministic set of SSCs; risk-informed through 10 CFR 50.69 or LMP process.	Aging Management (REGDOC-2.6.3); Licensee selects SSCs based on importance and degradation potential.	Similar programs focused on maintaining the functional capability of SSCs. NRC focus on passive SSCs, CNSC provides flexibility to align in-scope SSCs.

Reliability Area	NRC Measures	CNSC Measures	Assessment
Pre- and In-Service Testing and Inspection of Pressure-Retaining Components and Supports	Per 10 CFR 50.55a (f) and (g), LWR safety-related pressure-retaining components are required to be tested and inspected to meet the ASME Code. RG 1.246 and DANU-ISG-2022-07 address Reliability and Integrity Management for advanced reactors. In addition,10 CFR 50.55a(b)(3)(iii)(D) requires that licensees assess the operational readiness of pumps, valves, and dynamic restraints within the scope of RTNSS.	CSA N285 Standard series addresses testing and inspection of CANDU reactor components to ASME code. CSA N287.7 addresses testing and inspection of concrete containments. The CNSC provides guidance to consider alternative approaches for advanced reactors.	Similar testing and inspection programs that apply to a similar scope of water-cooled reactor pressure-retaining components and supports. The CNSC guidance provides flexibility to align with NRC endorsed standards applicable to advanced reactor designs.
Other Pre- and In-Service Testing and Inspection	License application requirements address testing and inspection. Technical Specification surveillance requirements for safety-related SSCs and SSCs with high safety significance.	License application requirements address plans for testing and inspection. The Safe Operating Envelope includes surveillance requirements for SSCs with high safety significance.	Highly similar programs for managing tests and inspections of SSCs with high safety significance.

7 Design of Specific Structures, Systems, and Components

7.1 Pressure-Retaining Components and Supports

7.1.1 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 U.S. and Canada normally reference consensus standards from the 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 BPVC provides 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 governs 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. *Table 14* provides the organization of the ASME BPVC, Section III, Division 1:

Table 14: Organization of ASME BPVC, Section III, Division 1

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 following table:

Table 15:Organization of ASME BPVC, Section III, Division 5

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 Metallic Supports	A	Low Temperature Service
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 [73], “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 [74], “Power Piping,” provides engineering design rules for construction of pressure-retaining piping in power plants.

7.1.2 CNSC Regulations, Guidance, and Standards

In Canada, the CNSC uses regulatory documents and licence conditions as the means to establish both requirements and guidance. CNSC REGDOC-1.1.2 [75], “Licence Application Guide: Licence to Construct a Nuclear Reactor Facility,” 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. The application should also describe the pressure boundary codes and standards 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 [76], “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:

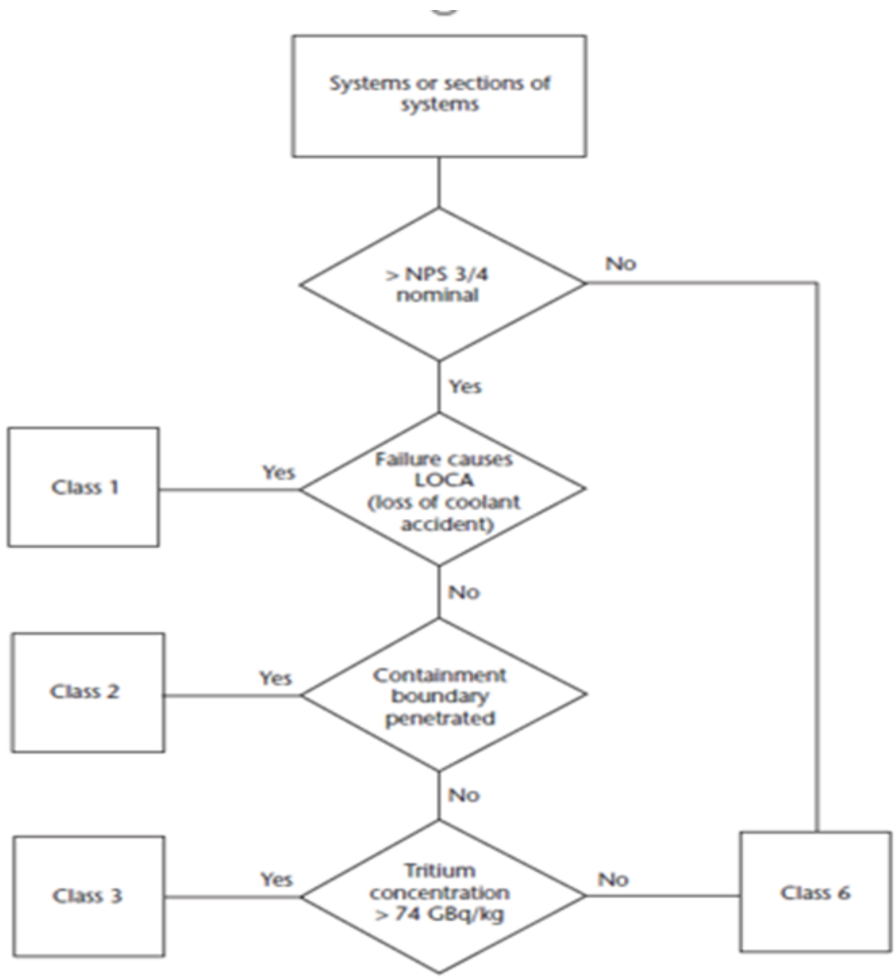


Figure 15: N285.0 Simplified guide to the classification of process systems

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. *Table 16* shows the CSA Std. N285 classes with the assigned design code.

Table 16: Canadian Standards Association N285 Classification and Standards

CSA 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 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

The 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 states that the design authority should ensure that 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 indirect 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.

7.1.3 NRC Regulations, Guidance, and Standards

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[14]. 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 BPVC. 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[15] 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 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

[14] 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.
[15] 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."

are part of the reactor coolant pressure boundary 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[16].

Quality Groups

The primary NRC guidance in RG 1.26 [77], “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:

- Quality Group A consists of the reactor coolant pressure boundary components of LWRs, 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 quality assurance, 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 17* provides assignment of standards for system components based on the Quality Group designation.

[16] 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.

Table 17: NRC Classification and Standards for LWR Pressure-Retaining Components

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

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 American National Standards Institute (ANSI)/American Nuclear Society (ANS)-58.14-2011 [78] 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 SSG-30, applicants may propose the 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 [79], "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 PDC 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. *Table 18* 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 18: NRC Classification and Standards for High Temperature Pressure-Retaining Components

Classification Method	**Component Classification**		
Traditional	Quality Group A	Quality Group B	Quality Group C
Risk-Informed (10 CFR 50.69)[Note A]	RISC-1	RISC-1	RISC-2, RISC-3
Risk-Informed (RG 1.233)	SR	SR	NSRST
Components	SR Quality Design Standards		Important to Safety Design Standards
Pressure Vessels	ASME Code, Section III, Division 5, Class A	ASME Code, Section III, Division 5, Class B	ASME Code, Section VIII, Division 1 or Division 2[Note B]
Piping			ASME B31.1/B31.3 [Note B]
Pumps			
Valves			ASME B31.1/B31.3 [Note B]
Atmospheric Storage Tanks			
Storage Tanks (0-15 pounds per square inch gauge)			ASME Code, Section VIII, Division 1 or Division 2[Note B]
Metallic Core Support Structures	ASME Code, Section III, Division 5, Subsection HG	N/A	
Nonmetallic Core Support Structures	ASME Code, Section III, Division 5, Subsection HH	N/A	

- Note A: Alternative treatment under 10 CFR 50.69 for SSCs categorized as RISC-1, RISC-2, RISC-3, or RISC-4 requires NRC review and approval in accordance with 10 CFR 50.69.
- Note B: Application of these standards should be justified on a case-by-case basis.

7.1.4 Similarities and Differences

Table 19: 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

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. *Table 19***Error! Reference source not found.** shows a correlation of CSA N285.0, NRC RG 1.26, and ANSI/ANS 58.14 code classifications or quality groups with the applicable ASME BPVC Subsection. 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. CSA 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 may use commercial-grade standards, such as ASME BPVC Section VIII, with appropriate justification.

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 (RG 1.87) 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.

7.1.5 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 CSA 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 risk-informed 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 the 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 Power

7.2.1 Overview

This section covers code classification and assignment of engineering design rules for electrical power systems and their associated components. Such systems and components include classes of power, standby and emergency generators, batteries, rectifiers, cables, etc. A highly reliable electric power supply may be necessary to control and monitor the nuclear facility for any deviation from normal operation under all plant states, consistent with the safety analysis. In addition, the electrical power systems may support safety systems that accomplish fundamental safety functions or contribute to defence-in-depth.

7.2.2 CNSC Regulations, Guidance, and Standards

The preferred power supplies (fed by either the offsite power systems from the grid or the main generator systems) play an important role in the normal operation and performance of important safety functions. However, these power supplies are classified as not important to safety and their loads are those that may tolerate long term power interruption without affecting safety.

The safety classification of the onsite power systems is dependent on the loads that are being supplied by those systems. If those loads can tolerate short term power interruptions or no interruptions at all based on their associated safety functions, then the classification of those electrical components varies with the importance of electrical power. The availability and reliability of the electrical power systems is commensurate with the safety significance of the connected loads.

All cables used in the nuclear plant are either classified as important to safety or not important to safety cables depending on the safety significance of the end device.

Regulatory Framework

In Canada, the Class I Nuclear Facilities Regulations require that applications contain a description of the systems and equipment to be installed, including their design and design operating conditions, but there are no specific regulations related to electrical power systems. Consequently, the CNSC uses regulatory documents and licence conditions as the means to establish both requirements and guidance.

CNSC REGDOC-1.1.2 sets out requirements and guidance on submitting an application 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 electrical power systems in order to meet the expectations of REGDOC-2.5.2.

Section 5.1 of REGDOC-2.5.2 describes that all SSCs shall be identified as either important or not important to safety. This section provides general design requirements and guidance on safety classification of SSCs. The design bases, design criteria, regulatory documents, standards, and other documents that will be used to design the electrical power systems should be specified.

Section 6.9, Electrical power systems, of CNSC REGDOC-2.5.2 specifies design criteria for normal (preferred) alternating current (AC) power systems, standby/emergency AC power systems and sources, direct current (DC) and uninterruptible power systems; and alternate AC power supplies. REGDOC-2.5.2 guidance states that a systematic approach should be followed to identify the electrical power systems needed in order to ensure that SSCs necessary to fulfill the safety functions are powered from electrical power supplies with appropriate safety classification and reliability. This section of REGDOC-2.5.2 references Section 5.10, Safety support system, for additional guidance related to the reliability of portions of the electrical distribution system that directly support safety systems.

For each of the electrical power systems, the design bases should include:

- consideration of all modes of operation, including DBAs, DECs, and all credible events that could impact the electrical power systems;
- reliability and availability targets for systems and key equipment;

- identification of all loads (i.e., the systems and equipment that require electric power to perform their safety functions) including electrical characteristics, maximum demand conditions, and safety classification; and
- protective schemes and specification of acceptable operating ranges.

Normal power supplies from offsite (preferred) sources should have the capacity to supply all electrical loads during normal operational states, DBAs, and DECs. The number and independence of offsite power lines should be consistent with the safety functions performed by the AC power system.

Standby/emergency power systems have capacity and reliability to support a defined mission time in the presence of a single failure to maintain the plant in a safe shutdown state, ensure nuclear safety in DBAs and DECs, and support severe accident management actions. Offsite power is the preferred power supply to the standby electrical system. Each standby electrical source should be a complete electrical generating unit. The standby/emergency power system preferably initiates automatically and is capable of being tested under full load conditions.

Uninterruptible AC power systems and DC power systems important to safety should be designed to be independent of the effects of DBAs to which they must respond and be fully functional during and following such accidents. A DC power supply division consists of one or more batteries, one or more battery chargers, a distribution system, and protection and isolation features. Each division of an uninterruptible AC power system should consist of an AC power supply and a DC power supply to an inverter, a separate AC power supply from the same division, and a feature to automatically switch between the inverter output and the separate AC supply.

The electrical power system design shall include provisions for mitigating the complete loss of onsite and offsite AC power. This is accomplished by the use of an alternate AC power supply consisting of onsite portable, transportable, or fixed power sources; or offsite portable or transportable power sources; or a combination of these. The alternate AC power source is connectable (but not normally connected) to the offsite or onsite standby and emergency AC power systems, diverse in design with respect to the standby/emergency power sources, available in a timely manner after onset of a station blackout, and capable of coping with a station blackout and bringing the plant to a safe shutdown state.

Applicable Codes and Standards

Additional information with respect to some guidance on electrical power systems can be found in the following codes and standards referenced in REGDOC-2.5.2:

- CSA Group, N290.5 [80], “Requirements for electrical power and instrument air systems of CANDU nuclear power plants,” (note: CSA N290.5 is a CANDU specific document which particularly addresses the two-group design philosophy).
- IAEA, NS-G-1.8 [81], “Design of Emergency Power Systems of Nuclear Power Plants,” Vienna, Austria, 2004.
- IAEA, SSG-34 [82], “Design of Electrical Power Systems for Nuclear Power Plants,” Vienna, Austria, 2016.
- IEEE Std. 141 [83], “IEEE Recommended Practice for Electric Power Distribution for Industrial Plants,” Piscataway, New Jersey, 1993.
- IEEE Std. 242 [84], “IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems,” Piscataway, New Jersey, 2001.

- IEEE Std. 308 [85], "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," Piscataway, New Jersey, 2001.
- IEEE Std. 387 [86], "IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations," Piscataway, New Jersey, 1995.
- IEEE Std. 279 [87], "IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations," Piscataway, New Jersey, 1971.

7.2.3 NRC Regulations, Guidance, and Standards

The regulations pertinent to electrical power systems are in 10 CFR 50, Appendix A, GDC 17, "Electric Power Systems." The regulations include provisions for an onsite electric power system and an offsite electric power system to be provided to permit functioning of structures, systems, and components important to safety, when applicable. The classification of power systems is dependent on the classification of the loads and what SSCs are needed to accomplish safety functions. Classifications for electrical power systems include Class 1E/safety-related/safety, non-Class 1E/nonsafety-related/non-safety, and important to safety. Important to safety could be either safety-related or nonsafety-related SSCs. However, for electrical, important to safety is typically the connection of the onsite AC distribution system with the offsite power sources and the transmission grid, which is treated as nonsafety-related with additional guidelines, such as redundancy. The figure below depicts the typical classification of electrical power systems.

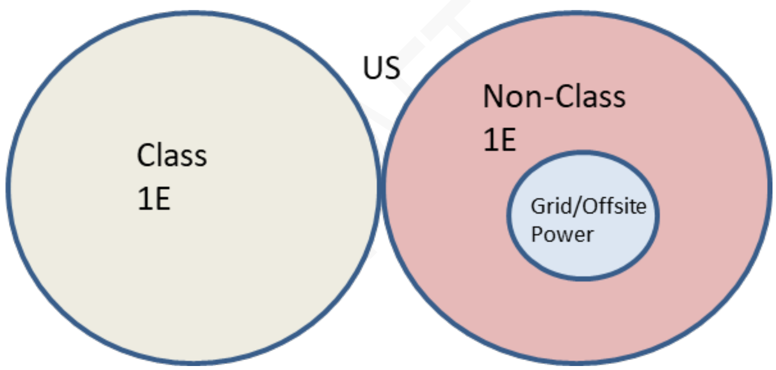


Figure 16: Relationship between Offsite Power and Internal Class 1E Distribution

GDC 17 states that the electric power systems shall have sufficient capacity and capability. Further, the onsite electrical power systems shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure. Additionally, for the offsite power system, grid stability is a consideration, and there shall be provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.
GDC 18, "Inspection and testing of electric power systems," relates to the periodic testing of components and systems.

Large LWR designs typically have Class 1E AC and DC onsite power systems, with the offsite power system being important to safety. Advanced reactor or SMR designs may not require Class 1E AC or DC power, depending on the design and electrical power requirements, as dictated by the safety analysis.

The term “station blackout” (SBO) refers to the complete loss of AC electric power to the essential and nonessential switchgear buses. An SBO; therefore, involves the loss of the offsite electric power system concurrent with a turbine trip and unavailability of the emergency AC power system. An SBO does not include the loss of available AC power to buses fed by Class 1E station batteries through inverters or by alternate AC (AAC) sources specifically provided for SBO mitigation. Because many LWR safety systems necessary for reactor core decay heat removal depend on ac power, an SBO could result in a severe core damage accident.

The SBO Rule at 10 CFR 50.63, “Station Blackout,” applies to LWRs and requires that the plant coping capability exceed the specified coping duration. The SBO coping duration is defined as the time from the onset of an SBO to the time when either offsite or onsite AC power is expected to be restored to at least one of the safe shutdown buses. The necessary coping duration is determined based on the redundancy and reliability of onsite standby/emergency ac power sources and the expected frequency and duration of loss of offsite power events. The SBO coping capability is the time from the onset of an SBO that plant equipment and implementing procedures support establishing and maintaining shutdown conditions of hot standby or hot shutdown. An AAC source capable of starting and powering at least one complete set of normal shutdown loads is one approach to ensure the coping capability exceeds the necessary coping duration. Passive plant designs that demonstrate all safety-related shutdown functions can be performed without reliance on ac power for 72 hours comply without redundant standby/emergency onsite power sources and an AAC source. Additional information is provided in RG 1.155, “Station Blackout.”

10 CFR 50.155, “Mitigation of beyond-design-basis events,” provides strategies and guidelines to mitigate beyond-design-basis external events from natural phenomena that are developed assuming a loss of all ac power concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink. an extended loss of ac power consists of a loss of offsite power affecting all units at a plant site and installed sources of emergency onsite ac power and SBO alternate ac power sources are assumed to be not available and not imminently recoverable. Portable, onsite equipment can be utilized until offsite resources can be obtained. Additional information is available in RG 1.226, “Flexible Mitigation Strategies for Beyond-Design-Basis Events.”

Chapter 8 of the SRP provides guidance for the review of electrical system designs typical of large LWRs. This SRP chapter addresses the offsite power system (Section 8.2), the onsite AC power system including standby power supplies (Section 8.3.1), the onsite DC power system including supporting uninterruptible power supplies for vital systems (Section 8.3.2), and Station Blackout coping capability (Section 8.4). In addition, Chapter 9 of the SRP provides guidance for the review of onsite standby/emergency diesel generator auxiliary systems that prescribe consumable fluid inventories to support 7 days of operation and stored energy capability for multiple start attempts.

The IEEE has developed many consensus standards, guides, and recommended practices related to electrical equipment and systems. The NRC endorses several IEEE standards, with exceptions and clarifications noted in regulatory guides.

Other Applicable Codes and Standards

The list below provides applicable high-level standards for important electrical systems. Additional standards and guidance are referenced in the Standard Review Plan.

- IEEE Std. 308, “IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations.”
- IEEE Std. 279-1971, “IEEE Standard Criteria for Protection Systems for Nuclear Power Generating Stations.”
- IEEE Std. 603-1991 [88], “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations.”
- IEEE Std. 741 [89], “IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations.”
- IEEE Std. 765 [90], “IEEE Standard for Preferred Power Supply (PPS) for Nuclear Power Generating Stations.”

7.2.4 Similarities and Differences

Both the CNSC and the NRC have established regulations and guidance and endorsed standards for reliable sources of electrical power to support performance of fundamental safety functions. The guidance and standards address electrical configurations used for water-cooled reactors that rely on active components to perform those safety functions. These electrical configurations include preferred sources of offsite power; AC onsite power distribution systems, including standby/emergency AC sources and distribution; DC onsite power distribution systems, including uninterruptible vital power systems; and alternate AC onsite power sources. In addition, both regulatory bodies recognize the connection between electrical distribution design and the role of the electrical system in responding to postulated events and achieving and maintaining safe shutdown. Consequently, the electrical distribution system may be designed commensurate with its importance to safety when the safety analyses rely less on the electrical distribution system for the mitigation of events.

For the preferred power supply or offsite power system, the pertinent regulations and guidance similarly specify: (1) at least two independent circuits, (2) acceptable switchyard configurations, (3) assessment of capacity and capability, and (4) the automatic availability of one circuit. The CNSC guidance specifically addresses house load operation, frequency transients, and transmission system studies, including the effects of solar magnetic disturbances, that are not specifically discussed in NRC guidance. However, NRC guidance does address transmission system stability in the context of withstanding transmission system transients, including loss of the largest connected generator.

For the onsite AC standby/emergency power systems, the pertinent regulations and guidance similarly specify independence, separation, diversity, and coordination of protection. Guidance for onsite standby/emergency power sources address automatic initiation, fuel storage, stored energy for starting, mission times, and testing. Beyond design-basis events are also addressed similarly with respect to the capability of the onsite AC distribution to support connection of alternate AC power sources assuming a loss of the normal preferred and standby/emergency sources.

For the onsite DC standby/emergency power systems and vital/uninterruptible power supplies, the pertinent regulations and guidance similarly specify independence, separation, diversity, and coordination of protection. Guidance for onsite batteries and other DC power sources address design features, capabilities, and provisions for testing. Mission times for batteries are established for beyond-design-basis conditions where supported equipment must operate to mitigate a loss of AC supplied sources of DC power.

