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Enclosure 3

EVR-SAR-GL-007-NP, Revision 0
“Use of LMP Methodology for the eVinci™ Microreactor Design”
(Non-Proprietary)

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REVISION SUMMARY

Revision	Revision Description
0	Initial Issue

OPEN ITEMS

Open Item #	Section	Open Item Description	Status
None			

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Acronyms and Trademarks

Acronyms used in the document are defined in EVR-PGD-GX-001 (Reference 1) or included below to ensure unambiguous understanding of their use within this document.

Acronym	Definition
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BDBA	Beyond Design Basis Accident
BDBE	Beyond Design Basis Event
CFR	Code of Federal Regulations
CNA	Canadian Nuclear Association
CNSC	Canadian Nuclear Safety Commission
DBA	Design Basis Accident
DBE	Design Basis Event
DCA	Standard Design Certification Application
DEC	Design Extension Condition
DID	Defence-in-Depth or Defense-in-Depth
DSA	Deterministic Safety Analysis
EPA	Environmental Protection Agency (U.S.)
F-C	Frequency-Consequence
HALEU	High-Assay, Low-Enriched Uranium
IAEA	International Atomic Energy Agency
ICE	Instrumentation, Control, and Electrical
LBE	Licensing Basis Event
LMP	Licensing Modernization Project
LTC	Licence to Construct
LTO	Licence to Operate
LTPS	Licence to Prepare Site
LWR	Light Water Reactor
NEI	Nuclear Energy Institute
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NSCA	Nuclear Safety and Control Act
NSRST	Non-Safety-Related with Special Treatment
NST	Non-Safety-Related with No Special Treatment
PAG	Protective Action Guide
PCS	Power Conversion System
PHX	Primary Heat Exchanger
PIE	Postulated Initiating Event
POS	Plant Operating State
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
RG	Regulatory Guide
RSF	Required Safety Functions
RXS	Reactor System
SR	Safety-Related
SSC	Structure, System, and Component

TEDE	Total Effective Dose Equivalent
TRISO	Tristructural Isotropic
VDR	Vendor Design Review

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Glossary of Terms

Standard terms used in the document are defined in EVR-PGD-GX-001 (Reference 1) or included below to ensure unambiguous understanding of their use within this document. The Canadian Nuclear Safety Commission (CNSC) uses different definitions in REGDOC-3.6, "Glossary of CNSC Terminology" (Reference 2), for the same term. Therefore, to improve clarity, definitions from both EVR-PGD-GX-001 and REGDOC-3.6 are provided below as separate glossaries.

CNSC Definitions:

Term	Definition
Anticipated Operational Occurrence (AOO)	**REGDOC-3.6**: An operational process deviating from normal operation that is expected to occur at least once during the operating lifetime of a reactor facility but, because of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions. AOOs have frequencies greater than $1x10^{-2}$/reactor-year. AOO is a facility state.
Beyond Design Basis Accident (BDBA)	**REGDOC-3.6**: An accident that occurs less frequently and is potentially more severe than a design-basis accident, BDBAs have frequencies less than $1x10^{-5}$/reactor-year. For a reactor facility, a beyond-design-basis accident may or may not involve fuel degradation.
Design Basis Accident (DBA)	**REGDOC-3.6**: Accident conditions for which a nuclear facility is designed according to established design criteria and for which damage to the fuel and the release of radioactive material are kept within authorized limits. DBAs have frequencies between $1x10^{-5}$/reactor-year and $1x10^{-2}$/reactor-year. DBA is a facility state.
Facility State	**REGDOC-3.6**: A configuration of reactor facility components, including the physical and thermodynamic states of the materials and the process fluids in them. This is equivalent to the term "plant state" used by the International Atomic Energy Agency (IAEA).
Important to Safety	**(REGDOC-2.5.2, Reference 3)**: Structures, systems, and components (SSCs) important to safety shall include safety systems, complementary design features, safety support systems, and other SSCs whose failure may lead to safety concerns (e.g., process and control systems). Appropriately designed interfaces shall be provided between SSCs of different classes in order to minimize the risk of having SSCs less important to safety adversely affecting the function or reliability of SSCs of greater importance. Important to safety is equivalent to Nuclear Energy Institute (NEI) 18-04 (Reference 4) safety-