Both regulatory bodies consider a loss of all AC power as a beyond-design-basis condition for plants that rely on AC power to support systems performing fundamental safety functions. The NRC regulation at 10 CFR 50.63 establishes risk-informed requirements to cope with station blackout events at LWRs, including the potential to identify a fixed alternate AC power supply to reduce the likely duration of a station blackout. Guidance provided in RG 1.155, "Station Blackout," provides expected design rules and specifications for the alternate AC power supply that specify limited quality controls and no specific seismic qualification. The NRC established an additional risk-informed and performance-based regulation at 10 CFR 50.155 for mitigation of BDBEs, which requires all nuclear power reactor operating license holders to develop, maintain, and implement strategies for core cooling, containment function, and spent fuel cooling. These strategies often incorporate the use of portable electric generators. Similarly, Section 6.9.3 of REGDOC-2.5.2 specifies provision of mitigating the beyond-design-basis complete loss of onsite and offsite AC power sources using portable, transportable, or fixed generators that are evaluated against performance criteria. Both regulatory bodies specify designs that minimize potential for common-cause failures and are available in a timely manner after the onset of a station blackout.

7.2.5 Assessment of Impact

Overall, the CNSC and the NRC regulation of electrical power distribution systems is very similar and commensurate with the safety significance of the system. Both regulatory bodies provide requirements and guidance related to preferred offsite power sources; onsite AC power distribution systems, including standby/emergency AC power; onsite DC power distribution, including vital/uninterruptible instrument control power; and alternate AC power sources that may be fixed, portable, or transportable.

The engineering design rules and specifications for electrical power systems are risk-informed and adaptable to advanced reactors and passive SMR designs where the safety significance of AC distribution systems and sources is reduced. Portions of the electrical power sources and distribution meeting the criteria of a CNSC supporting safety system would be classified as safety-related under NRC regulations, and therefore, would be subject to similar design rules, such as consideration of single failures, and quality control measures. Other important to safety portions of the electrical power and distribution systems that principally provide DID functions would also be subject to similar design rules and graded quality assurance measures. The CNSC and the NRC have endorsed many of the same international standards for the detailed design of power sources and electrical distribution systems. Therefore, common application information addressing the design, including application of engineering design rules and specifications, is expected to be acceptable for both regulatory bodies.

7.3 Instrumentation and Control

7.3.1 Overview

This section covers code classification and assignment of engineering design rules for instrumentation and control (I&C) systems. The reliability and robustness of the system architecture and adherence to consensus standards is also evaluated in making a determination of the design adequacy.

7.3.2 CNSC Regulations, Guidance, and Standards

Regulatory Framework

In Canada, the Class I Nuclear Facilities Regulations require that applications contain a description of the systems and equipment to be installed, including their design and design operating conditions, but there are no specific regulations related to I&C systems. Consequently, the CNSC uses regulatory documents and licence conditions as the means to establish both requirements and guidance.

CNSC REGDOC-1.1.2 sets out requirements and guidance on submitting an application 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 I&C systems in order to meet the expectations of REGDOC-2.5.2, including general I&C design capabilities, design for reliability, and human factors.

Section 5.1, "Safety classification of structures, systems and components," of REGDOC-2.5.2 describes that all SSCs shall be identified as either important or not important to safety. This section provides general design requirements and guidance on safety classification of SSCs. The design bases, design criteria, regulatory documents, standards, and other documents that will be used to design the I&C systems should be specified.

Section 5.6, "Design for reliability," of REGDOC-2.5.2 describes that all SSCs shall be designed with sufficient quality and reliability to meet the design requirements. This section provides general design requirements and guidance especially important to I&C systems. The section specifically addresses avoidance of common-cause failures, separation, diversity, independence, fail-safe design, limits on sharing of instrumentation between systems, and provisions supporting online maintenance and testing.

Section 5.9, Instrumentation and control, of CNSC REGDOC-2.5.2 specifies the following design capabilities as generally applicable to I&C systems:

- monitor plant variables affecting the fission process, the integrity of the reactor core, the reactor cooling systems, and containment, over the respective ranges for normal operational states, DBAs, and DECs;
- operate safety systems and safety support systems reliably and independently, either automatically or manually, when necessary;
- provide for testing, including self-checking capabilities and the capability to periodically test the entire channel of instrumentation logic from sensing device to actuating device;
- provide reliable controls for maintaining plant variables within specified operational ranges;
- provide adequate capability to measure plant parameters for emergency response purposes;
- minimize the likelihood of inadvertent manual or automatic override, while providing for situations when it is necessary to override interlocks to use equipment in a non-standard way; and
- automate various safety actions so that operator action is not necessary within a justified period of time after the onset of an AOO or DBA.

Section 5.9.1 of REGDOC-2.5.2 provides additional guidance related to the general design of I&C systems. This section includes guidance addressing the design of safety systems for reliability, automatic initiation of safety functions, and continuation of safety system actuation once initiated.

Section 5.9.2, Use of computer-based systems or equipment, of REGDOC-2.5.2 provides the following additional expectations for computer-based functions that are important to safety:

- Functions not essential to safety are separate from important to safety functions
- Safety functions are executed in processors separate from software that implements control, monitoring, or display functions.
- Computer-based systems performing similar safety functions are justifiably diverse.
- The design incorporates fail-safe and fault-tolerant features.

Section 5.9.3, Accident monitoring instrumentation, of REGDOC-2.5.2 specifies that instrumentation providing the following information essential for implementation of plant procedures be available during and following DBAs and DECs:

- plant status;
- locations and estimated quantities of radioactive material;
- vital plant parameters; and
- factors facilitating accident management.

Section 5.21, Human factors, of REGDOC-2.5.2 addresses selection of the displayed information necessary to support operator management of the plant in operational, DBA, and DEC states.

Codes and Standards:

The following I&C related international codes and standards are among those referenced in CNSC REGDOC-2.5.2:

- IEEE Std. 603, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” Piscataway, New Jersey, 2009.
- IEEE Std. 7-4.3.2 [91], “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” Piscataway, New Jersey, 2010.
- International Electrotechnical Commission (IEC), IEC 61226 [92], “Nuclear Power Plants – Control Functions, Geneva,” 2009.
- IEEE Std. 497 [93], “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” Piscataway, New Jersey, 2010.
- CSA Group, CSA N290.6 [94], “Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident”.

7.3.3 NRC Regulations, Guidance, and Standards

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. More specifically, 10 CFR 50.55a, “Codes and Standards,” requires that applicants for CPs, OLs, DCs, SDAs, and COLs regarding nuclear power plants of all types meet the requirements for safety systems in IEEE Standard 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations.” The definition of safety system in IEEE Std 603 is equivalent to the definition of safety-related SSC in 10 CFR 50.2.

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. Appendix A to 10 CFR Part 50 includes the following GDC directly relevant to I&C:

- GDC 13—"Instrumentation and control."
- GDC 20—"Protection system functions."
- GDC 21—"Protection system reliability and testability."
- GDC 22—"Protection system independence."
- GDC 23—"Protection system failure modes."
- GDC 24—"Separation of protection and control systems."
- GDC 29—"Protection against anticipated operational occurrences."

In accordance with 10 CFR 50.34(f), applicants for DCs, SDAs, COLs, and MLs must demonstrate compliance with technically relevant portions of TMI-related requirements[17]. These requirements include provisions such as consideration of human factors engineering in the control room design, inclusion of displays indicating safety parameter and core cooling status, instrumentation to monitor containment status, and post-accident monitoring capability.

Based on the above requirements, the scope of the regulatory review of a nuclear power plant design includes the I&C systems that are needed to monitor variables and systems over their anticipated operating ranges, to maintain these variables and systems within their prescribed operating ranges, to automatically initiate the operation of systems and components to assure that fuel design limits are not exceeded because of AOOs, to sense accident conditions and initiate the operation of safety systems, and provide important to safety equipment to monitor sources of radioactivity that may be released to the environment under normal operating or accident conditions. Primarily, I&C systems provide essential support functions to the SSCs that perform fundamental safety functions of preventing or mitigating the consequences of accidents, which could result in substantial offsite exposures. These I&C systems are classified as safety-related under both traditional and LMP approaches.

An important to safety class of SSCs, which is defined by deterministic or risk-informed criteria to perform other functions identified in the PDC or to support DID, are classified as nonsafety-related with additional regulatory or special treatment. Graded nonsafety-related classification is applied to the I&C systems that provide essential support to safety-significant functions based on their required regulatory or special treatment. Hence the I&C systems are classified consistent with the significance of the assigned safety functions.

The NRC staff uses the applicable review guidance (e.g., Section 7 of the SRP; Section 7 of a Design Specific Review Standard (NUREG-0800, Part 2) [95]; and/or Design Review Guide (DRG), “Instrumentation and Controls for Non-Light Water Reactor (Non-LWR) Reviews”[96]).

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[17] SECY-22-0052 (ML21159A055) describes 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 [RIN 3150 AI66; Regulations.gov Docket: NRC-2009-0196]. The NRC is proposing to include applicants for CPs and OLs under 10 CFR Part 50 among the applicants that must demonstrate compliance with technically relevant portions of TMI-related requirements per 10 CFR 50.34(f).

For example, the NRC staff uses the DRG to assess whether the applicant demonstrates how the specified I&C systems support the overall nuclear power plant performance objectives for a particular plant design. The reviewer considers the systematic assessment used in the application to assess the adequacy of the I&C architecture and systems design. The reviewer considers whether the assessment provides assurance that the I&C design is reliable and robust by demonstrating that: (1) the design criteria and testing and qualification requirements have been met and (2) credible hazards and failure modes of the design are identified and controlled.

The reviewer focuses on verifying the applicable attributes of the I&C system design that support the plant level performance objectives as depicted in *Figure 17*. The I&C performance objectives are typically achieved through demonstrating that the I&C architecture and systems are sufficiently reliable and robust commensurate with their safety significance:

1. "Reliability" of the I&C design is the probability that a system or component will meet its functional requirements under defined plant conditions. Reliability is achieved using quantitative and qualitative performance measures and criteria. These measures and criteria include but are not limited to surveillance tests, verification and validation, failure data, self-diagnostic features, and fail-safe design. The I&C quantitative reliability goals should be aligned with the plant's PRA and other risk assessment results.

2. "Robustness" of the I&C design is the degree to which a system or component can function correctly in the presence of invalid inputs or stressful environmental conditions. Robustness is achieved by having various measures of DID and qualification.

The I&C design should ensure that the I&C equipment or components can be qualified, procured, installed, commissioned, operated, and maintained to be capable of withstanding, with sufficient reliability and robustness, all conditions specified in the plant design basis. Subsequent NRC staff review steps focus on those safety-significant functions and the SSCs selected to meet those functions. Safety-significant functions include those classified as risk-significant or credited for DID. The overall purpose of the NRC staff evaluation is to confirm that the safety-significant functions, and the corresponding SSCs, adequately support the overall plant level or I&C system level performance objectives. For SSCs that the NRC staff determines are not safety-related and do not receive special treatment, the design-related review may be less detailed or lower in depth than the review of safety-related SSCs. Specifically, the staff review focuses on ensuring that these SSCs will not adversely impact safety-related I&C SSCs and I&C SSCs that are not safety-related but warrant special treatment in their performance of safety-significant functions.

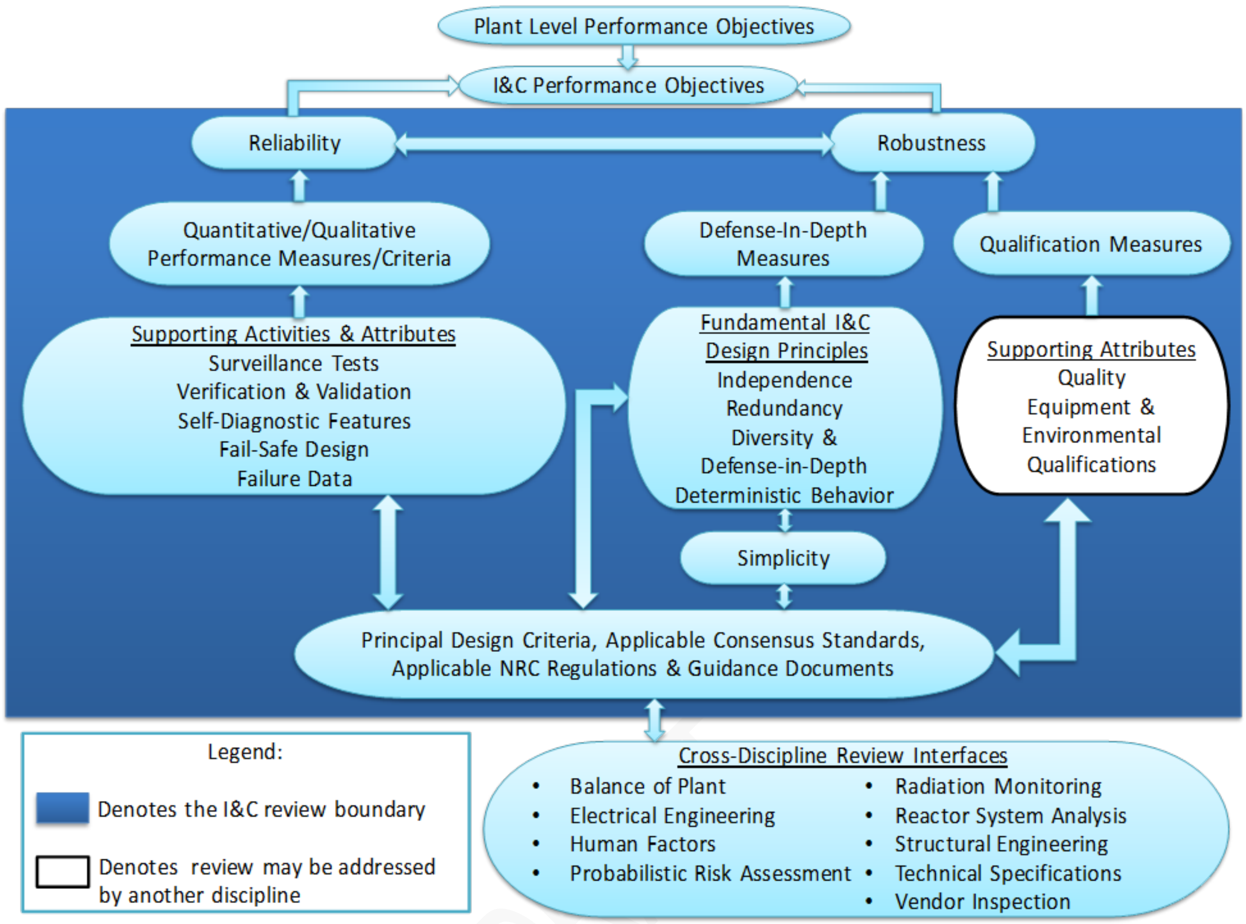

Figure 17: I&C System Review Framework

The I&C design reviewer also confirms that the applicant has established the appropriate set of PDC, applicable industry consensus standards, and applicable NRC regulatory guidance documents that are being used to ensure that the performance measures, adequacy of DID, and qualification measures are met. The reviewer confirms that the applicant has met the applicable regulations or requested appropriate exemptions if necessary. The reviewer also coordinates with the appropriate cross-discipline interfaces in order to verify that any cross-discipline issues are adequately identified and resolved.

The NRC regulatory guides (including endorsed industry codes and standards) provide guidance for the applicants and licensees on an acceptable way to meet the applicable regulatory requirements. *Table 20* maps the I&C design technical review areas, associated regulations, and regulatory guides. These same set of regulations apply to the I&C design developed under either the deterministic, deterministic-risk-informed, or the risk-informed performance-based LMP framework. These regulatory requirements are graded for application to nonsafety-related but important to safety I&C systems based on their risk significance.

Table 20: Mapping of I&C Review Areas to Regulations and Guidance

I&C Technical Review Area	Regulatory Requirements	Regulatory Guidance NOTE
Criteria for Safety Systems	10 CFR Part 50, Appendix A GDC 1, 2, 4, 13, 19, 20, 21, 22, 24, 25, 34 10 CFR Part 50, Appendix B Quality Assurance Criteria 10 CFR 50.55a(h)	RG 1.22 RG 1.30 (endorses IEEE Std 336) RG 1.47 RG 1.53 (endorses IEEE Std 379) RG 1.62 RG 1.75 (endorses IEEE Std 384) RG 1.118 (endorses IEEE Std 338) RG 1.153
Criteria for Safety System Programmable Digital Devices	10 CFR Part 50, Appendix A GDC 21 10 CFR Part 50, Appendix B Quality Assurance Criteria 10 CFR 50.55a(h)	RG 1.152 (endorses IEEE Std 7-4.3.2)
Digital Development & Reliability	10 CFR Part 50, Appendix A GDC 1, 21 10 CFR Part 50, Appendix B Quality Assurance Criteria 10 CFR 50.55a(h)	RG 1.168 (endorses IEEE Std 1012 & IEEE Std 1028) RG 1.169 (endorses IEEE Std 828) RG 1.170 (endorses IEEE Std 829) RG 1.171 (endorses IEEE Std 1008) RG 1.172 (endorses IEEE Std 830) RG 1.173 (endorses IEEE Std 1074) RG 1.152 (endorses IEEE Std 7-4.3.2)
Commercial-Grade Dedication	10 CFR Part 50, Appendix A GDC 1, 21 10 CFR Part 50, Appendix B Quality Assurance Criteria 10 CFR 50.55a(h)	RG 1.164 (endorses EPRI 3002002982, which references EPRI TR-106439 and EPRI TR-107330 for digital equipment) RG 1.250 (endorses NEI 17-06, IEC 61508, and ISO/IEC 17065) RG 1.152 (endorses IEEE Std 7-4.3.2)
Equipment Qualification	10 CFR Part 50, Appendix A GDC 1, 2, 4,13, 21, 23 10 CFR Part 50.49 10 CFR 50.55a(h)	RG 1.89 (endorses IEC/IEEE 60780-323) RG 1.209 (endorses IEEE Std 323-2003) RG 1.100 (endorses IEEE Std 344, IEEE Std C37.98, ASME QME-1) RG 1.180 (endorses IEEE Std 1050, IEEE Std C62.45, IEEE Std C62.41.1, IEEE Std C62.41.2, MIL-STD-461G, IEC 61000-3, IEC 61000-4, IEC 61000-6) RG 1.152 (endorses IEEE 7-4.3.2)
Accident Monitoring Instrumentation	10 CFR Part 50, Appendix A, GDC 19 10 CFR Part 50.34(f)(2)(xix) 10 CFR 50.155(e)	RG 1.97 (endorses IEEE Std 497)
Setpoint Establishment & Maintenance	10 CFR Part 50, Appendix A GDC 13, 20 10 CFR Part 50.36(c)(1)(ii)(A) 10 CFR 50.55a(h)	RG 1.105 (endorses ANSI/ISA-67.04.01)
Instrument Sensing Lines	10 CFR Part 50, Appendix A GDC 1, 2, 13, 21, 22, 23, 24 10 CFR Part 50.36(c)(2)(ii)	RG 1.151 (endorses ANSI/ISA-67.02.01, IEEE Std 622)
Cybersecurity	10 CFR Part 73 (programmatic controls)	RG 5.71 RG 5.83
Note: This table provides a listing of NRC RGs that include references to endorsed standards and that are available at: https://www.nrc.gov/reading-rm/doc-collections/reg-guides/index.html.		

7.3.4 Similarities and Differences

The approach to I&C SSC classification and designation of engineering design rules and specifications is similar among the CNSC approach and the NRC Traditional and LMP approaches. The design criteria for all approaches address normal operational states (including AOOs), design basis accident conditions (i.e., DBAs for CNSC and NRC traditional; DBEs and DBAs for LMP), and beyond-design-basis conditions included in the licensing basis (i.e., CNSC DECs, NRC traditional regulated events, and BDBEs under LMP).

The scope of functions performed by I&C systems and the engineering design rules applied to these systems is similar. The design evaluations for both regulatory bodies consider control functions supporting normal operation, safety systems to address accident conditions, human factors engineering principles in the design of important I&C systems, and post-accident monitoring capability.

For both the NRC and the CNSC regulatory frameworks, the classification and application of engineering design rules are driven by the functional importance to safety and risk significance of the systems and components that are supported by the I&C systems. The similarities in engineering design rule application include (1) I&C system safety criteria and fundamental I&C design principles, (2) development and reliability of digital I&C systems, (3) requirements for addressing software common-cause failures (4) digital I&C system real-time software quality assurance, (5) secure digital I&C system development and operational environment, (6) accident monitoring instrumentation, and (7) setpoint establishment and maintenance. While there are differences in endorsed standards and the specification of a quantitative reliability criterion for CNSC safety systems, both regulatory bodies are familiar with standards developed and issued by the IEEE and the IEC. Therefore, it is expected that common approaches that rely on these standards may be developed for applications to both regulatory bodies.

7.3.5 Assessment of Impact

Overall, the Canadian and U.S. classifications and design rules for I&C systems are very similar. In both countries, the I&C design rules are in accordance with the applicable IEEE and IEC standards. Significant emphasis is placed by both countries on formal methods for developing the real-time software based digital I&C systems, including all the lifecycle activities from conception to retirement. There are minor differences or nuances in the regulations or guidance. Both the CNSC and the NRC are familiar with and have endorsed many IEC and IEEE standards, and the IEEE and IEC standards development organizations develop and issue many joint standards. Both regulators accept unendorsed standards with appropriate justification. Therefore, the similar scope of regulatory review and the commonality of international standards, including application of engineering design rules and specifications, support the development of common application information likely to be acceptable to both regulatory bodies.

7.4 Civil Structures

7.4.1 Overview

This section covers safety classification and assignment of engineering design rules for structures that contain SSCs performing important to safety functions or otherwise contribute to those functions. Such structures include concrete containments, steel containments, and other concrete or steel structures that perform one or more of the following ITS functions:

- protection of ITS SSCs from internal and external hazards
- containment of radioactive materials
- structural supports
- radiation shielding

National consensus standards provide sets of engineering design rules applied to the construction of concrete and steel structures. Construction, as used here, is an all-inclusive term that includes material selection, design, fabrication, installation, examination, testing, inspection, and certification. The analysis and design of structures cover static and dynamic loads, load factors, load combinations, and safety criteria for both normal service loads and abnormal/environmental loads.

7.4.2 CNSC Regulations, Guidance, and Standards

In Canada, the Class I Nuclear Facilities Regulations require that applications contain plans showing the locations of structures of the nuclear facility and describe the design and design characteristics of those structures. The CNSC uses regulatory documents as the means to establish the detailed information necessary to support the application.

CNSC REGDOC-1.1.2 sets out requirements and guidance on submitting an application to the CNSC to obtain a licence to construct a reactor facility in Canada. To the extent practicable and commensurate with the state of the design, the application should describe the site layout and design of civil engineering works and structures with sufficient detail to show that the design is in accordance with the sections addressing civil structures and containment design within REGDOC-2.5.2. This description should include:

- design and analysis procedures, including boundary conditions and computer codes;
- design basis information and applicable codes and standards;
- the safety classification and seismic classification of each structure containing important to safety SSCs;
- the design loading combinations and demonstration of sufficient safety margins; and
- any design considerations or mitigation measures included to deal with BDBEs.

Section 5.15, "Civil structure," of REGDOC-2.5.2 states that:

- The facility design shall specify the required safety function performance of the civil structures in operational states, DBAs, and DECs.
- Civil structures shall be designed and located to minimize the probabilities and effects of internal hazards and consider external hazards and natural phenomena in their design.
- Civil structures important to safety shall be designed to meet serviceability, strength, and stability requirements for all load combinations under the categories of normal operation, AOO, DBA, and DEC, including the effects of external hazards.

Section 6.6, "Containment and means of confinement," of REGDOC-2.5.2 states that each nuclear reactor facility shall have a reactor containment structure to minimize the release of radioactive material to the environment during operational states and DBAs and that the containment will assist in mitigating the consequences of DECs. Section 6.6.12 of REGDOC-2.5.2 addresses design extension condition considerations for containment design, including demonstrating that the containment structure can withstand the loads associated with accident heat sources, combustion of gases, and generation of non-condensable gases.

Concrete containment structures should be designed and constructed in accordance with CSA N287.3 [97], “Design requirements for concrete containment structures for nuclear power plants,” and other standards in the series. The CSA 287 series of standards was originally written for CANDU Structures and can be applied to other containment structures that are made entirely or in part of concrete. This standard is organized around the design, construction, examination, testing and aging management of nuclear containment concrete structures. It also references other pertinent standards, and REGDOC-2.5.2 references the following standards and guidance for further information:

- American Concrete Institute (ACI), 349 [98], “Code Requirements for Nuclear Safety-Related Concrete Structures & Commentary.”
- ASME, BPVC, Section III, Division 2 [99], “Code for Concrete Containments.”
- U.S. NRC SRP (NUREG-0800), Section 3.8.1, “Concrete Containment.”

Steel containment structures should be designed according to the ASME BPVC, Section III, Division 1, Subsection NE, Class MC Components or equivalent standard. Stability of the containment vessel and appurtenances should be evaluated using ASME Code Case N-284-1, *Metal Containment Shell Buckling Design Methods.*

Important to safety structures other than containment should be designed and constructed in accordance with CSA N291 [100], “Requirements for nuclear safety-related structures.” This standard specifies requirements for the material, analysis and design, construction, fabrication, inspection, examination, and aging management of safety-related structures constructed of structural steel, reinforced concrete, and reinforced masonry. This standard also covers the design and analysis of irradiated fuels and radioactive waste storage structures. The minimum design requirements specified in this standard reference non-nuclear CSA construction standards and the National Building Code of Canada (NBCC) [101].

The NBCC sets out technical provisions for the design and construction of new buildings. It also applies to the alteration, change of use and demolition of existing buildings.” It is referenced in the nuclear civil structure codes and regulations and is present to cover conventional aspects of the design such as foundations, site preparation, connections, etc. The NBCC is meant to promote consistency among the provincial and territorial building codes and is often consulted in conjunction with local codes and by-laws during design.

Structures covered by the CSA N291 standard include:

- structures that support, house, or protect safety systems in nuclear power plants;
- structures or their elements, where failure of the structures or their elements, would impede the function of safety systems;
- structures and their elements that are required for the safe operation and/or safe shutdown of the reactor in nuclear power plants;
- structures for the storage of wet irradiated fuel; and
- structures for the storage of dry irradiated fuel and other radioactive waste material.

REGDOC-2.5.2 lists the following structures that should typically be designed consistent with CSA N-291:

- internal structures of reactor building;
- service (auxiliary) building;

- fuel storage building;
- control building;
- diesel generator building;
- containment shield building, if applicable;
- other safety-related structures defined by the design; and
- turbine building (for boiling water reactors).

7.4.3 NRC Regulations, Guidance, and Standards

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.

As discussed in Section 3.2.2 of this report, NRC regulations require that applicants describe the PDC applicable to ITS structures. The applicable PDC for LWR applications that pursue a traditional licensing approach are:

- GDC 1, “Quality standards and records.”
- GDC 2, “Design bases for protection against natural phenomena.”
- GDC 4, “Environmental and dynamic effects design bases.”
- GDC 16, “Containment design.”
- GDC 50, “Containment design basis.”

In addition to the listed GDC, the following regulations must be considered:

- 10 CFR Part 50, Appendix S, which requires that safety-related SSCs be designed to withstand a safe shutdown earthquake (SSE). RG 1.29, “Seismic Design Classification for Nuclear Power Plants,” defines list of typical large LWR SSCs that should be designed to withstand an SSE and designates them as seismic Category I.
- 10 CFR 50.44(c)(5), which requires that analysis of containment structures around water-cooled reactors that do not contain inert atmospheres demonstrate that containment integrity would be maintained following burning of combustible gas generated by a severe accident.
- 10 CFR 50.44(d), which requires that applicants for non-water-cooled reactors for which accidents involving combustible gas generation are technically relevant demonstrate that the safety impacts of combustible gas have been addressed to ensure adequate protection of public health and safety and common defense and security.
- 10 CFR 20.1101(b), which provides that licensees shall use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as reasonably achievable.

The NRC has developed guidance applicable to LWRs that provides acceptable methods to comply with the above listed regulations. The following SRP Sections are most applicable to the design of civil structures, including containment, and would generally apply to non-LWR structures as well:

- SRP Section 3.8.1, “Concrete Containment”
- SRP Section 3.8.2, “Steel Containment”
- SRP Section 3.8.4, “Other Seismic Category I [Safety-Related] Structures”

- SRP Section 3.8.5, “Foundations”

Representative major plant structures listed in SRP Section 3.8.4 include the containment enclosure building, the auxiliary building, the fuel storage building, the control building, and the diesel generator building. The principal consensus construction standards used for these structures are:

- American Institute of Steel Construction (AISC) N690 [102], “Specification for Safety-Related Steel Structures for Nuclear Facilities.”
- American Concrete Institute (ACI), 349, “Code Requirements for Nuclear Safety-Related Concrete Structures & Commentary.”
- American Society of Civil Engineers (ASCE) 37 [103], “Design Loads on Structures During Construction,” American Society of Civil Engineers.”

The principal construction standards for containments are:

- ASME BPVC, Section III, Division 2, “Code for Concrete Containments.”
- ASME BPVC, Section III, Division 1, Subsection NE, Class MC, “Rules for Construction of Nuclear Facility Components,” for metal containment vessels.

Table 21 provides a crosswalk of referenced guidance mapped to the important to safety functions performed by civil structures. Applicants of non-LWR reactors may leverage this guidance along with RG 1.232 to inform the technology-specific PDC. It is important to note that the guidance is applicable under a traditional or LMP approach. That is, the safety-related design and quality assurance criteria applies to those structures classified as SR under either approach. A summary of the applicability of engineering design rules for SR and NSRST SSCs under LMP is included in Table 6.

Table 21: Mapping of NRC Civil Structure Review Areas to Guidance and Standards

Basis of Design and Construction Aspect	Important to Safety Functional Review Area				
	Internal events mitigation	External events mitigation	Containment	Combustible gas impacts	Radiation protection
Design guidance NOTE	RG 1.142	RG 1.76 RG 1.221 RG 1.102 RG 1.142 RG 1.199 RG 1.217	RG 1.57 RG 1.84 RG 1.136	RG 1.7 RG 1.216	RG 1.69 RG 1.143
Construction code	ACI 349-13 ASCE/SEI 37-14 ANSI/AISC N690	ACI 349-13 ASCE/SEI 37-14 ANSI/AISC N690	ASME BPV Code, Section III, Division 1 - Subsection NE (Steel Containment) ASME BPV Code, Section III, Division 2 (Concrete Containment) ANSI/AISC N690	ASME BPV Code, Section III, Division 1 - Subsection NE (Steel Containment) ASME BPV Code, Section III, Division 2 (Concrete Containment)	ACI 349-06 ACI 349.1R-07
Seismic classification	RG 1.29	RG 1.29	RG. 1.29	None specified for BDBE	N/A
Seismic inputs	N/A	10 CFR 50 Appendix S RG 1.208	N/A	N/A	N/A
Applicable consensus standards		ASCE 4 ASCE 43			ANSI/ANS-6.4-2006
Note: This table provides a listing of NRC RGs that include references to endorsed standards and are available at: https://www.nrc.gov/reading-rm/doc-collections/reg-guides/index.html.					

7.4.4 Similarities and Differences

The approach to civil structure classification and designation of engineering design rules and specifications is similar among the CNSC approach and the NRC Traditional and LMP approaches. The design criteria for all approaches address normal operational states (including AOOs), design basis accident conditions (i.e., DBAs for CNSC and NRC traditional; DBEs and DBAs for LMP), and beyond-design-basis conditions included in the licensing basis (i.e., CNSC DECs, NRC traditional regulated events, and BDBEs under LMP).

The scope of functions performed by civil structures is similar. The design evaluations for both regulatory bodies consider loading conditions imposed by a spectrum of external events, internal events, and natural phenomena. Load combinations are developed where the acceptable loadings increase as the likelihood of the load combination decreases. Requirements and guidance address capabilities to withstand beyond-design-basis conditions, notably combustible gas burns within containment, to ensure acceptable protection of the public health and safety. The structural analyses consider the methods used, boundary conditions, and appropriate acceptance criteria related to the integrity and stability of structures under load.

For both the NRC and the CNSC regulatory frameworks, the classification and application of engineering design rules are driven by the functional importance to safety and risk significance of the systems and components that are supported by the civil structures, as indicated by the listing of typical structures designed to nuclear- rather than civil- standards. The similarities in engineering design rule application include common consideration of ASME B&PV Code, Section III, standards for the design of steel and concrete containments, although the CNSC has endorsed the CSA Std. N287 series for the design and construction of concrete containments.

For civil structures other than containment, there is little commonality with respect to specific codes and standards endorsed for structural design and analysis. The NRC endorses nuclear-specific standards (i.e., ACI-349 and ANSI/AISC N690) for the design of safety-related structures, and these standards include provisions for foundation design and other supporting elements. However, the corresponding CSA Std. N291-19 standard references standard civil codes, including the NBCC, for foundation design and other structural supporting elements. A full comparison of the content of these standards is beyond the scope of this report, but both regulators expect differences in standards from those endorsed for the particular purpose to be addressed to an extent commensurate with the safety importance of the civil structure.

7.4.5 Assessment of Impact

Overall, the CNSC and the NRC approaches to classification and assignment of design rules for civil structures share many commonalities, especially with regard to containment design. In both regulatory bodies, structures performing similar functions have been identified for rigorous review. Significant emphasis is placed by both countries on evaluation of methods of analysis for civil structures that is comparable. Both the CNSC and the NRC are familiar with and have endorsed ASME B&PV Code standards for design of containment, and application materials based on these standards are likely to be useable with both regulatory bodies. For other civil structures, both regulators accept unendorsed standards with appropriate justification. Therefore, the development of common application information to a specified set of standards is likely to be useable in applications to both regulatory bodies, with appropriate justification for the selected standard based on the safety significance of the associated civil structures.

8 Engineering Design Rules for Hazard Protection

8.1 Seismic Design Rules

8.1.1 Overview

This section covers the rules and regulations that drive the design of important to safety SSCs to provide reasonable assurance that they will be able to perform their safety functions following a design basis earthquake. This process considers the safety classification of SSCs, credible interactions that could affect the safety function, and the design considerations that provide assurance that the safety function would be satisfied following a design basis earthquake with acceptable margin. The methods of determining the design basis earthquake and associated ground motion are beyond the scope of this report.

8.1.2 CNSC Regulations, Guidance, and Standards

In Canada, the Class I Nuclear Facilities Regulations require that applications to construct contain a description of the environmental baseline characteristics of the site and the safety analysis demonstrating the adequacy of the design. The CNSC uses regulatory documents as the means to establish the detailed information necessary to support the application. Section 5.13, “Seismic qualification and design,” of REGDOC-2.5.2 stipulates the following high-level requirements associated with the seismic design of SSCs:

- A design basis earthquake is defined as a part of the abnormal or extreme load category corresponding to a DBA.
- SSCs meeting any of the following criteria shall be seismically qualified to retain all essential attributes (e.g., pressure boundary integrity, leak tightness, operability, and proper position) in the event of a design basis earthquake:
    1. SSCs whose failure could directly or indirectly cause an accident leading to core damage,
    2. SSCs restricting the release of radioactive material to the environment,
    3. SSCs that assure the subcriticality of stored nuclear material, or
    4. SSCs such as radioactive waste tanks containing radioactive material that, if released, could exceed regulatory dose limits.
- The design ensures that no substantive damage would be caused to the SSCs meeting the above criteria by the failure of any other SSC under DBE conditions.
- The seismic qualification (SQ) of all SSCs shall meet the requirements of Canadian national (e.g., CSA N289 series) or equivalent standards.
- Seismic fragility levels shall be evaluated for SSCs important to safety by analysis or, where possible, by testing.

A beyond-design-basis-earthquake is defined for identification of DECs, and SSCs selected to function in these DECs shall be demonstrated to provide a high confidence of low probability of failure (HCLPF).

Standards and methodologies:

The following CSA N289 series standards provide requirements (e.g., seismic hazard evaluation, seismic design, qualification, evaluation, and testing and monitoring) to implement high-level seismic requirements that are discussed in the CNSC REGDOC-2.5.2. The following CSA N289 series standards are used in conjunction with each other:

- CSA N289.1 [104], General requirements for seismic design and qualification of nuclear power plants;
- CSA N289.2, Ground motion determination for seismic qualification of nuclear power plants;
- CSA N289.3, Design procedures for seismic qualification of nuclear power plants;
- CSA N289.4, Testing procedures for seismic qualification of nuclear power plant structures, systems, and components; and
- CSA N289.5, Seismic instrumentation requirements for nuclear power plants and nuclear facilities.

The CSA N289 series standards have been developed for areas with low to moderate seismicity, like those where NPPs are built in Canada.

CSA N289.1 defines that the design basis earthquake is "*an engineering representation of potentially severe effects at the site due to earthquake ground motions having a selected mean probability of exceedance of 1 × 10–4 per year, or such a probability level as is acceptable to the regulatory authority,*" which is consistent with REGDOC-2.5.2 guidance. Seismic ground motions, from the DBE, should be based on seismicity and geologic conditions at the site and expressed in such a manner that it can be applied for the qualification of SSCs.

Seismic classification is a necessary design activity for all SSCs, but special emphasis is paid to those that are required to shut down the plant, cool the fuel, remove decay heat, and maintain a containment boundary. The following two seismic categories are specified in CSA N289.1 to identify the extent to which SSCs shall remain operational during and/or after an earthquake:

- Seismic Category A includes those SSCs that shall maintain their structural integrity and retain their pressure boundary integrity during and/or following an earthquake.
- Seismic Category B includes those SSCs that shall maintain their structural integrity and detailed functional requirements during and/or following an earthquake, and shall also retain their pressure boundary integrity, where applicable.

In Canada, for pressure boundary SSCs, the design basis earthquake is classified as an ASME Level C service loading condition as per the ASME BPVC, Section III, Division 1, NCA-2142. Seismic-induced loads alone (see Table 1 for loads for the SQ) are considered in the seismic qualification by analysis. Therefore, loads induced by a loss-of-coolant accident are not combined with the seismic-induced loads in level C.

CSA N289.1 also provides the requirements for post-earthquake actions and activities in order to establish a plant's condition, operability and adherence to its licensing basis.CSA N289.2 was developed to determine the appropriate seismic ground motion parameters for Eastern North American regions of low to moderate seismic hazard, comparable to the levels near Canada's existing nuclear power plants. This standard describes the investigations required to obtain the seismological and geological information necessary to determine the seismic ground motion that will be used in seismic qualification of safety-related nuclear power plant SSCs, and the potential for secondary earthquake effects (e.g., tsunami, seiche, volcanism, slope instability, surface faults, surface instability, and dam failures) that can have a direct or indirect effect on plant safety or operation. It also provides requirements and guidance for site response analysis.

CSA N289.3 provides requirements for seismic qualification by analysis. This includes a comprehensive set of requirements that address the different methods that can be used (e.g. simplified static, response spectrum and time domain analyses); decoupling criteria; requirements for pressure boundary components in conjunction with CSA N285 and the ASME BPVC; and soil-structure analyses.

CSA N289.4 provides the requirements for seismic qualification done by testing. The purpose of this standard is to provide a basis f or the development of specifications for seismic qualification by testing and to assist purchasers, suppliers, and testing laboratories in selecting the appropriate test method(s) for performing acceptable seismic qualification tests that meet a quality and standard commensurate with the safety principles necessary to comply with the Canadian nuclear safety philosophy.

CSA N285.0 provides general requirements for pressure-retaining systems and components in CANDU nuclear power plants/Material Standards for reactor components for CANDU nuclear power plants. The seismic design of pressure-retaining SSCs is governed by CSA N285.0 (For nuclear Class SSCs) and CSA B51 (For Class 6 SSCs). CSA N289.3 has certain specific rules for pressure boundary requirements in Canadian practice that modify some ASME requirements.

The seismic design of structural systems should be categorized according to seismic design category (SDC) 1 to SDC 5 as per ASCE 43 [105]. All structures important to safety are classified as SDC 5. However, the designer may still classify some structures as SDC 3, 4 and 5 provided that they include proper justification. Guidance on important to safety structural systems is provided as follows:

- for concrete containment, the design should be based on ASCE 43-05 (SDC 5, limit state D) and CSA N287.3.
- for steel containment, the design should be based on ASCE 43-05 (SDC 5); ASME BPVC, Section III, Division 1, Subsection NE: Class MC Components; and NRC RG 1.57 [106], “Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components.”
- for concrete and steel safety-related structures, the design should be based on ASCE 43-05 (SDC 5, limit state D) and CSA N291, “Requirements for Safety-Related Structures for CANDU Nuclear Power Plants.”

For all safety design categories in a reactor facility, ductility requirements should provide margins for the BDBE.

Seismic Qualification

The design basis earthquake is used for the SQ of certain SSCs that are categorized as either seismic Category A or Category B. The SQ as per CSA N289.1 is a process to verify the ability of the SSC to maintain its design-intended performance during and/or following the DBE. CSA N289.1 provides SQ and evaluation requirements and guidance for both quantitative methods and qualitative methods. For new reactors, REGDOC-2.5.2 specifies the use of quantitative methods of SQ, which consist of qualification by test, analysis, or combination of the two. The results are assessed in a deterministic manner. The following paragraphs provide a summary of SQ methods along with an overview of the applicable CSA standard for each SQ method:

a) SQ by analysis - Quantitative method

The CSA N289.3 specifies the requirements, criteria, and methods of analysis for: a) determining the design response spectra and ground motion time-histories to be used in the analysis; b) establishing design criteria for SSCs and supports that require the SQ; and c) performing seismic analyses, including the effects of the soil-structure-interaction. In particular, the SQ requirements for nuclear Class 1 pressure-retaining systems and components are adopted from ASME BPVC, Section III, for level C stress limit.

b) SQ by test - Quantitative method

The CSA N289.4 defines the processes and requirements for performing the SQ by testing and presents the test methods that may be used for the SQ of the nuclear power plant SSCs. Seismic qualification by testing is typically used for SSCs that will be performing both an active function and that are required to change state during or following a seismic event in order to perform a safety function while maintaining structural and/or pressure boundary integrity.

c) SQ by combination of analysis and test - Quantitative method

The SQ of a pressure relief valve is a typical example of the SQ by the combination of the analysis and the test: It is almost impossible to fully demonstrate the functionality of a given piece of active mechanical equipment by solely analyzing it because of the presence of moving parts. Hence, the active mechanical component important to safety, such as a pressure relief valve, shall be qualified to ensure its functionality during and/or following the DBE as per CSA N289.4 and to confirm its structural integrity against the design basis earthquake as per CSA N289.3 and ASME BPVC, Section III.

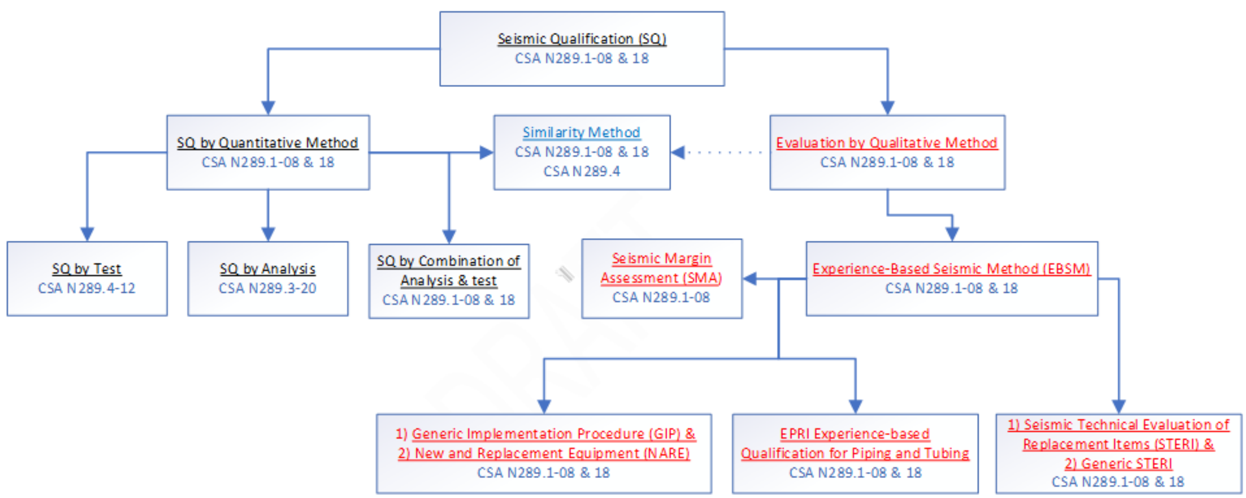

Figure 18: Seismic Qualification and Evaluation Methods in CSA N289 Series

Seismic Evaluation for Beyond-Design-Basis Seismic Event

CSA N289.1 mandates that each facility shall have a periodic evaluation to demonstrate readiness for a beyond-design-basis seismic event. This requirement's purposes are to: 1) ensure that safety-related SSCs in success paths in a given NPP have seismic capacity; and 2) identify the weak links in success path SSCs in a given NPP against the beyond-design-basis seismic event that could jeopardize the design concepts of redundancy and DID through common-cause failures. The evaluation can consist of the following two parts: a) evaluation of the ability to withstand seismic events beyond the design basis earthquake and perform the safety functions for safe shutdown, fuel cooling, the containment of potential releases of radioactive material, and the monitoring and control of essential safety-related functions in the event of an earthquake; and b) evaluation of the ability to cope with the potential consequences of a beyond-design-basis seismic event. This is addressed in areas outside the scope of this standard (e.g., safety analysis, severe accident management). The two approaches available for seismic risk analysis are seismic PSA and seismic margin assessment (SMA).

The purpose of seismic PSA is to estimate the probability of occurrence of various levels of earthquakes, including the exceedance of the design basis earthquake used in the NPP design and to estimate the plant and SSCs responses resulting from the estimated earthquakes by calculating a Seismic Core Damage Frequency and a Large Release Frequency (LRF) to quantify risk. In some cases, the PSA-based SMA approach has also been used in Canadian plants. The SMA will be described in the next section that will discuss a plant level seismic capacity against the beyond-design-basis seismic event.

The SMA describes the additional seismic margin plants have, by virtue of their conservative design, to withstand earthquakes larger than the design basis earthquake and to perform their safety function. This margin can be defined in terms of the HCLPF capacity of each critical SSC or the overall HCLPF of the plant. The HCLPF is the selection of seismic analysis parameters at statistically consistent levels of probability of failure or exceedance such that the seismic capacity of a structure, system, or component has a 95% confidence of a 5% probability of failure. The components having the lowest HCLPF capacity and part of critical risk contributing cut sets (in the case of seismic PSA) are the weak links, and determine the plant level HCLPF capacity. Modifications to plant design, procurement, operations, and maintenance procedures can be necessary to ensure that seismic evaluation of success path SSCs is maintained for the life of the facility.

8.1.3 NRC Regulations, Guidance, and Standards

Regulatory Framework

For CP applications submitted on or after January 10, 1997, 10 CFR 50.34(a)(12) requires that stationary power reactor applicants comply with the earthquake engineering criteria of Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants." to 10 CFR Part 50. In addition, all applicants for design certifications, combined licenses, standard design approvals, and manufacturing licenses pursuant to 10 CFR Part 52 must comply with the earthquake engineering criteria in Appendix S to 10 CFR Part 50. Appendix S to 10 CFR Part 50 defines that SSCs required to withstand the effects of the safe shutdown earthquake (SSE) ground motion or surface deformation are those SSCs necessary 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 that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1).

Appendix S to 10 CFR Part 50 implements GDC 2, "Design bases for protection against natural phenomena," which requires that SSCs important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety functions. The SSE ground motion, as specified in Appendix S to 10 CFR Part 50, must be characterized by free-field ground motion response spectra at the free ground surface and the horizontal component of that ground motion must have a peak ground acceleration of at least 0.1 g at the foundation level.

Appendix S to 10 CFR Part 50 states that the nuclear power plant must be designed such that certain SSCs will remain functional and within applicable stress, strain and deformation limits if the SSE occurs. The required safety functions of SSCs must be assured during and after the SSE vibratory ground motion through design, testing, or qualification methods. The potential for surface deformation at any location and in any direction must be taken into account unless evidence indicates that the assumption is not appropriate.

The NRC has developed guidance applicable to LWRs that provides acceptable methods to comply with the above listed requirements in Appendix S to 10 CFR Part 50 regarding defining the ground motion response and analysis of the structural response. The following SRP Sections are applicable to modeling the seismic motion of civil structures, substructures, systems, and would generally apply to non-LWR structures as well:

- SRP Section 3.7.1, "Seismic Design Parameters"
- SRP Section 3.7.2, "Seismic System Analysis"
- SRP Section 3.7.3, "Seismic Subsystem Analysis"

RG 1.29 [107], "Seismic Design Classification for Nuclear Plants," describes a method that NRC staff considers acceptable for use in identifying and classifying those features of LWRs that must be designed to withstand the effects of the SSE and designates these plant features as seismic Category I. The features designated in RG 1.29 as seismic Category I include:

- SSCs generally satisfying the definition of safety-related SSCs in 10 CFR 50.2;
- SSCs that are neither safety-related nor part of the radioactive waste systems whose postulated failure could result in offsite dose consequences greater than 0.005 sievert (0.5 rem) to the whole body or its equivalent; or
- Post-accident monitoring instrumentation.

The applicants for LWRs whose safety-related classification definition is consistent with 10 CFR 50.2 may directly apply RG 1.29 to facilitate identification and classification of seismic Category I SSCs. Advanced reactor applicants using the traditional approach should ensure that safety-related SSCs, SSCs containing significant radioactive material inventories, and SSC s necessary for post-accident monitoring are appropriately identified for qualification as seismic Category I SSCs. For both SMRs and advanced reactors, RG 1.29 provides guidance to identify those SSCs whose continued function is not required but whose failure could reduce the functioning of seismic Category I SSCs to an unacceptable safety level or could result in incapacitating injury to occupants of the control room.

For applicants that use the LMP approach, Section 3.4.9 of this report describes the process for using safety classification of SSCs to assign engineering design rules and specifications. *Table 6* identifies the applicability of engineering design rules for items classified as SR and NSRST. The application of engineering design rules for LMP is consistent with the traditional approach in that full seismic qualification is expected for those SSCs classified as SR under a risk-informed technology-inclusive method. Advanced reactor SSCs classified as NSRST or NST should be evaluated on a case-by-case basis to ensure that the plant design protects required safety functions from interference due to failure of NSRST or NST SSCs during and following an SSE.

Qualification of SSCs that must retain their function during and following an SSE is a more technology-neutral process. Qualification of pressure boundary components is demonstrated by analysis of specified loading combinations in Section III of the ASME BPVC, as discussed in Section 7.1.3 of this report. Similarly, qualification of structures is demonstrated by analysis of structures using the appropriate standard, as described in Section 7.4.3 of this report. Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment," of the SRP provides guidance for the staff review of seismic qualification of seismic Category I SSCs. This guidance specifies that seismic qualification be performed through testing, analysis, or a combination of testing and analysis, and references RG 1.100 [108], "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," for seismic qualification of active mechanical and electrical equipment. RG 1.100 endorses the following standards for this purpose:

- IEEE Std 344-2013 [109], "IEEE Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations."
- IEEE Std C37.98-2013 [110}, "IEEE Standard for Seismic Qualification Testing of Protective Relays and Auxiliaries for Nuclear Facilities."
- ASME Qualification of Mechanical Equipment (QME)-1-2017 [111], "Qualification of Active Mechanical Equipment Used in Nuclear Facilities."

RG 1.29 states that the seismic analysis of seismic Category I SSCs must extend into connected systems to the extent necessary to maintain the validity of the analysis and the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of seismic Category I SSCs.

Further, RG 1.29 specifies that those portions of SSCs whose continued function during or following an SSE is not required, but whose failure could reduce the functioning of any seismic Category I SSC to an unacceptable safety level or could result in incapacitating injury to occupants of the control room, should be designed and constructed so that the SSE would not cause such failure. This second group of SSCs is commonly designated seismic Category II. Finally, RG 1.29 references several guidance documents that address seismic qualification of specific SSCs. These guidance documents address radioactive waste processing systems, instrument sensing lines, and fire protection equipment.

RG 1.143 [112], "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," provides design guidance in regard to natural phenomena hazards, internal and external man-induced hazards, and quality group classification and quality assurance provisions for radioactive waste management systems, structures, and components. This guidance provides a design basis ground motion response spectrum of one-half the SSE spectrum for qualification of higher-hazard radioactive waste SSCs whose postulated failure could result in offsite dose consequences greater than 0.005 sievert (0.5 rem) to the whole body or its equivalent. Other radioactive waste SSCs are designed with some margin for seismic activity but are not qualified to a specific ground motion response spectrum.

RG 1.151 [113], "Instrument Sensing Lines," provides guidance for acceptable design of safety-related instrument sensing lines (up to 1 inch [25.4 mm] outside diameter lines or three-quarter nominal pipe [1.060 inch (26.67 mm)]) outside diameter, starting at the root valve/piping class change up to but not including the manufacturer-supplied instrument connection. RG 1.151 endorses the design standards, including seismic design, specified in ANSI/ISA-67.02.01, “Nuclear Safety-Related Instrument-Sensing Line Piping and Tubing Standard for Use in Nuclear Power Plants”.

RG 1.189 [114], “Fire Protection for Nuclear Power Plants”, provides guidance used to establish the design requirements for portions of fire protection SSCs to meet the requirements of GDC 2, as they relate to designing those SSCs to withstand the effects of the SSE. This guidance specifies that the following fire protection SSCs be seismically qualified:

- fire pump(s) providing at least 100 percent of design capacity;
- a piping system supplying water to at least two standpipes and hose connections in areas containing seismic Category I SSCs;
- piping connecting a separate seismic Category I source of water to the qualified hose standpipe system;
- fire detection and alarm systems at high seismic-hazard sites; and
- consider seismic design of fire suppression systems at high seismic hazard sites.

SRP Section 19.0, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors,” provides guidance for reviewing PRA-based SMAs submitted in support of a DC or COL application. Regulations (10 CFR 52.47(a)(27), 10 CFR 52.79(a)(46), 10 CFR 52.79(d)(1), and 10 CFR 50.71(h))[18] require that DC applicants and COL applicants and holders develop, update, and maintain a PRA that supports, in part, evaluation of seismic margins in terms of HCLPF. DC/COL-ISG-20, “Interim Staff Guidance on Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors,” discusses post-DC activities to update the PRA-based SMA throughout the licensing process of new reactors, including COL action items and post-licensing activities, to ensure a coherent and consistent process for the quality of PRA-based SMA. These activities support identification of PRA-based insights that include design, site, and operational attributes.

In support of the proposed 10 CFR Part 53 regulation, the NRC has begun the process of developing guidance on risk-informed options for seismic design. The NRC staff has circulated a pre-decisional draft document [115] entitled, “Technology-Inclusive, Risk-Informed, and Performance-Based Methodology for Seismic Design of Commercial Nuclear Plants,” to obtain stakeholder feedback on the document. As described in the draft document, the first seismic design option for 10 CFR Part 53 applicants would rely on existing guidance described above. The second option would use the safety classification developed through the LMP for individual SSCs to assign initial seismic design categories (SDCs) per ASCE 43 and associated limit states(LSs). These are iteratively developed considering seismic PRA information, including consideration of sequence-dependent design-basis ground motions and SSC fragilities, to finalize SDCs and LSs for the design. A third option is similar to the second option but describes alternative means of identifying initial SDCs and LSs for evaluation through a seismic PRA.

[18] 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 [RIN 3150 AI66; Regulations.gov Docket: NRC-2009-0196]. The NRC proposed adding 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. The proposed changes would also modify 10 CFR 50.71(h) to require OL license holders to update and maintain the PRA, which supports development of a seismic margin assessment.

8.1.4 Similarities and Differences

The approach to SSC seismic classification and designation of engineering design rules and specifications is similar among the CNSC approach and the NRC Traditional and LMP approaches. The design criteria for all approaches address a design basis ground motion that is used for qualification of SSCs, and both the CNSC and the NRC consider beyond-design-basis conditions through evaluation of seismic margins. For all approaches, qualification is based on analysis, testing, or a combination of analysis and testing.

The scope and classification of SSCs subject to seismic qualification varies between the CNSC and the NRC approaches. The CNSC specifies seismic qualification of a broad spectrum of SSCs including nearly all SSCs classified as important to safety, and the seismic qualification classification differentiates between SSCs performing an active function and those SSCs performing passive functions. The NRC regulations require seismic qualification of a narrower set of SSCs limited to those classified as safety-related and those nonsafety-related SSCs whose failure could affect the capability of other SSCs to perform their safety-related functions. NRC guidance adds seismic qualification of certain SSCs that could initiate an accident condition if failed, SSCs providing specific fire protection capabilities, SSCs containing significant gaseous or liquid radioactive waste inventories, and certain post-accident monitoring capabilities.

The approach to seismic qualification of components for both the CNSC and the NRC is similar and applies some of the same standards. For both the NRC and the CNSC regulatory frameworks, the classification and application of engineering design rules are driven by the functional importance to safety. The similarities in engineering design rule application include: (1) separate seismic qualification standards for SSCs that must retain active functional capability and a second qualification for SSCs that must maintain structural / pressure boundary integrity; (2) qualification based on analysis, testing, or a combination of the two methods; (3) qualification of pressure boundary components that retain functionality to the ASME BPVC; (4) qualification of structures, although referencing different standards for structures other than metal containments; and (5) assessment of seismic margin to confirm capabilities for BDBEs.

The identified standards provide particular engineering design rules for design and/or qualification of SSCs. Standards for structures and pressure-retaining systems and components provide detailed design rules, loading combinations, and analysis methods. Similarly, qualification standards provide acceptable techniques to demonstrate the seismic ruggedness of active SSCs. The endorsed standards related to metal containment structures and electrical equipment seismic qualification, but, even where endorsed standards differ between regulatory bodies, the underlying purpose and methodologies of the standards are often similar. While differences in endorsed standards exist for some purposes, both regulatory bodies are familiar with and reference standards developed and issued by the ASME, ASCE, IEEE, and ACI. Therefore, common approaches that rely on these standards may be developed, with appropriate justification for standards differing from guidance, for applications to both regulatory bodies.

8.1.5 Assessment of Impact

Overall, the Canadian and the U.S. approaches to seismic classification and design rules for seismic qualification share many commonalities, especially with regard to containment seismic design and qualification of electrical or electronic components. There are some differences in the scope of SSCs subject to seismic qualification, particularly with respect to SSCs that provide DID functions. These differences in scope may be mitigated by a risk-informed evaluation of seismic risk and consideration of the generally lower seismic hazard for Canadian nuclear power plant sites. In both regulatory bodies, structures performing similar safety-significant functions have been identified for rigorous review and qualification.

Significant emphasis is placed by both countries on evaluation of methods of analysis for civil structures that is comparable. Both the CNSC and the NRC are familiar with and have endorsed ASME BPVC standards for design of containment, and application materials based on these standards is likely to be useable with both regulatory bodies. For other civil structures, both regulators accept unendorsed standards with appropriate justification. Therefore, common application information developed to a specified set of standards is likely to be useable in applications to both regulatory bodies, with appropriate justification for the selected standard based on the safety significance of the associated civil structures and the seismic hazard at the intended site.

8.2 Fire Protection Design Rules

8.2.1 Overview

The objectives of the nuclear power plant fire protection program are to minimize both the probability of occurrence and the consequences of a fire. The fire protection program is designed to provide reasonable assurance through DID that any fire that occurs does not prevent the performance of necessary safe shutdown functions and that any radioactive release to the environment in the event of a fire will be minimized.

8.2.2 CNSC Regulations, Guidance, and Standards

Section 5.12, "Fire safety," of CNSC REGDOC-2.5.2 specifies that suitable incorporation of operational procedures, redundant SSCs, physical barriers, spatial separation, fire protection systems, and design for fail-safe operation achieves the following general objectives:

- prevent the initiation of fires;
- limit the propagation and effects of fires that do occur by quickly detecting and suppressing fires to limit damage and confining the spread of fires and fire by-products that have not been extinguished;
- prevent loss of redundancy in safety and safety support systems;
- provide assurance of safe shutdown;
- ensure that monitoring of safety-critical parameters remains available,
- prevent exposure, uncontrolled release, or unacceptable dispersion of hazardous substances, nuclear material, or radioactive material, due to fires;
- prevent the detrimental effects of event mitigation efforts, both inside and outside of containment; and
- ensure structural sufficiency and stability in the event of fire.

The CNSC's regulatory model in fire protection is based upon the implementation of the DID concept described in REGDOC-2.5.2 to ensure the protection of the health and safety of persons and the environment. From a fire protection perspective, DID is achieved through a combination of design (e.g., physical barriers, spatial separation, fire protection detection and suppression systems), management of fire protection (e.g., operational procedures), quality assurance, and emergency arrangements. The principal of DID applies to fire protection at all levels of the facility and its associated activities, from establishing high-level facility objectives to defining the detailed procedures and equipment required to meet those objectives.

Fire Protection Standards

The CSA standards are prescriptive in nature but allow for the use of alternative or performance-based solutions to meet prescriptive requirements. The required level of DID is achieved by the implementation of CSA N293 [116], “Fire protection for nuclear power plants.” The CSA N293 standard specifies that:

- Licensees implement and maintain a comprehensive fire protection program (FPP) in order to reduce the occurrence of fires and limit their consequences and severity (the required elements of the program are prescribed in the CSA standards). The FPP is defined as a set of planned, coordinated, and controlled activities which is documented and integrated into the operation of the facility; and
- Complete fire safety assessments include Fire hazard assessment (FHA), Fire safe shutdown analysis (FSSA), and Code Compliance review. The FHA and FSSA are deterministic analyses.

The CSA standard requires the implementation of the NBCC, and the National Fire Code of Canada (NFCC) [117]. The NBCC and the NFCC are objective-based Codes and contain prescriptive requirements but state the objectives and functional statements for each prescriptive requirement. The standard supports alternative solutions to meet these requirements. In addition, the CSA standard references many National Fire Protection Association (NFPA) standards addressing areas including fire detection and alarm systems, design and installation of manual and automatic fire suppression systems and components, firefighter training and operations, maintenance and testing of firefighting systems, and overall FPPs.

To achieve the noted objectives, REGDOC-2.5.2 specifies that new NPPs comply with CSA N293, the NBCC, and the NFCC. Both the CSA standard and the REGDOC-recommend the following as guidance documents to address fire safety aspects associated with operator actions, fire safe shutdown circuit analysis, and spurious operations:

- NRC NUREG-1852 [118], “Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire,” 2007; and
- NEI 00-01 [119], Guidance for Post-Fire Safe Shutdown Circuit Analysis, 2005.

Conformance with the fire protection program, which implements the REGDOC requirements and the CSA standard, is established by incorporation in the License Condition Handbook issued with the facility operating license.

Probabilistic Fire Evaluation

A Level 2 PSA is expected to address external and internal fires as initiating events through the license requirement for the preparation of a PSA in accordance with REGDOC-2.4.2, “Probabilistic Safety Assessment (PSA) for Nuclear Power Plants,” but this activity is not currently required to be incorporated into to the FPP. REGDOC-2.4.2 specifies that applicants seek CNSC acceptance of the methodology and computer codes to be used for the PSA before using them. The methodology typically used for the completion of Fire PSAs is NUREG/CR-6850 [120], “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities: Detailed Methodology, Final Report.”

8.2.3 NRC Regulations, Guidance, and Standards

Nuclear power plant fire protection regulations in the U.S. provide both deterministic (prescriptive) and risk-informed, performance-based approaches. For new reactors 10 CFR Part 50.48, “Fire protection,” and GDC 3, “Fire protection,” of Appendix A to 10 CFR Part 50 apply. The draft fire protection rule in 10 CFR Part 53, “Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors,” would establish requirements similar to 10 CFR 50.48 for an FPP supporting operations.

In accordance with 10 CFR 50.48, holders of a power reactor operating license under 10 CFR Part 50 or a combined license under 10 CFR Part 52 must have a fire protection plan that satisfies GDC 3. The fire protection plan must describe the overall fire protection plan for the facility, outline the programs for fire protection, automatic fire detection and suppression capability, and limitations of fire damage. The fire protection plan must also describe specific features necessary to implement the program, such as administrative controls (e.g., policies and procedures) and personnel requirements for fire prevention and manual fire suppression activities, and the means to limit fire damage to SSCs important to safety so that the capability to safely shutdown the plant is ensured. RG 1.189, “Fire Protection for Nuclear Power Plants,” describes an approach acceptable to the NRC staff to meet the requirements of 10 CFR 50.48(a).

In 10 CFR 50.48(c), the NRC approved incorporation by reference of NFPA Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition." Although specific to operating LWRs, the standard provides a performance-based approach rather than a strictly deterministic approach. In 10 CFR 50.48(c)(4), the NRC provided a license amendment process to submit alternatives to compliance with NFPA-805 provided that the alternatives:

- satisfy the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;
- maintain safety margins; and
- maintain fire protection DID (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

Each applicant for a standard design approval, design certification, or manufacturing license under 10 CFR Part 52, must have a description and analysis of the fire protection design features for the portions of the standard plant covered by the application necessary to demonstrate compliance with GDC 3.

Beyond the program elements described in 10 CFR 50.48, GDC 3 requires that SSCs important to safety be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and firefighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on SSCs important to safety. Nuclear power plant SSCs important to safety are those required to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. GDC 3 also requires that firefighting systems be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs.

The fire protection programs for new reactors that submit applications in accordance with 10 CFR Part 52 are subject to 10 CFR 50.48(a). The NRC staff uses the guidance and acceptance criteria in NUREG-0800, Chapter 9, SRP Section 9.5.1.1 to review Part 52 fire protection programs.

Defense-in-depth Consideration

Fire protection for nuclear power plants uses the concept of DID to achieve the required degree of reactor safety. This concept integrates administrative controls, fire protection systems, and safe shutdown capability to achieve the following objectives:

- Preventing fires from starting.
- Rapidly detecting, controlling, and extinguishing those fires that do occur, thereby limiting fire damage.
- Providing an adequate level of fire protection for SSCs important to safety, so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant.

The DID approach uses the design and operation of nuclear power plants in a manner that prevents and mitigates accidents that release radiation or hazardous materials. The key is to create 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 relied upon exclusively. The DID approach includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures.

Codes and Standards

Consensus codes and standards have been integral to the NRC fire protection regulatory process. Codes and standards promote safe operation of nuclear power plants. The NRC establishes fire protection regulations that have to be satisfied to receive and maintain an operating nuclear power plant license. Methods of satisfying NRC regulations are explained in NRC regulatory guides, while standard review plans explain how NRC staff reviews applications for licenses. Both widely reference codes and standards. Guidance in RG 1.189 specifies that, for those fire protection SSCs installed to satisfy the NRC requirements and designed to NFPA codes and standards, the code or standard of record should be those NFPA codes and standards in effect 180 days before the submittal of the applicable OL, SDA, DC, ML, or COL application under 10 CFR Part 50 or 10 CFR Part 52.

The FPP for new LWR designs, including SMRs, should comply with the provisions specified in NFPA 804, “Standard for Fire Protection for Advanced Light-Water Reactor Electric Generating Plants,” as they relate to the protection of post-fire safe shutdown capability and the mitigation of a radiological release resulting from a fire. However, the NRC has not formally endorsed NFPA 804, and some of the guidance in the NFPA standard may conflict with regulatory requirements. When conflicts occur, the applicable regulatory requirements and guidance in RG 1.189 apply.

General Fire Protection Guidelines and Regulatory Acceptance Criteria

The following guidance documents outline the acceptance criteria for the NRC review of SMR and advanced reactor FPPs:

- RG 1.189 provides fire protection guidance that identifies the scope and depth of fire protection that the NRC staff would consider as one acceptable way for nuclear power plants to meet fire protection regulations. The regulatory guide provides guidance to licensees on the proper content and quality of engineering equivalency evaluations used to support the FPP. The NRC staff developed the regulatory guide to provide a comprehensive fire protection guidance document and to identify the scope and depth of fire protection that the NRC would consider acceptable for nuclear power plants.
- RG 1.205 [121],"Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," provides guidance for use in complying with the requirements that the NRC has promulgated for risk-informed/performance-based fire protection programs that comply with 10 CFR 50.48 and the referenced 2001 Edition of the NFPA 805 standard. It endorses portions of Nuclear Energy Institute (NEI) 04-02, where it has been found to provide methods acceptable to the NRC for implementing NFPA 805 and complying with 10 CFR 50.48(c). This regulatory guide also indicates that Chapter 3 of NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," when used in conjunction with NFPA 805 and the regulatory guide, provides one acceptable approach to circuit analysis for a plant implementing a FPP under 10 CFR 50.48(c).
- SRP Section 9.5.1.1 (ADAMS Accession No. ML090510170), "Fire Protection Program," February 2009, focuses on deterministic fire protection plans.
- SRP Section 9.5.1.2 (ADAMS Accession No. ML092590527), Revision 0, "Risk-Informed, Performance-Based Fire Protection Program," December 2009, provides guidance on risk-informed, performance-based fire protection plan licensing actions submitted pursuant to 10 CFR 50.48(c).
- DANU-ISG-2022-09 [122], "Advanced Reactor Content of Application Project – 'Risk-Informed, Performance-Based Fire Protection Program (for Operations)'," addresses development of an advanced reactor application for an OL under 10 CFR Part 50 or COL under 10 CFR Part 52 associated with a proposed risk-informed, performance-based fire protection program for operations. Although 10 CFR 50.48(c) is not applicable to non-LWRs, elements and concepts in NFPA 805; the guidance in RG 1.205; and the guidance RG 1.189 can be applied to advanced reactor applications, with justified exceptions or deviations.

The FPP is established as an operating license requirement through a license condition. The SSCs incorporated into the facility design to support the FPP are classified as important to safety rather than safety-related SSCs. Engineering design rules and specifications applied to these SSCs include the specific quality assurance and management provisions incorporated in the FPP, as well as design rules and specifications included in referenced national consensus standards. As described in Section 8.1.3 of this report, at least two standpipes and hose connections for manual firefighting are required in areas containing equipment required for a safe plant shutdown in the event of a safe shutdown earthquake, and, in areas of high seismic activity, fire detection, alarm systems, and fire suppression systems are designed to function following a safe shutdown earthquake.

8.2.4 Similarities and Differences

The regulatory model REGDOC-2.5.2, provides the source of fire protection regulations in Canada, while the core of commercial nuclear power plant fire protection regulations in the U.S. comes from 10 CFR 50.48 and Appendix A to 10 CFR Part 50. While the objectives of these documents are very similar, the regulatory design and specifications contain some differences.

Similarities

- The primary objectives of the FPP for nuclear power plants for both the CNSC and the NRC are to minimize the probability of occurrence and consequences of a fire. Meeting these objectives provides reasonable assurance, through DID, that a fire will not prevent the necessary safe shutdown functions from being performed, and that radioactive releases to the environment in the event of a fire will be minimized.
- The concept of DID for both the CNSC and the NRC is to protect the health and safety of the public from fires at nuclear power plants, with the following objectives:
  - To prevent fires from starting;
  - To detect rapidly, control, and extinguish promptly those fires that do occur;
  - To protect the reactor so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant.
- Effective fire suppression is a primary objective of nuclear power plant safety. Both the CNSC and the NRC use a more deterministic approach to suppression requirements and focus more on the protection of the redundant safe shutdown trains. Redundant safe shutdown important to safety trains are required to be separated by means of rated structural fire barriers or protected with fire-resistive barrier assemblies. Fire detection systems and fire suppression systems should be installed as determined by the fire hazards analysis to protect redundant systems or components necessary for a safe shutdown and SSCs important to safety.
- Both the CNSC and the NRC utilize conditions on the operating license to establish expectations for licensees to achieve DID and implement industry codes and standards incorporated into the FPP.
- Both the CNSC and the NRC fire protection requirements allow use of performance-based methods using fire probabilistic risk assessment (PRA) results to implement the FPP.
- The CNSC and the NRC reference many of the same consensus standards for the design and installation fire protection systems and components as well as fire protection guidance documents (NUREG reports) for fire hazard analysis and fire safe shutdown analysis.

Differences

- Nuclear power plant fire protection regulations in the United States are both deterministic/prescriptive and risk-informed/performance-based. Nuclear power plant applicants have the option of using either type of fire protection regulations. The CNSC fire protection regulations are deterministic/prescriptive but allow the use of alternative or performance-based methods to meet deterministic/prescriptive requirements. Through its endorsement of NFPA-805, the NRC has established precedent for more risk-informed and performance-based FPP implementation.

8.2.5 Assessment of Impact

The CNSC and the NRC expect that their fire protection requirements will result in similar outcomes. Although the NRC approach explicitly provides for the use of risk-informed FPPs for LWRs, both the NRC and the CNSC provide the flexibility to adopt these principles for FPPs developed for all reactor types. With regard to SSC classification and application of engineering design rules and specifications, the fire protection expectations of the CNSC and the NRC are very similar. The structural fire protection and fire suppression systems are considered important to safety using a graded approach under both the CNSC and the NRC approaches. Both regulatory bodies rely on similar deterministic approaches to establish the design basis for structural fire protection and fire suppression SSCs and reference common standards establishing the design rules and specifications for structural fire protection components and fire suppression systems.

8.3 Environmental Qualification and Hazard Barriers

8.3.1 Overview

This section covers code classification and assignment of engineering design rules for the environmental qualification (EQ) of equipment. EQ is the demonstration that equipment important to safety can perform its safety function(s) during and after postulated DBEs without experiencing common-cause failures and while withstanding anticipated harsh environments.

8.3.2 CNSC Regulations, Guidance, and Standards

Regulatory Framework

In Canada, the environmental qualification of equipment is a licensing condition for every nuclear facility. In addition, CSA N290.13 [123], "Environmental qualification of equipment for nuclear power plants," is also a compliance verification criterion. Consequently, the CNSC relies on the CSA standard as well as licensing basis documents and other regulatory documents to enforce both requirements and guidance.

REGDOC-1.1.2 sets out requirements and guidance on submitting an application 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 equipment EQ in order to meet the expectations of Section 5.8, "Equipment environmental qualification," of REGDOC-2.5.2. 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.

As REGDOC-2.5.2 guidance states, a systematic approach should be followed to identify the EQ of equipment in order to ensure that SSCs necessary to fulfill the safety functions have an appropriate safety classification and reliability. The EQ of equipment ensures that the following functions can be carried out:

- The reactor can be safely shut down and kept in a safe shutdown state during and following AOOs and DBAs.
- Residual heat can be removed from the reactor after shutdown, and also during and following AOOs and DBAs.
- The potential for release of radioactive material from the plant can be limited, and the resulting dose to the public from AOOs and DBAs can be kept within the dose acceptance criteria.
- Post-accident conditions can be monitored to indicate whether the above functions are being carried out.

The design bases, design criteria, regulatory documents, standards, and other documents that will be used to establish the EQ of equipment should be specified.
The EQ elements should include:

- The design should identify:
    - systems and equipment required to perform safety functions in a harsh environment, including their safety functions and applicable DBAs;
    - nonsafety-related equipment whose failure due to a harsh post-accident environment could prevent safety-related equipment from accomplishing its safety function; and
    - accident monitoring equipment.
- The design should provide a distinction between mild and harsh environments and a listing of bounding DBAs and associated environmental conditions considered harsh.
- The design should identify environmental condition profiles for the bounding DBAs over the specified equipment mission time.

Safety Classification and Qualification

The safety classification of EQ equipment should consider the following:

- equipment specification which identifies the equipment's safety function(s)
- systems and equipment required to perform safety functions in a harsh environment, including their safety functions and applicable DBAs
- nonsafety-related equipment whose failure due to harsh post-accident environment could prevent safety-related equipment from accomplishing its safety function
- accident monitoring equipment
- typical equipment mission time during DBAs

The safety classification of equipment credited under DECs is dependent on the importance of the equipment (type and location used to perform necessary functions) to achieve a safe shutdown state. The goal is to attain a reasonable assurance that the equipment will survive to perform its safety function in the accident timeframes.

For cases where protective barriers are included in the design, the barriers themselves should be addressed in a qualification program and ultimately given an appropriate safety classification. Protective barriers are used to isolate equipment from possible harsh environmental conditions (e.g., steam-protected rooms, enclosures, and water-protected areas).

For harsh environment qualification, the following elements should be considered:

- For equipment and components located in a DBA harsh environment, type tests are the preferred method of qualification (particularly for electrical equipment); where type tests are not feasible, justification by analysis or operating experience (or a combination of both) may be used.
- Equipment should be reviewed in terms of design, function, materials and environment, to identify significant aging mechanisms caused by operational and environmental conditions occurring during normal operation. Where a significant aging mechanism is identified that aging should be taken into account in the equipment qualification.
- The qualification should systematically address the sequence of age conditioning, including sequential, simultaneous, synergistic effects, and the method for accelerating radiation degradation effects.
- Appropriate margins, as given in EQ-related standards, should be applied to the specified environmental conditions.

- For certain equipment (e.g., digital I&C equipment, and new advanced analog electronics), additional environmental conditions – such as electromagnetic interference, radio frequency interference, and power surges – should be addressed.

Applicable Codes and Standards

Section 5.8 of REGDOC-2.5.2 references the following codes and standards for guidance on EQ of equipment:

- ASME QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants.”
- CSA, N290.13, “Environmental qualification of equipment for CANDU nuclear power plants.”
- ERPI, Technical Report 1021067 [124], Nuclear Power Plant Equipment Qualification Reference Manual.
- IAEA, Safety Reports Series No. 3 [125], “Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving and Reviewing.”
- IEC/IEEE, Standard 60780-323 [126], Nuclear facilities – Electrical equipment important to safety – Qualification.”
- IEEE, Standard 627 [127], “Qualification of Equipment Used in Nuclear Facilities.”

8.3.3 NRC Regulations, Guidance, and Standards

The mechanical, electrical, and I&C, including digital I&C equipment designated as important to safety, is addressed in the EQ program to verify it is capable of performing its design functions under all normal environmental conditions, AOOs, and accident and post-accident environmental conditions. EQ is the design verification process by which important to safety equipment is demonstrated to remain capable of performing its design functions during and after exposure to a design basis accident in a harsh environment.

The objective of the EQ program is to confirm that the set of equipment to be environmentally qualified includes safety-related equipment, nonsafety-related equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of specified safety functions, and instrumentation to monitor parameters specified in RG 1.97. The applicable regulations are: 10 CFR 50.49; 10 CFR 50.55a(h); 10 CFR 50.69; 10 CFR Part 50, Appendix A, GDCs 1, 2, 4, and 23; and 10 CFR Part 50, Appendix B. The scope of equipment that is included in the EQ program is:

- Safety-related electric equipment:
  - (i) This equipment is that relied upon to remain functional during and following design basis events to ensure—
    - (A) The integrity of the reactor coolant pressure boundary;
    - (B) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
    - (C) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures.
  - (ii) Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed.

- Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions by the safety-related equipment.
- Certain post-accident monitoring equipment.

There are numerous guidance documents on EQ. RG 1.89 [128], “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” provides the overall methodology and other RGs address qualification of specific types of equipment (i.e., actuators, connectors, cables, etc). This RG endorses, with clarifications, IEC/IEEE Standard Std. 60780-323, “Nuclear Facilities—Electrical Equipment Important to Safety—Qualification,” Edition 1, 2016-02.

Combustible gas control and equipment survivability is addressed in 10 CFR 50.44. Equipment survivability consists of evaluating best estimate containment conditions against tests and analysis by relying on equipment qualification results for actual plant equipment, relying on results from research or experiments performed on similar equipment, or relying on a thermal lag analysis to account for the real time for the temperature of the critical component to rise to the actual accident conditions. The equipment needed to mitigate a severe accident should be identified, and the containment atmospheric assessments of temperature, pressure, and radiation should be completed. Further, equipment should be evaluated for containment flooding.

10 CFR 50.155, “Mitigation of beyond-design-basis events,” considers equipment location to maximize survivability following a design basis accident.

<u>Hazards Barriers</u>

The EQ program provides reasonable assurance that important to safety equipment will perform its required safety functions under a harsh environment. Furthermore, protection from a harsh environment, effects of piping failures such as jet impingement, and other internal events, is accomplished via the following design considerations: (1) separation of high and moderate energy systems from essential systems[19] and components, or (2) barriers, deflectors, shields, guard pipes or enclosures designed to withstand the effects of postulated piping failures and other internal events. Where protection by separation or functional barriers is impractical (e.g., interconnections between fluid systems and essential systems and components), credit may be taken for redundant design features. Barriers designed to protect essential equipment from the effects of postulated piping failures have the same classification as the essential equipment and are designed in accordance with the applicable design criteria.

8.3.4 Similarities and Differences

- The CNSC and the NRC have slightly different definitions of harsh environment. Both regulatory bodies consider components in harsh environment to generally not be serviceable (i.e., not able to be accessed to be replaced or maintained). However:
  - The CNSC defines harsh environment as a significant change in the normal ambient environment caused by a DBA. Harsh environment screening criteria (e.g., change in ambient temperature where the temperature rise exceeds 10 °C; the temperature exceeds 50 °C, steam, humidity, pressure, etc.) are

[19] Essential system is one necessary to shut down the reactor and mitigate the consequences of a postulated pipe rupture without offsite power.

    - conservatively based on electrical equipment and instruments and some mechanical equipment with limited passive safety functions (Annex A of CSA N290.13).
    - The NRC considers a harsh environment as one that is not mild, where a mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including AOOs.
- The CNSC relies on hazard barriers (shielding, etc.) for protection from harsh conditions. The NRC considers a combination of separation from the hazard and barriers to mitigate exposure conditions.
- For the NRC, equipment survivability for postulated severe accident conditions is demonstrated per 10 CFR 50.44. For the CNSC, REGDOC-2.5.2 Section 7.8 provides requirements and guidance on the use of survivability assessments under DEC.

8.3.5 Assessment of Impact

Overall, the EQ requirements are very similar. Both regulatory bodies consider essentially the same scope of equipment for qualification. The equipment subject to harsh environment qualification considers DBAs and the resulting environments over the equipment mission time as well as any mild environment aging effects in determining the qualification methodology. Environmental qualification is based on the same international consensus standard (IEC/IEEE 60780-323-2016) for both countries. Both countries include consideration of beyond-design-basis environments in determining survivability of equipment intended to operate after such events.

9 Interface with Standards Development Organizations (SDOs)

9.1 CNSC Interface with SDOs

The CNSC maintains an efficient and streamlined regulatory framework by making appropriate use of industry standards. Section 3.4, "Proven Engineering Practices," of REGDOC-2.5.2 states that:

> The design authority shall identify the modern codes and standards that will be used for the plant design, and evaluate those codes and standards for applicability, adequacy, and sufficiency to the design of SSCs important to safety.
>
> Where needed, codes and standards shall be supplemented to ensure that the final quality of the design is commensurate with the necessary safety functions. SSCs important to safety shall be of proven design and shall be designed according to the standards and codes identified for the nuclear power plant.

These include standards created by independent third-party standard-setting organizations, such as the CSA Group, the ASME, the International Commission on Radiological Protection, and the IEEE. Industry or international standards may be referenced in CNSC regulatory documents.

The CNSC's regulatory documents often reference CSA Group nuclear standards. These standards provide information on best practices and complement CNSC regulatory documents. The CNSC provides free access to all CSA Group nuclear standards.

9.2 NRC Interface with SDOs

The NRC uses and participates in the development of voluntary consensus codes and standards as an integral part of its regulatory framework. Codes and standards contain technical requirements, safety requirements, guidelines, characteristics, and recommended practices for performance. The NRC's participation in the development and use of consensus standards is consistent with the provisions of the National Technology Transfer and Advancement Act (NTTAA) (Public Law 104-113), OMB Circular A-119, "Federal Participation in the Development and Use of Voluntary Consensus Standards and in Conformity Assessment Activities," Management Directive (MD) 6.5, "NRC Participation in the Development and Use of Consensus Standards," and the Nuclear Energy Innovation and Modernization Act (NEIMA) (Public Law 115-439). These documents direct and guide the NRC's participation in the development and use of consensus codes and standards as follows:

- **NTTAA:** Directs Federal agencies to participate with voluntary consensus bodies in the development of consensus standards and adopt voluntary consensus standards as a means to carry out an agency's policy objectives or activities, wherever possible; also establishes reporting requirements.

- **OMB Circular A-119:** Provides guidance for agencies participating in the work of voluntary consensus standards bodies and describes procedures for satisfying the reporting requirements of the NTTAA.

- **NRC MD 6.5:** Provides direction to the NRC staff for implementing the NTTAA and OMB Circular A-119. Accordingly, it provides guidance and process for NRC's participation in the development and use of consensus codes and standards. The process consists of (1) identifying and prioritizing the need for new and revised technical standards, (2) participating in codes and standards development (on an equal basis with other committee members), and (3) endorsing codes and standards.

- **NEIMA:** Directs the NRC to collaborate with standards-setting organizations to (1) identify specific technical areas for which new or updated standards are needed to support the commercial advanced nuclear reactor licensing process and (2) incorporate the respective consensus-based codes and standards into the regulatory framework.

The NRC staff participates in several voluntary consensus SDOs, including the ASME, the ANS, the ASCE, the IEEE, and the NFPA, among other SDOs. NRC staff members participate and contribute at all levels of the standards development process, from working groups that write standards through consensus boards that vote on approval of standards to be promulgated by the SDO. The benefits of being actively involved in developing and using standards include improved safety, cost savings, improved efficiency and transparency, and regulatory requirements and guidance with high technical quality.

As an initiative to enhance agency use of standards and to exchange standards information with external stakeholders, the NRC hosts the NRC Standards Forum, usually once a year. The goals of the NRC Standards Forum are to facilitate discussions on codes and standards needs within the nuclear industry and to explore how to collaborate in accelerating the development of codes and standards and the subsequent NRC endorsement of codes and standards. The Forum encourages collaboration among stakeholders including researchers producing technical information and standards writers who build upon their findings. Additional information on the

NRC's participation in the development of codes and standards is available on the NRC website at: https://www.nrc.gov/about-nrc/regulatory/standards-dev.html.

The NRC may use a consensus code or standard as a mandatory requirement or as a voluntary provision. Mandatory use occurs through incorporation of a consensus code or standard in a regulation (e.g., 10 CFR 50.55a), license condition, order, or technical specification for individual licensees. Regulatory guides, which identify an acceptable method for applicants and licensees to comply with NRC regulations, as well as other regulatory guidance documents, provide a mechanism for allowing voluntary use of consensus codes or standards by applicants and licensees. Applicants or licensees can use codes and standards that may not have yet been endorsed by the NRC in which case the NRC performs a case-by-case review of such use. The NRC continues to facilitate the development and use of consensus codes and standards needed to support advanced reactors technologies. Enclosure 1 to SECY-23-0022 [129], "Advanced Reactor Program Status," discusses progress summary and future plans regarding the NRC's advanced reactor implementation action plan, including the strategic area on codes and standards. The discussion covers NRC staff activities related to several ASME, ANS, and ASCE standards. Regarding ASME standards, the NRC participates in working groups and subgroups associated with the development of ASME BPVC, Section III, Division 5 (Section III-5), "High Temperature Reactors," and Section XI, Division 2 (Section XI-2), "Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants." RG 1.87, Revision 2, and RG 1.246 endorse the 2017 Edition of Section III-5 and the 2019 Edition of Section XI-2, respectively.

Regarding ANS standards, the NRC participates in several ANS standards development working groups and consensus committees, including: the Research and Advanced Reactor Consensus Committee (RARCC), Risk-Informed, Performance-Based Principles and Policy Committee (RP3C), working group (WG) for ANS 53.1, "Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants," WG for ANS 20.1, "Nuclear Safety Criteria and Design Process for Fluoride Salt-Cooled High-Temperature Reactor Nuclear Power Plants," WG for ANS 20.2, "Nuclear Safety Design Criteria and Functional Performance Requirements for Liquid-Fuel Molten-Salt Reactor Nuclear Power Plants," and WG for ANS 30.2, "Categorization and Classification of Structures, Systems, and Components for New Nuclear Power Plants," amongst other.

Additionally, the NRC participates in the ASME/ANS Joint Committee on Nuclear Risk Management. Related to this committee, the NRC staff reviewed ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants," and endorsed it in RG 1.247 [130] (For Trial Use). Applicants may use this guidance on a trial basis. Before finalizing the RG, the NRC staff anticipates incorporating lessons learned from piloted applications of the guidance, as well as any revisions to ASME/ANS RA-S-1.4.

Also, the NRC participates in the standard committee for ASCE/SEI 43, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," among others. On October 3, 2022, to support early external stakeholder interactions, the NRC staff circulated two pre-decisional draft documents, "Technology-Inclusive, Risk-Informed, and Performance-Based Methodology for Seismic Design of Commercial Nuclear Plants," and "Seismically Isolated Nuclear Power Plants," for stakeholder feedback. These documents proposed the endorsement ASCE/SEI 43-19 and ASCE/SEI 4-16, "Seismic Analysis of Safety-Related Nuclear Structures."

As described above, the NRC actively participates in the development and use of voluntary consensus codes and standards as an integral part of its regulatory framework. Such participation improves the efficiency of the NRC's regulatory process and results in regulatory requirements and guidance with high technical quality.

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10 Conclusions

The collaborative review of the CNSC and the NRC licensing approaches and technical design expectations identified many areas of commonality that support the development of substantial technical information that could be shared in applications for SMRs or advanced reactors to both the CNSC and the NRC. These commonalities extend to the engineering design rules applied in the design, fabrication, and construction of components used in SMRs and advanced reactors. Although limited, differences in SSC classification and the resulting application of engineering design rules and specifications are most pronounced with respect to quality assurance measures applied to the most safety-significant SSCs, where NRC regulations are more prescriptive, and classification plays a significant role in the graded application of those measures. Another key area of difference is the effect of varying dose consequence target values under the various regulatory approaches for advanced reactor DBAs, which may affect the design of barriers to radionuclide release and siting of reactors if higher-consequence events are identified among LMP DBEs and higher frequency BDBEs. However, the CNSC and the NRC regulatory frameworks support the use of risk information to help ameliorate these differences.

Appendix A. Comparison of Design Criteria

License applications under 10 CFR Part 50 and 10 CFR Part 52 must include the principal design criteria (PDC) used in the preliminary design of the nuclear facility. The NRC general design criteria (GDC) in Appendix A, “General Design Criteria for Nuclear Power Plants,” of Title 10 of the Code of Federal Regulations (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” establish minimum requirements for the PDC for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The GDC are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the PDC for such other units. The NRC has provided guidance on how the GDC may be adapted for non-light-water reactor (non-LWR) designs in RG 1.232.

The CNSC provides comparable design criteria in REGDOC-2.5.2, Design of Reactor Facilities, Version 2.1. The table below aligns the NRC GDC with selected criteria from REGDOC-2.5.2 considered comparable. The table demonstrates considerable alignment in specific criteria applied to individual design elements and considerations. However, the NRC design criteria are somewhat more prescriptive based on their applicability to large LWRs similar to those previously licensed by the NRC. The REGDOC-2.5.2 criteria reflect somewhat greater flexibility reflecting licensing considerations associated with both CANDU and LWR reactors.

An example of this increased flexibility in the CNSC criteria is the treatment of the reactor core and reactivity control system designs. The NRC GDC prescribe criteria for inherent reactor protection in GDC 11 and reactivity control system performance in GDC 26 through GDC 28. In comparison, REGDOC-2.5.2 considers the core and reactivity control systems in an integrated manner that compensates for decreased reactor inherent protection with increased reactivity control system reliability standards.

NRC General Design Criteria	Comparable CNSC Criteria
Quality Standards and Records *Criterion 1—Quality standards and records*. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and	*REGDOC-2.5.2, Section 3.4, Proven Engineering Practices:* The design authority shall identify the modern codes and standards that will be used for the plant design, and evaluate those codes and standards for applicability, adequacy, and sufficiency to the design of SSCs important to safety. Where needed, codes and standards shall be supplemented to ensure that the final quality of the design is commensurate with the necessary safety functions. SSCs important to safety shall be of proven design and shall be designed according to the standards and codes identified for the nuclear power plant. *REGDOC-2.5.2, Section 3, Safety management in design:*

NRC General Design Criteria	Comparable CNSC Criteria
components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.	The applicant or licensee shall be ultimately responsible for the design of the reactor facility and shall establish a management system for ensuring the continuing safety of the plant design throughout the lifetime of the reactor facility. Processes, procedures and practices shall be established as part of the overall management system so as to achieve the design objectives. This shall include identifying all performance and assessment parameters for the plant design, as well as detailed plans for each SSC, in order to ensure consistent quality of the design and the selected components.

NRC General Design Criteria	Comparable CNSC Criteria
Design Bases for Protection Against Natural Phenomena *Criterion 2—Design bases for protection against natural phenomena*. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.	*REGDOC-2.5.2, Section 5.4.2, External hazards* All natural and human-induced external hazards that may be linked with significant radiological risk shall be identified. External hazards which the plant is designed to withstand shall be selected and classified as design basis accidents (DBAs) or design extension conditions (DECs). Applicable natural external hazards shall include such hazards as earthquakes, droughts, floods, high winds, tornadoes, tsunami, and extreme meteorological conditions. *REGDOC-2.5.2, Section 5.4.3, Combination of Events* Combinations of randomly occurring individual events that could credibly lead to anticipated operational occurrences (AOOs), DBAs, or DECs shall be considered in the design. Such combinations shall be identified early in the design phase and shall be confirmed using a systematic approach. *REGDOC-2.5.2, Section 5.15.1, Civil Structure – Design* External hazards such as earthquakes, floods, high winds, tornadoes, tsunamis, and extreme meteorological conditions shall be considered in the design of civil structures.
Fire Protection *Criterion 3—Fire protection*. Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of	*REGDOC-2.5.2, Section 5.12, Fire Safety* Suitable incorporation of operational procedures, redundant SSCs, physical barriers, spatial separation, fire protection systems, and design for fail-safe operation shall achieve the following general objectives: • prevent the initiation of fires • limit the propagation and effects of fires that do occur by:

NRC General Design Criteria	Comparable CNSC Criteria
appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.	◦ quickly detecting and suppressing fires to limit damage ◦ confining the spread of fires and fire by-products that have not been extinguished • prevent loss of redundancy in safety and safety support systems • provide assurance of safe shutdown • ensure that monitoring of safety-critical parameters remains available • prevent exposure, uncontrolled release, or unacceptable dispersion of hazardous substances, nuclear material, or radioactive material due to fires • prevent the detrimental effects of event mitigation efforts, both inside and outside of containment • ensure structural sufficiency and stability in the event of fire
Environmental and Dynamic Effects Design Bases *Criterion 4—Environmental and dynamic effects design bases*. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe	*REGDOC-2.5.2, Section 5.4.1, Internal Hazards:* SSCs important to safety shall be designed and located in a manner that minimizes the probability and effects of hazards (e.g., fires and explosions) caused by external or internal events. The plant design shall take into account the potential for internal hazards, such as flooding, missile generation, pipe whip, jet impact, fire, smoke, and combustion by-products, or release of fluid from failed systems or from other installations on the site. Appropriate preventive and mitigation measures shall be provided to ensure that nuclear safety is not compromised.

NRC General Design Criteria	Comparable CNSC Criteria
ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.	Internal events which the plant is designed to withstand shall be identified, and AOOs, DBAs and DECs shall be determined from these events. The possible interaction of external and internal events shall be considered, such as external events initiating internal fires or floods, or that may lead to the generation of missiles. REGDOC-2.5.2, Section 5.8, Equipment Environmental Qualification The design shall include an equipment environmental qualification program.… The environmental conditions to be accounted for shall include those expected during normal operation, and those arising from AOOs and DBAs. Operational data and applicable design assist analysis tools, such as the probabilistic safety assessment, shall be used to determine the envelope of environmental conditions.
Sharing of Structures, Systems, and Components *Criterion 5—Sharing of structures, systems, and components*. Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.	*REGDOC-2.5.2 Section 5.6.10, Sharing of SSCs between Reactors:* SSCs important to safety shall typically not be shared between two or more reactors. In exceptional cases when SSCs are shared between two or more reactors, such sharing shall exclude safety systems and turbine generator buildings that contain high-pressure steam and feedwater systems, unless this contributes to enhanced safety. If sharing of SSCs between reactors is arranged, then the following requirements shall apply: • safety requirements shall be met for all reactors during operational states, DBAs and DECs

NRC General Design Criteria	Comparable CNSC Criteria
	• in the event of an accident involving one of the reactors, orderly shutdown, cool down, and removal of residual heat shall be achievable for the other reactor(s)
Reactor Design *Criterion 10—Reactor design*. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. *Criterion 11—Reactor inherent protection*. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity. *Criterion 12—Suppression of reactor power oscillations*. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.	*REGDOC-2.5.2, Section 5.3.2, Anticipated Operational Occurrences:* The design shall include provisions such that releases to the public following an AOO do not exceed the dose acceptance criterion of 0.5 millisievert (mSv) (50 mrem) for average members of the critical groups at or beyond the site boundary over a 30-day period. The design shall also provide that, to the extent practicable, SSCs not involved in the initiation of an AOO shall remain operable following the AOO. The response of the plant to a wide range of AOOs shall allow safe operation or shutdown, if necessary, without the need to invoke provisions beyond Level 1 defence-in-depth or, at most, Level 2. *REGDOC-2.5.2, Section 6.1, Reactor core:* The design of the core shall be such that: 1. the fission chain reaction is controlled during operational states 2. the maximum degree of positive reactivity and its maximum rate of increase by insertion in operational states and DBAs are limited by a combination of the inherent neutronic characteristics of the core, its thermal-hydraulic characteristics, and the capabilities of the control system and means of shutdown, so that no resultant failure of the reactor pressure boundary will occur, cooling capability will be maintained, and no significant damage will occur to the reactor core.

NRC General Design Criteria	Comparable CNSC Criteria
Instrumentation and Control *Criterion 13—Instrumentation and control*. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	*REGDOC-2.5.2, Section 5.9, Instrumentation and Control:* The design shall include provision of instrumentation to monitor plant variables and systems over the respective ranges for operational states, DBAs and DECs, in order to ensure that adequate information can be obtained on plant status. This shall include instrumentation for measuring variables that can affect the fission process, the integrity of the reactor core, the reactor cooling systems, and containment, as well as instrumentation for obtaining any plant information that is necessary for its reliable and safe operation. The design shall be such that the safety systems and any necessary support systems can be reliably and independently operated, either automatically or manually, when necessary. *REGDOC-2.5.2, Section 8.1.2, Control systems:* The control system, combined with the inherent characteristics of the reactor and the selected operating limits and conditions, shall minimize the need for shutdown action. The control system and the inherent reactor characteristics shall keep all critical reactor parameters within the specified limits for a wide range of AOOs.
Reactor Coolant Pressure Boundary *Criterion 14—Reactor coolant pressure boundary*. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.	*REGDOC-2.5.2, Section 5.7, Pressure-retaining structures, systems, and components:* All pressure-retaining SSCs shall be protected against overpressure conditions, and shall be classified, designed, fabricated, erected, inspected, and tested in accordance with established standards. For DECs, relief capacity shall be sufficient to

NRC General Design Criteria	Comparable CNSC Criteria
	provide reasonable confidence that pressure boundaries credited in severe accident management will not fail. All pressure-retaining SSCs of the reactor coolant system and auxiliaries shall be designed with an appropriate safety margin to ensure that the pressure boundary will not be breached, and that fuel design limits will not be exceeded in operational states, or DBA conditions.
Reactor Coolant System Design *Criterion 15—Reactor coolant system design*. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.	*REGDOC-2.5.2, Section 6.2, Reactor Coolant System:* The design shall provide the reactor coolant system (RCS) and its associated components and auxiliary systems with sufficient margin to ensure that the appropriate design limits of the reactor coolant pressure boundary are not exceeded in operational states or DBAs.
Containment Design *Criterion 16—Containment design*. Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.	*REGDOC-2.5.2, Section 6.6, Containment:* Each nuclear power reactor shall be installed within a containment structure, so as to minimize the release of radioactive materials to the environment during operational states and DBAs. Containment shall also assist in mitigating the consequences of DECs. In particular, the containment and its safety features shall be able to perform their credited functions during DBAs and DECs, including melting of the reactor core. To the extent practicable, these functions shall be available for events more severe than DECs. The design shall include a clearly defined continuous leaktight containment envelope, the boundaries of which are defined for all conditions that could exist in the operation or maintenance of the reactor or following an accident.

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Electric Power Systems *Criterion 17—Electric power systems*. An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled, and containment integrity and other vital functions are maintained in the event of postulated accidents. The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure. Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few	*REGDOC-2.5.2, Section 6.9, Electrical power systems:* The design shall specify the required functions and performance characteristics of each electrical power system that provides normal, standby, emergency, and alternate power supplies to ensure: 1. sufficient capacity to support the safety functions of the connected loads in operational states, DBAs and DECs 2. availability and reliability are commensurate with the safety significance of the connected loads. The requirements of both the standby and emergency power systems may be met by a single system. *REGDOC-2.5.2, Section 6.9.1, Standby and emergency power systems:* The standby and emergency power systems shall have sufficient capacity and reliability, for a specified mission time, and in the presence of a single failure to provide the necessary power to 1. maintain the plant in a safe shutdown state and ensure nuclear safety in DBAs and DECs 2. support severe accident management actions *REGDOC-2.5.2, Section 6.9.2, DC and uninterruptible power systems:* The design of the direct current (DC) power systems and uninterruptible AC power systems (if applicable) shall specify operating mission times when performing the intended safety functions of the connected loads and meet the capacity requirements of Section 7.10.

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seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained. Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. *Criterion 18—Inspection and testing of electric power systems*. Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.	*REGDOC-2.5.2, Section 6.9.1, Standby and emergency power systems:* The standby and emergency power sources shall: 1. preferably be initiated automatically 2. be capable of being periodically tested under load conditions representing full load demand and full mission time *REGDOC-2.5.2, Section 6.9.2, DC and uninterruptible power systems:* The design shall include provisions for periodic testing for DC power and uninterruptible AC power supplies to confirm their capability.
Control Room *Criterion 19—Control room.* A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit	*REGDOC-2.5.2, Section 6.10, Control Facilities:* The design shall provide for a main control room (MCR) from which the plant can be safely operated, and from which measures can be taken to maintain the plant in a safe state or to bring it back into such a state after the onset of AOOs, DBAs or DECs.

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access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures. Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident. *TMI Action Plan Items* at 10 CFR 50.34 (f) (2) include the following NRC requirements related to control room design and shielding: (iii) Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of	The design shall identify events both internal and external to the MCR that may pose a direct threat to its continued operation and shall provide practicable measures to minimize the effects of these events. The safety functions that can be initiated by automatic control logic in response to an accident shall be capable of being initiated manually from the MCR. The layout of the controls and instrumentation, and the mode and format used to present information, shall provide operating personnel with an adequate overall picture of the status and performance of the plant and provide the necessary information to support operator actions. The design shall provide visual and, if appropriate, audible indications of plant conditions and processes that have deviated from normal operation and that could affect safety. The design shall also allow for the display of information needed to monitor the effects of the automatic actions of all control, safety, and safety support system. *REGDOC-2.5.2, Section 6.10.2 Secondary control room:* The design shall provide a secondary control room (SCR) that is physically and electrically separate from the MCR, and from which the plant can be placed and kept in a safe shutdown state when the ability to perform essential safety functions from the MCR is lost. The design shall identify all events that may pose a direct threat to the continued operation of the MCR and the SCR. The design of the MCR and the SCR shall be such that no event can simultaneously affect both control rooms to the extent that the

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fabricated control room panels and layouts. (I.D.1) (iv) Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. (I.D.2) (v) Provide for automatic indication of the bypassed and operable status of safety systems. (I.D.3) (vii) Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. (II.B.2)	essential safety functions cannot be performed. *REGDOC-2.7.1, Section 4.4.1 Engineered controls for radiation protection; Shielding:* The provision of shielding can be an effective form of engineered control. At the design stage, adequate thickness of the shield material should be provided to give an acceptable level of protection to the workers during normal as well as abnormal situations. The adequacy of the shielding in abnormal conditions, including accident situations leading to maximum foreseeable (worst-case) radiological consequences, should be evaluated and, where necessary, additional shielding or other engineered controls (e.g., interlocks) should be considered.

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Protection System Functions *Criterion 20—Protection system functions*. 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.	*REGDOC-2.5.2, Section 5.3, Plant states:* An anticipated operational occurrence (AOO) is a deviation from normal operation that is expected to occur once or several times during the operating lifetime of the reactor facility but that, in view of the appropriate design provisions, does not cause any significant damage to items important to safety, or lead to accident conditions. *REGDOC-2.5.2, Section 5.3.2, Anticipated operational occurrences:* The design shall include provisions such that releases to the public following an AOO do not exceed the dose acceptance criterion [of 0.5 mSV (50 mrem)] provided in Section 2.2.1. The design shall also provide that, to the extent practicable, SSCs not involved in the initiation of an AOO shall remain operable following the AOO. The response of the plant to a wide range of AOOs shall allow safe operation or shutdown, if necessary, without the need to invoke provisions beyond Level 1 defence-in-depth or, at most, Level 2. *REGDOC-2.5.2, Section 5.3.3 Design-basis accidents:* In order to prevent progression to a more severe condition that may threaten the next barrier, the design shall include provisions to automatically initiate the necessary safety systems when prompt and reliable action is required in response to a PIE [postulated initiating event].

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Protection System Reliability and Testability *Criterion 21—Protection system reliability and testability*. The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred. *Criterion 22—Protection system independence*. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function. *Criterion 23—Protection system failure modes*. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss	*REGDOC-2.5.2, Section 5.9, Instrumentation and control:* The design shall be such that the safety systems and any necessary support systems can be reliably and independently operated, either automatically or manually, when necessary. The design shall include provision for testing, including self-checking capabilities. The design shall provide for periodic testing of the entire channel of instrumentation logic, from sensing device to actuating device. The design shall facilitate maintenance, detection and diagnosis of failure, safe repair or replacement, and recalibration. *REGDOC-2.5.2, Section 5.6.1 Common-cause:* The potential for common-cause failures of items important to safety shall be considered in determining where to apply the principles of separation, diversity and independence so as to achieve the necessary reliability. Such failures could simultaneously affect a number of different items important to safety. The event or cause could be a design deficiency, a manufacturing deficiency, an operating or maintenance error, a natural phenomenon, a human-induced event, or an unintended cascading effect from any other operation or failure within the plant. *REGDOC-2.5.2, Section 5.6.5, Single failure criterion:* All safety groups shall function in the presence of a single failure. The single failure criterion requires that each safety group can perform all safety functions required for a PIE in the presence of any single component failure, as well as:

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of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced. *Criterion 24—Separation of protection and control systems*. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.	1. all failures caused by that single failure 2. all identifiable but non-detectable failures, including those in the non-tested components 3. all failures and spurious system actions that cause (or are caused by) the PIE *REGDOC-2.5.2, Section 5.6.6, Fail-safe design:* The concept of fail-safe design shall be incorporated, as appropriate, into the design of systems and components important to safety. To the greatest extent practicable, the application of this principle shall enable plant systems to pass into a safe state if a system or component fails, with no necessity for any action. *REGDOC-2.5.2, Sections 5.6.2, 5.6.3, and 5.6.4, generically address Separation, Diversity, and Independence, respectively, for safety systems.*
Reactivity Control System Redundancy and Capability *Criterion 25—Protection system requirements for reactivity control malfunctions*. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods. *Criterion 26—Reactivity control system redundancy and capability*. Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of	*REGDOC-2.5.2, Section 6.4 Means of Shutdown:* The design shall provide means of reactor shutdown capable of reducing reactor power to a low value, and maintaining that power for the required duration, when the reactor power control system and the inherent characteristics are insufficient or incapable of maintaining reactor power within the requirements of the Operational Limits and Conditions (OLCs). The design shall include 2 separate, independent, and diverse means of shutting down the reactor. At least one means of shutdown shall be independently capable of quickly rendering the nuclear reactor subcritical from normal operation in AOOs and DBAs, by an adequate margin, on the assumption of a

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normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions. *Criterion 27—Combined reactivity control systems capability*. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained. *Criterion 28—Reactivity limits*. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition. *Criterion 29—Protection against anticipated operational occurrences*. The protection and reactivity control systems	single failure. For this means of shutdown, a transient recriticality may be permitted in exceptional circumstances if the specified fuel and component limits are not exceeded. At least one means of shutdown shall be independently capable of rendering the reactor subcritical from normal operation, in AOOs and DBAs, and maintaining the reactor subcritical by an adequate margin and with high reliability, for even the most reactive conditions of the core. Means shall be provided to ensure that there is a capability to shut down the reactor in DECs, and to maintain the reactor subcritical even for the most limiting conditions of the reactor core, including severe degradation of the reactor core. Redundancy shall be provided in the fast-acting means of shutdown if, in the event that the credited means of reactivity control fails during any AOO or DBA, inherent core characteristics are unable to maintain the reactor within specified limits. The effectiveness of the means of shutdown (i.e., speed of action and shutdown margin) shall be such that specified limits are not exceeded, and the possibility of recriticality or reactivity excursion following a PIE is minimized. *REGDOC-2.5.2, Section 6.1, Reactor core:* The nuclear design should establish the following for reactivity device configurations, including (where applicable) control rod patterns, and reactivity worth for: • maximum worth of individual rods or banks as a function of position for power and lifecycle conditions appropriate to rod withdrawal, rod ejection (or drop) accidents and other conceivable failures of reactivity control components leading to positive reactivity insertions; and

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shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.	maximum rates of reactivity increase associated with reactivity device withdrawals and any other conceivable change in the configuration of reactivity devices due to failures in the reactivity control system. It should also include experimental confirmation of rod worth, or other factors justifying the reactivity increase rates used in control rod accident analyses, as well as equipment, administrative procedures and alarms that may be employed to restrict potential rod worth.
Quality of Reactor Coolant Pressure Boundary *Criterion 30—Quality of reactor coolant pressure boundary*. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. *Criterion 31—Fracture prevention of reactor coolant pressure boundary*. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state, and transient stresses, and (4) size of flaws.	*REGDOC-2.5.2, Section 3, Safety management in design* The design authority shall identify the modern codes and standards, and these codes and standards shall be supplemented to ensure that the final quality of the design is commensurate with the necessary safety functions. *REGDOC-2.5.2, Section 5.7, Pressure-retaining structures, systems, and components* The design shall minimize the likelihood of flaws in pressure boundaries. This shall include timely detection of flaws in pressure boundaries important to safety. For example, the reactor coolant pressure boundary should be designed with sufficient margin to ensure that, under all operating configurations, the material selected will behave in a nonbrittle manner and minimize the probability of rapidly propagating fractures. Pressure-retaining components whose failure will affect nuclear safety shall be designed to permit inspection of their pressure boundaries throughout the design life. If full inspection is not achievable, then it shall be augmented by indirect methods such as a program of surveillance of reference components.

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Criterion 32—Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.	*REGDOC-2.5.2, Section 6.2, Reactor coolant system* The design shall provide the reactor coolant system (RCS) and its associated components and auxiliary systems with sufficient margin to ensure that the appropriate design limits of the reactor coolant pressure boundary are not exceeded in operational states or DBAs. The design shall take into account all conditions of the boundary material in normal operation (including maintenance and testing), AOOs, DBAs and DECs, as well as expected end-of-life properties affected by aging mechanisms, the rate of deterioration, and the initial state of the components. The design shall provide a system capable of detecting and monitoring leakage from the reactor coolant system.
Reactor Coolant Makeup *Criterion 33—Reactor coolant makeup*. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.	*REGDOC-2.5.2, Section 6.2.2, Reactor coolant system inventory:* Taking volumetric changes and leakage into account, the design shall provide control of coolant inventory and pressure so as to ensure that specified design limits are not exceeded in operational states. This requirement shall extend to the provision of adequate capacity (flow rate and storage volumes) in the systems performing this function. The design should take into account the provision of adequate capacity, volumetric changes, leakage, flow rate and storage volumes in the systems performing this function.

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Residual Heat Removal *Criterion 34—Residual heat removal*. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.	*REGDOC-2.5.2, Section 6.2.4, Removal of residual heat from reactor core:* The design shall provide a means (i.e., backup) of removing residual heat from the reactor for all conditions of the RCS. The backup shall be independent of the configuration in use. The means of removing residual heat shall meet reliability requirements on the assumptions of a single failure and the loss of offsite power, by incorporating suitable redundancy, diversity, and independence. Interconnections and isolation capabilities shall have a degree of reliability that is commensurate with system design requirements. Heat removal shall be at a rate that prevents the specified design limits of the fuel and the reactor coolant pressure boundary from being exceeded.
Emergency Core Cooling *Criterion 35—Emergency core cooling*. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function	*REGDOC-2.5.2, Section 6.5, Emergency core cooling system* Nuclear reactor facilities shall be equipped with an emergency core cooling system (ECCS). The function of this safety system is to transfer heat from the reactor core following a loss of reactor coolant that exceeds makeup capability. All equipment required for correct operation of the ECCS shall be considered part of the system or its safety support system(s). Systems that supply electrical power or cooling water to equipment used in the operation of the ECCS shall be classified as safety support systems and shall be subject to all relevant requirements and expectations. The ECCS shall meet the following criteria for all DBAs involving loss-of-coolant:

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can be accomplished, assuming a single failure. *Criterion 36—Inspection of emergency core cooling system*. The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system. *Criterion 37—Testing of emergency core cooling system*. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.	1. All fuel assemblies and components in the reactor shall be kept in a configuration such that continued removal of the residual heat produced by the fuel can be maintained. 2. A continued cooling flow (recovery flow) shall be supplied to prevent further damage to the fuel after adequate cooling of the fuel is re-established by the ECCS. The ECCS recovery flow path shall be such that impediment to the recovery of coolant following a loss-of-coolant accident by debris or other material is avoided. The design shall ensure that maintenance and reliability testing can be carried out without a reduction in the effectiveness of the system below the OLCs, if the testing is conducted when ECCS availability is required. *REGDOC-2.5.2, Section 5.7, Pressure-retaining structures, systems, and components* Pressure-retaining components whose failure will affect nuclear safety shall be designed to permit inspection of their pressure boundaries throughout the design life.
Containment Heat Removal *Criterion 38—Containment heat removal*. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels. Suitable redundancy in components and features, and suitable interconnections,	*REGDOC-2.5.2, Section 6.6.9, Containment pressure and energy management:* The design shall enable heat removal and pressure reduction in the reactor containment in operational states, DBAs and DECs. Systems designed for this purpose shall be treated as part of the containment system, and are capable of: 1. minimizing the pressure-assisted release of fission products to the environment 2. preserving containment integrity

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leak detection, isolation, and containment capabilities shall be provided to assure…. *Criterion 39—Inspection of containment heat removal system*. The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system. *Criterion 40—Testing of containment heat removal system*. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.	3. preserving required leak tightness *REGDOC-2.5.2, Section 5.7, Pressure-retaining structures, systems, and components* Pressure-retaining components whose failure will affect nuclear safety shall be designed to permit inspection of their pressure boundaries throughout the design life. If full inspection is not achievable, then it shall be augmented by indirect methods such as a program of surveillance of reference components. *REGDOC-2.5.2, Section 5.14, In-service testing, maintenance, repair, inspection and monitoring:* In order to maintain the reactor facility within the boundaries of the design, the design shall be such that the SSCs important to safety can be calibrated, tested, maintained and repaired (or replaced), inspected, and monitored over the lifetime of the plant.
Containment Atmosphere Cleanup *Criterion 41—Containment atmosphere cleanup*. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents	*REGDOC-2.5.2, Section 6.6.10, Control and cleanup of the containment atmosphere* The design shall provide systems to control the release of fission products, hydrogen, oxygen, and other substances into the reactor containment, as necessary, to: 1. reduce the amount of fission products that might be released to the environment during an accident 2. prevent deflagration or detonation that could jeopardize the integrity or leak tightness of the containment

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to assure that containment integrity is maintained. Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure…. *Criterion 42—Inspection of containment atmosphere cleanup systems*. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems. *Criterion 43—Testing of containment atmosphere cleanup systems*. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.	*REGDOC-2.5.2, Section 5.7, Pressure-retaining structures, systems, and components* Pressure-retaining components whose failure will affect nuclear safety shall be designed to permit inspection of their pressure boundaries throughout the design life. If full inspection is not achievable, then it shall be augmented by indirect methods such as a program of surveillance of reference components. *REGDOC-2.5.2, Section 5.14, In-service testing, maintenance, repair, inspection and monitoring:* In order to maintain the reactor facility within the boundaries of the design, the design shall be such that the SSCs important to safety can be calibrated, tested, maintained and repaired (or replaced), inspected, and monitored over the lifetime of the plant.
Cooling Water *Criterion 44—Cooling water*. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components	*REGDOC-2.5.2, Section 6.7, Heat transfer to an ultimate heat sink* The design shall include systems for transferring residual heat from SSCs important to safety to an ultimate heat sink. This overall function shall be subject to very high levels of reliability during operational states, DBAs and DECs. All systems that contribute to the transport of heat by

NRC General Design Criteria	Comparable CNSC Criteria
under normal operating and accident conditions. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure. *Criterion 45—Inspection of cooling water system*. The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system. *Criterion 46—Testing of cooling water system*. The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.	conveying heat, providing power, or supplying fluids to the heat transport systems shall therefore be designed in accordance with the importance of their contribution to the function of heat transfer as a whole. *REGDOC-2.5.2, Section 5.7, Pressure-retaining structures, systems, and components* Pressure-retaining components whose failure will affect nuclear safety shall be designed to permit inspection of their pressure boundaries throughout the design life. If full inspection is not achievable, then it shall be augmented by indirect methods such as a program of surveillance of reference components. *REGDOC-2.5.2, Section 5.14, In-service testing, maintenance, repair, inspection and monitoring:* In order to maintain the reactor facility within the boundaries of the design, the design shall be such that the SSCs important to safety can be calibrated, tested, maintained and repaired (or replaced), inspected, and monitored over the lifetime of the plant.
Containment Design Basis *Criterion 50—Containment design basis*. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment	*REGDOC-2.5.2, Section 6.6.2, Strength of the containment structure:* The strength of the containment structure shall provide sufficient margins of safety based on potential internal overpressures, underpressures, temperatures, dynamic

NRC General Design Criteria	Comparable CNSC Criteria
structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters. *Criterion 51—Fracture prevention of containment pressure boundary*. The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.	effects such as missile generation, and reaction-forces anticipated to result in the event of DBAs. Strength margins shall be applied to access openings, penetrations, and isolation valves, and to the containment heat removal system. The margins shall reflect: 1. effects of other potential energy sources, such as possible chemical reactions and radiolytic reactions 2. limited experience and experimental data available for defining accident phenomena and containment responses 3. conservatism of the calculation model and input parameters The positive and negative design pressures within each part of the containment boundary shall include the highest and lowest pressures that could be generated in the respective parts as a result of any DBA. The seismic design of the concrete containment structure shall have an elastic response when subjected to seismic ground motions. The special detailing of reinforcement shall allow the structure to possess ductility and energy-absorbing capacity, which permits inelastic deformation without failure. *REGDOC-2.5.2, Section 5.7, Pressure-retaining structures, systems, and components* All pressure-retaining SSCs shall be protected against overpressure conditions, and shall be classified, designed, fabricated, erected, inspected, and tested in accordance with established standards. Unless otherwise justified, all pressure boundary SSCs shall be designed to withstand static and dynamic loads anticipated in operational states, and DBAs.

NRC General Design Criteria	Comparable CNSC Criteria
Provisions for Containment Testing and Inspection *Criterion 52—Capability for containment leakage rate testing*. The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure. *Criterion 53—Provisions for containment testing and inspection*. The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.	*REGDOC-2.5.2, Section 6.6.3 Capability for pressure tests:* The containment structure shall be subject to pressure testing at a specified pressure in order to demonstrate structural integrity. Testing shall be conducted before plant operation commences and at appropriate intervals throughout the plant's lifetime. *REGDOC-2.5.2, Section 6.6.4, Leakage:* The containment structure and the equipment and components affecting the leak tightness of the containment system shall be designed to allow leak rate testing: 1. for commissioning, at the containment design pressure 2. over the service lifetime of the reactor, in accordance with applicable codes and standards *REGDOC-2.5.2, Section 6.6.5 Containment penetrations:* All penetrations shall be designed to allow for periodic inspection and testing. If resilient seals such as elastomeric seals, electrical cable penetrations, or expansion bellows are used with penetrations, they shall have the capacity for leak testing at the containment design pressure. To demonstrate continued integrity over the lifetime of the plant, this capacity shall support testing that is independent of determining the leak rate of the containment as a whole.

NRC General Design Criteria	Comparable CNSC Criteria
Systems Penetrating Containment *Criterion 54—Piping systems penetrating containment*. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits. *Criterion 55—Reactor coolant pressure boundary penetrating containment*. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.	*REGDOC-2.5.2, Section 5.7, Pressure-retaining structures, systems, and components* Adequate isolation shall be provided at the interfaces between the reactor coolant system and connecting systems operating at lower pressures, in order to prevent the overpressure of such systems and possible loss-of-coolant accidents. Consideration shall be given to the characteristics and importance of the isolation and its reliability targets. Isolation devices shall either be closed or close automatically on demand. The response time and speed of closure shall be in accordance with the acceptance criteria defined for postulated initiating events. *REGDOC-2.5.2, Section 6.6.6, Containment isolation:* Each line of the reactor coolant pressure boundary that penetrates the containment, or that is connected directly to the containment atmosphere, shall be automatically and reliably sealed. This requirement is essential to maintaining the leak tightness of the containment in the event of an accident, and preventing radioactive releases to the environment that exceed prescribed limits. Automatic isolation valves shall be positioned to provide the greatest safety upon loss of actuating power. Piping systems that penetrate the containment system shall have isolation devices with redundancy, reliability, and performance capabilities that reflect the importance of isolating the various types of piping systems. Alternative types of isolation may be used where justification is provided.

NRC General Design Criteria	Comparable CNSC Criteria
Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety. Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for in-service inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs. *Criterion 56—Primary containment isolation*. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or	Where manual isolation valves are used, they shall be readily accessible and have locking or continuous monitoring capability. *Reactor coolant system auxiliaries that penetrate containment* Each auxiliary line that is connected to the reactor coolant pressure boundary, and that penetrates the containment structure, shall include 2 isolation valves in series. The valves shall be normally arranged with one inside and one outside the containment structure. Where the valves provide isolation of the heat transport system during normal operation, both valves shall be normally in the closed position. Systems directly connected to the reactor coolant system that may be open during normal operation shall be subject to the same isolation requirements as the normally closed system, with the exception that manual isolating valves inside the containment structure will not be used. At least one of the two isolation valves shall be either automatic or powered, and operable from the main and secondary control rooms. *Systems connected to containment atmosphere* Each line that connects directly to the containment atmosphere, that penetrates the containment structure and that is not part of a closed system shall be provided with two isolation barriers that meet the following requirements: 1. 2 automatic isolation valves in series for lines that may be open to the containment atmosphere 2. 2 closed isolation valves in series for lines that are normally closed to the containment atmosphere

NRC General Design Criteria	Comparable CNSC Criteria
(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety. *Criterion 57—Closed system isolation valves*. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.	3. the line up to and including the second valve is part of the containment envelope *Closed systems* All closed piping service systems shall have at least one single isolation valve on each line penetrating the containment, with the valve being located outside of, but as close as practicable to, the containment structure. Where failure of a closed loop is assumed to be a PIE or the result of a PIE, the isolations appropriate to the system shall apply.
Control of Releases of Radioactive Materials to the Environment *Criterion 60—Control of releases of radioactive materials to the environment*. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.	*REGDOC-2.5.2, Section 6.11, Waste treatment and control* The design of the reactor facility shall minimize the generation of radioactive and hazardous waste. The design shall also include adequate provision for the safe onsite handling and storage of radioactive and hazardous wastes, for a period of time consistent with options for offsite management or disposal. To ensure that emissions and concentrations remain within prescribed limits, the design shall include suitable means for controlling liquid releases to the environment in a manner that conforms to the As Low as Reasonably Achievable (ALARA) principle. This shall include a liquid waste management system of sufficient capacity to

NRC General Design Criteria	Comparable CNSC Criteria
	collect, hold, mix, pump, test, treat, and sample liquid waste before discharge, taking expected waste and accidental spills or discharges into account. The design shall include gaseous waste management systems capable of: 1. controlling all gaseous contaminants so as to conform to the ALARA principle and ensure that concentrations remain within prescribed limits 2. collecting all potentially active gases, vapours, and airborne particulates for monitoring 3. passing all potentially active gases, vapours, and airborne particulates through pre-filters, absolute filters, charcoal filters, or high efficiency particulate air filters where applicable 4. delaying releases of potential sources of noble gases by way of an off-gas system of sufficient capacity
Fuel Storage and Handling and Radioactivity Control *Criterion 61—Fuel storage and handling and radioactivity control*. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel	*REGDOC-2.5.2, Section 6.12, Fuel handling and storage:* The design of the handling and storage systems for irradiated fuel shall: 1. ensure nuclear criticality safety 2. permit adequate heat removal in operational states, DBAs and DECs… 9. provide proper means for radiation protection.… The design of the fuel handling and storage systems for non-irradiated fuel shall: 1. ensure nuclear criticality safety.… A design for a water pool used for fuel storage shall include provisions for:

NRC General Design Criteria	Comparable CNSC Criteria
storage coolant inventory under accident conditions. *Criterion 62—Prevention of criticality in fuel storage and handling*. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations. *Criterion 63—Monitoring fuel and waste storage*. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.	1. controlling the chemistry and activity of any water in which irradiated fuel is handled or stored 2. monitoring and controlling the water level in the fuel storage pool 3. detecting leakage 4. preventing the pool from emptying in the event of a pipe break…. *REGDOC-2.5.2, Section 6.13.3, Radiation monitoring:* Equipment shall be provided to ensure that there is adequate radiation monitoring in operational states, DBAs and DECs. Stationary alarming dose rate meters shall be provided: 1. for monitoring the local radiation dose rate at places routinely occupied by operating personnel… 3. to indicate, automatically and in real-time, the general radiation level at appropriate locations in operational states, DBAs and DECs 4. to give sufficient information in the control room or at the appropriate control location for operational states, DBAs and DECs, to enable plant personnel to initiate corrective actions when necessary
Monitoring Radioactivity Releases *Criterion 64—Monitoring radioactivity releases*. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated	*REGDOC-2.5.2, Section 5.9.3, Accident monitoring instrumentation:* Instrumentation and recording equipment shall be such that essential information is available to support plant procedures during and following DBAs and DECs by: 1. indicating plant status

NRC General Design Criteria	Comparable CNSC Criteria
operational occurrences, and from postulated accidents.	2. identifying the locations of radioactive material 3. supporting estimation of quantities of radioactive material *REGDOC-2.5.2, Section 6.13.5, Monitoring environmental impact:* The design shall provide the means for monitoring radiological and hazardous substances releases to the environment in the vicinity of the plant, with particular reference to: 1. pathways to the human population, including the food chain 2. the radiological impact, if any, on local ecosystems 3. the possible accumulation of radioactive and hazardous substances in the environment 4. the possibility of any unauthorized discharge routes

Appendix B. Comparison of Management System and Quality Assurance Attributes and Graded Approach to Implementation

Comparison of Management System and Quality Assurance Attributes

The CNSC and the NRC staffs compared the quality assurance (QA) and management system (MS) attributes for reactor license applicants (construction) and license holders (operation) using the respective regulations and associated implementation guidance. The comparison considers managerial, administrative, and quality assurance related attributes and relies primarily on the structure of 10 CFR Part 50, Appendix B, to structure the comparison. Table B-1 compares the quality assurance program criteria in 10 CFR Part 50, Appendix B, and select additional NRC requirements (i.e., implementing requirements for Appendix B, requirements for reporting nonconformances, and the Commission's Safety Culture Policy) with the CNSC implementing regulations and guidance from CSA Standard N286-12. The final column of Table B-1 lists an assessment of differences in the basic management system and quality assurance attributes.

Table B-1: Comparison of Quality Assurance and Management System Attributes

Attribute	NRC Applicability	CNSC Applicability	Assessment
Scope and Level of Detail	Managerial and administrative controls to ensure safe operation and quality assurance (QA) applied to design, fabrication, construction, and testing of SSCs. Pertinent QA requirements apply to all activities affecting the safety functions of safety-related SSCs. NRC requirements are more detailed and prescriptive.	Management system (MS) requirements apply throughout the lifecycle of the licensed activity and extend to all safety and control areas within the scope of the license.	Scope is similar in that both programs apply to the management of licensed activities to assure safe operation. CNSC scope includes additional considerations including health, environment, and security. NRC requirements are somewhat more detailed and prescriptive.
Gradation Within a Safety Classification	GDC-1 provides for graded approach; regulations require a distinction between quality groups for safety-related pressure-retaining components of LWRs. Criterion II of Appendix B provides control over activities affecting quality to an extent consistent with the importance to safety of affected SSCs.	Provisions in REGDOC-3.5.3 and Section 4.1.3 of CSA N286-12 for application commensurate with importance to safety. Specifies that criteria and grading process be defined. CSA N285.0 addresses quality groups for pressure-retaining components of CANDU reactors.	Both regulators specify graded quality groups for pressure-retaining components.

Attribute	NRC Applicability	CNSC Applicability	Assessment
Reporting Of Non-Compliant Conditions and Defective Components (QA)	10 CFR 50.55(e) (during construction) and 10 CFR Part 21: substantial safety hazards for defective SSCs that meet definition of basic component (equivalent to SR). Applies to organization that dedicates component (license holders or suppliers).	Licensees are fully responsible for assurance of conformance. If incorporated in the license, REGDOC-3.1.1 requires license holders to report counterfeit or substandard items.	NRC has prescriptive reporting requirements for defects and non-compliant conditions that apply to U.S. suppliers that dedicate services or SSCs as safety-related; both regulators establish requirements for license holders to report failure, degradation or weakening of safety-significant SSCs.
Organization (MS)	Criterion I of Appendix B: Defines duties and responsibilities of specific positions within organization; organizations performing quality assurance activities are independent of line organization – report directly to high-level management.	CSA N286-12, Clause 4.4 addresses organizational structure, authority and responsibility of positions, internal and external interfaces, and how and by whom decisions are made. Independence of QA organization not specified.	Equivalent requirements to define organization structure and establish responsibilities for each position. Independence of quality assurance organization from remainder of organization could potentially differ.
Training and Qualification (MS)	Criterion II of Appendix B states that the program shall provide for indoctrination and training necessary to assure suitable proficiency is achieved and maintained to implement the program. Special qualifications specified in individual attributes.	CSA N286-12 Clause 4.5.2 addresses competency and training requirements.	Equivalent requirements for training and qualification.
Establishment of QA Program (MS)	10 CFR 50.34(a)(7), 10 CFR 50.34(b)(6), and Criterion II of Appendix B: documented in written policies, procedures, and instructions that apply throughout the life of the facility. Implementation of program shall be regularly reviewed.	Class I Nuclear Facilities Regulations, REGDOC-2.2.2, and Clause 4.1 of CSA N286-12 address establishment of a quality assurance program. Requires periodic review by top management.	Both regulators require that management of quality be established at time of application and before significant activities begin. Extends throughout lifetime of facility, including decommissioning.

Attribute	NRC Applicability	CNSC Applicability	Assessment
Design Control (QA)	Criterion III of Appendix B: Translate design basis into appropriate specifications, drawings, procedures, and instructions. Coordination among design organizations. Measures shall be provided to independently check the adequacy of design, such as alternate calculational methods or performance of a testing program. Field design changes subject to same controls.	Section 3, "Safety management in design," of REGDOC-2.5.2 and Clause 7.3, "Design," of CSA N286-12 address design control measures.	Equivalent requirements for design control.
Procurement Documents (QA)	Criterion IV of Appendix B: Translate regulatory requirement, design bases, and other requirements into procurement documents.	Clause 7.6, "Supply chain," of CSA N286-12 addresses procurement, particularly Section 7.6.2, "Purchasing requirements."	Equivalent provisions to establish requirements for procurement of services and components.
Instructions and Procedures(MS)	Criterion V of Appendix B: Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these documents.	Clause 7 of CSA N286-12, especially those addressing Design, Construction, Commissioning, and Operating phases.	Equivalent requirements for instructions and procedures.
Document Change Control (MS)	Criterion VI of Appendix B: Measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings. Documents are reviewed and approved.	Clauses 4.10 (Change) and 7.5 (Configuration Management) of CSA N286-12.	Equivalent requirements for document change control.
Control of Contracted Equipment and Services (QA)	Criterion VII of Appendix B: Measures shall be established to assure that purchased material, equipment, and services conform to the procurement documents,	Clause 7.6 of CSA N286-12 (Supply Chain) addresses procurement of components and services.	Equivalent requirements for control of contracted equipment and services.

Attribute	NRC Applicability	CNSC Applicability	Assessment
	including provisions for source evaluation and selection, objective evidence of quality, inspection at the source, and examination of products upon delivery.		
Control of Material and Components (QA)	Criterion VIII of Appendix B: Identification of the item is maintained by appropriate means at all times. These identification and control measures shall be designed to prevent the use of incorrect or defective material, parts, and components.	Clause 7.6 of CSA N286-12 (Supply Chain) addresses identification of components and materials.	Equivalent requirements for control of material and components.
Control of Fabrication (QA)	Criterion IX of Appendix B: Measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.	Clause 7.7 of CSA N286-12 (Construction) addresses control of special processes.	Equivalent requirements for fabrication.
Inspection (QA)	Criterion X of Appendix B: A program for independent inspection of each important activity affecting quality shall be established by the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity.	Clause 4.8 of CSA N286-12 (Work Management).	Roughly equivalent requirements for inspection except degree of independence may differ.

Attribute	NRC Applicability	CNSC Applicability	Assessment
Test Control (QA)	Criterion XI of Appendix B: A test program shall be established to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service is performed in accordance with written test procedures which incorporate requirements and acceptance limits.	REGDOC-2.3.1 requires tests to be performed during the various phases of construction from prior to fuel load to high-power tests. Clauses 4.8.2 (Work Control), 7.8 (Commissioning), 7.9 (Operating) and 7.11.1 (Completion Assurance) of CSA N286-12.	Equivalent requirements for testing.
Measuring Equipment (QA)	Criterion XII of Appendix B: Tools, gauges, instruments, and other devices used in activities affecting quality are properly controlled, calibrated, and adjusted to maintain accuracy within necessary limits.	Clause 4.8.2 (Work Control) of CSA N286-12 includes requirements for control of measuring and test equipment Appendix E of REGDOC-2.3.1 specifies organizational responsibilities including ensuring a process is in place to control the calibration of test and measurement equipment.	Equivalent requirements for control of test and measuring equipment.
Handling and Storage (QA)	Criterion XIII of Appendix B: Handling, storage, shipping, cleaning and preservation of material and equipment is in accordance with work and inspection instructions to prevent damage or deterioration.	Clause 7.6.10 of CSA N286-12 (Storage and handling).	Equivalent requirements for handling and storage during construction.
Component Inspection and Test Status Marking (QA)	Criterion XIV of Appendix B: Measures shall be established to indicate the status of inspections and tests performed upon individual items to identify items that have satisfactorily passed tests or inspections and to indicate status.	Clause 4.7.4 (Records) of CSA N286-12 includes requirements for records to be traceable to the related item.	Roughly equivalent requirements for control of equipment status.
Nonconforming Materials or	Criterion XV of Appendix B: Procedures for identification,	Section 3.3.1 of REGDOC-2.1.1 and Clause 7.6 of CSA N286-	CNSC addresses counterfeit items and control of

Attribute	NRC Applicability	CNSC Applicability	Assessment
Components (QA)	documentation, segregation, disposition, and notification of nonconforming items shall be implemented to support dispositioning.	12 (Supply Chain) address nonconforming items.	nonconforming components during construction.
Corrective Action (MS)	Criterion XVI of Appendix B: Conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall determine the cause of the condition, require corrective action to preclude repetition, and require reporting to appropriate management.	Clause 4.9 (Problem Identification and Resolution) of CSA N286-12 addresses corrective action.	Equivalent requirements for corrective actions.
Records (MS)	Criterion XVII of Appendix B: Sufficient records shall be maintained to furnish evidence of activities affecting quality.	Clause 4.7.4 (Records) of CSA N286-12 addresses records.	Equivalent requirements for records.
Audits or Assessments (MS)	Criterion XVIII of Appendix B: A comprehensive system of independent audits shall be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program.	Clause 4.11 (Assessment) of CSA N286-12 addresses assessments.	Equivalent requirements for audits and assessments
Safety Culture	The Commission's Policy on Safety Culture encourages establishment and maintenance of a positive safety culture at both license holders and suppliers.	Clause 4.2 (Safety Culture) of CSA N286-12 specifies use of the management system to promote a positive culture.	Similar criteria, but incorporation of CSA N286-12 in the CNSC license would make the establishment of safety culture enforceable in Canada; while not enforceable in U.S.

Comparison of Graded Approach to Implementation of Quality Assurance Attributes

In Section 17.5, Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," the NRC established a less rigorous quality assurance structure to provide reasonable assurance of quality in systems that perform functions important to safety. This structure is referenced in guidance documents for both the important to safety components under the Traditional framework and components designated as Nonsafety-Related with Special Treatment components under the LMP framework. This structure primarily applies to components manufactured by suppliers and installed as finished components in the facility. The CNSC references a standard for assurance of quality in procured components and services, CSA N299-19, "Quality assurance program requirements for the supply of items and services for nuclear power plants," which provides a graded means of application based on safety significance.

As an example of the potential application of graded quality assurance, Table B-2 provides a list of NRC and CNSC guidance regarding individual quality assurance attributes intended for activities affecting equipment with lower safety significance. The NRC guidance is drawn from SRP Section 17.5 for nonsafety-related SSCs that are significant contributors to plant safety (i.e., items contributing to DID that are not within the scope of the QA program described in Appendix B to 10 CFR Part 50). The CNSC guidance primarily references CSA Standard N299-19 for a Category 3 approach to the design, procurement, and production of items under a supplier's internal quality assurance program with license holder oversight. Category 3 is applicable to items that may involve some complex processes or design changes in mass produced items whose failure in service could involve a limited risk to safety. This table reflects management and quality control measures expected to be incorporated into procurement contracts for manufactured components that may be used in nuclear power plants to satisfy DID functions. For the purposes of this example, the important to safety component is assumed to meet the Category 3 criteria in CSA N299.3. The table lists guidance that could be applied for license holder oversight and use of components manufactured by a supplier or vendor and assesses the degree of similarity. The comparison indicated that quality assurance attributes would be applied with similar levels of rigor (i.e., applied in a similar graded manner) under comparable guidance for SSCs with low to moderate safety significance.

Table B-2: Comparison of Quality Measures for Activities Affecting an SSC with Low to Moderate Safety Significance

Attribute	NRC Regulatory Treatment of Safety-Significant Nonsafety-Related SSCs (SRP Section 17.5, Paragraph II.U)	CNSC Management System Applicability to Activities Involving SSCs with Lower Safety Significance (CSA N299.3 used as basis)	Assessment of similarity in extent, depth, or timing of quality assurance attribute implementation
Scope	Managerial and administrative controls to ensure quality assurance (QA) applied to design, fabrication, and testing of	Managerial and administrative controls to ensure quality assurance (QA) applied to design, fabrication,	CNSC REGDOCs permit application of quality assurance on an applicant-defined spectrum compared with

Attribute	NRC Regulatory Treatment of Safety-Significant Nonsafety-Related SSCs (SRP Section 17.5, Paragraph II.U)	CNSC Management System Applicability to Activities Involving SSCs with Lower Safety Significance (CSA N299.3 used as basis)	Assessment of similarity in extent, depth, or timing of quality assurance attribute implementation
	SCCs important to safety other than those considered safety-significant and classified as safety-related.	and testing of SCCs important to safety with low to moderate safety significance and complexity.	one level addressed in NRC regulations and a reduced level for specific situations defined in guidance. In practice, Regulators expect QA measures would be applied under both regulatory frameworks in a similar step-wise fashion.
Graded Approach	Implements Commission Policy for Regulatory Treatment of Non-Safety Systems (RTNSS) in SECY 95-132 and GDC-1, which specifies that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of safety functions to be performed.	CNSC guidance and CSA standards permit application of applicant-defined graded approach. The difference between Category 1 and Category 3 of CSA N299-19 is representative of a potential graded approach to management of quality for important to safety SSCs of low to moderate safety significance.	The comparison of NRC QA guidance for important to safety SSCs classified as not safety-related with guidance drawn from CSA N299.3 is considered representative of how individual elements of quality management may be adjusted for lower safety significance SSCs.
Organization (MS)	Normal line organization may manage consistent with procedures. Independent verification not required.	The CSA standards do not require a separate organization. The degree of top-level management oversight is reduced and provisions for continuous improvement are not required for low safety significance SSCs.	Similar level of oversight for SSCs with low to moderate safety significance. The NRC guidance does not require organizational independence at this safety classification, which reflects grading of independence.
Establishment of QA Program (MS)	A separate or dedicated QA program is not required (i.e., may be implemented using procedures established under Appendix B QA Program).	The same QA program may govern different categories of equipment.	Neither regulator's guidance would require separate programs based on safety significance, but both regulators permit graded

Attribute	NRC Regulatory Treatment of Safety-Significant Nonsafety-Related SSCs (SRP Section 17.5, Paragraph II.U)	CNSC Management System Applicability to Activities Involving SSCs with Lower Safety Significance (CSA N299.3 used as basis)	Assessment of similarity in extent, depth, or timing of quality assurance attribute implementation
			levels of quality management.
Training and Qualification	Training of personnel addressed in SRP 17.5.	Specified in CSA N-286-12 for the licensee and CSA N299-19 for suppliers.	Training is specified.
Design Control (QA)	Contractually established design requirements and other applicable design inputs are included in design. Deviations are controlled. Normal Supervisory review of design considered adequate control.	Clause 7.4 of CSA N299.3 addresses changes to an existing mature design that require a minimal degree of effort and are of low complexity; simple independent verification specified. Clause 7.3 of CSA N286-12 lists areas that must be addressed by design; extent consistent with graded approach.	Similar extent of design verification for manufactured components with mature design. CSA N299.3 specifies slightly higher-level of independent design verification. New designs subject to similarly full scope of design control measures, but depth or extent of application may be graded by safety significance.
Procurement Documents (QA)	Applicable design bases and other design requirements necessary for adequate performance of ITS functions are included in procurement documents; deviations are controlled.	Clauses 7.4 – 7.6 of CSA N299.3 address design, documentation, and procurement for vendors and service providers. Clause 7.6 of CSA N286-12 specifies areas that must be defined in procurement documents; extent consistent with graded approach.	CNSC and NRC guidance provide for similar scope and information content in procurement documents.
Instructions and Procedures(MS)	Activities affecting quality shall be performed in accordance with documented instructions, procedures, or drawings. Instructions could be drawing notes or special instructions on work orders, provided appropriate degree of guidance is provided.	Clause 7.5 of CSA N299.3 address documentation. CSA N286-12 specifies areas that must be governed by documentation; extent consistent with graded approach.	Similar scope and level of detail for instructions.

Attribute	NRC Regulatory Treatment of Safety-Significant Nonsafety-Related SSCs (SRP Section 17.5, Paragraph II.U)	CNSC Management System Applicability to Activities Involving SSCs with Lower Safety Significance (CSA N299.3 used as basis)	Assessment of similarity in extent, depth, or timing of quality assurance attribute implementation
Document Change Control (MS)	Documents are controlled so that correct versions are used.	CSA N299.3 Clause 7.4.11 addresses design change control. CSA N286-12 applies to documents changed by license holder.	Similar guidance for document change control process.
Control of Contracted Equipment and Services (QA)	Measures are established to ensure purchased items and services conform to procurement documents.	CSA N299.3 Clause 7.6 addresses equipment and services contracted by supplier. CSA N286-12 applies to equipment and services contracted by license holder.	CSA guidance provides greater level of detail. but similar approach would satisfy both NRC and CNSC guidance.
Control of Material and Components (QA)	Measures are established where necessary to identify purchased items and store and deploy them in a manner that preserves their functional capability.	CSA N299.3, Clauses 7.11 and 7.12 specify that records are traceable to the related items and control of storage to limit deterioration.	Similar guidance. Extent of controls may be adjusted for lower safety significance SSCs under both NRC and CNSC guidance.
Control of Fabrication (QA)	Measures are established to control special processes such as welding, heat treatment, and nondestructive examination.	CSA N299.3 Clause 7.14 requires the control of special processes.	Similar guidance. Extent of controls may be adjusted for lower safety significance SSCs under both NRC and CNSC guidance.
Inspection (QA)	Inspections are performed to verify conformance of an item with applicable specified requirements and to verify activities are satisfactorily accomplished.	CSA N299.3 Clause 7.8 requires the verification of an item or service meeting the customer's documentation and technical requirements. CSA N286-12 applies to license holder verification of equipment.	Similar guidance.
Test Control (QA)	Measures are established to demonstrate that equipment conforms with design requirements and tests are performed in	CSA N299.3, Clause 7.8 requires the verification of an item or service meeting the customer's documentation and	Similar guidance.

Attribute	NRC Regulatory Treatment of Safety-Significant Nonsafety-Related SSCs (SRP Section 17.5, Paragraph II.U)	CNSC Management System Applicability to Activities Involving SSCs with Lower Safety Significance (CSA N299.3 used as basis)	Assessment of similarity in extent, depth, or timing of quality assurance attribute implementation
	accordance with test procedures that ensure test results are recorded and evaluated against requirements.	technical requirements. CSA N286-12 applies to license holder verification of equipment.	
Measuring Equipment (QA)	Measures ensure test and measuring equipment is controlled, calibrated, and adjusted at appropriate intervals.	CSA N299.3, Clause 7.10 requires a process for the selection, use, calibration, and control of measuring and test equipment.	Similar guidance. Extent of controls may be adjusted for lower safety significance SSCs under both NRC and CNSC guidance.
Handling and Storage (QA)	Handling and storage are controlled to prevent damage or loss and minimize deterioration.	CSA N299.3, Clause 7.11 specifies that records are traceable to the related items and control of storage to limit deterioration.	Similar guidance. Extent of controls may be adjusted for lower safety significance SSCs under both NRC and CNSC guidance.
Component Inspection and Test Status Marking (QA)	Measures are established to identify equipment that has satisfactorily passes required tests and inspections.	CSA N299.3, Clause 7.9 requires the identification of the inspection status of the item or service.	Similar guidance. Extent of controls may be adjusted for lower safety significance SSCs under both NRC and CNSC guidance.
Nonconforming Materials or Components (QA)	Items that do not conform with requirements are identified and controlled to prevent inadvertent installation or use.	CSA N299.3 Clause 7.17 specifies nonconforming items or services be identified, controlled to prevent unauthorized use, shipment, or mixing with conforming items and dispositioned.	Similar guidance. Extent of controls may be adjusted for lower safety significance SSCs under both NRC and CNSC guidance.
Corrective Action (MS)	Measures are established to ensure that failures, deficiencies, and nonconformances are properly identified, reported, and corrected.	CSA N299 Clause 7.18 specifies methods for identifying and reporting deficiencies and implementing corrective actions.	Similar guidance. Extent of corrective actions may be adjusted for lower safety significance SSCs under both NRC and CNSC guidance.
Records (MS)	Records are prepared and maintained to furnish evidence that the above requirements for design, procurement, document control, and inspection	CSA N299 Clause 7.16.1 requires records that are essential to provide evidence that items and services meet specified requirements and	Similar guidance.

Attribute	NRC Regulatory Treatment of Safety-Significant Nonsafety-Related SSCs (SRP Section 17.5, Paragraph II.U)	CNSC Management System Applicability to Activities Involving SSCs with Lower Safety Significance (CSA N299.3 used as basis)	Assessment of similarity in extent, depth, or timing of quality assurance attribute implementation
	and test activities have been met.	records that demonstrate the QA program has been effectively implemented.	
Audits or Assessments (MS)	Audits independent of line management are not required if line management periodically reviews and documents the adequacy of processes and takes any necessary corrective action. Line management determines if audits by independent organizations are necessary.	CSA N299.3 Clause 7.21.1 specifies internal quality audits to be performed. CSA N286-12 applies to license holder audits of procurement.	Similar guidance. Extent or independence of audits may be adjusted for lower safety significance SSCs under both NRC and CNSC guidance.
Reporting Of Non-Compliant Conditions and Defective Components (QA)	Not required outside organization.	Not required outside organization.	Reporting to regulator not specified due to decreased safety significance.

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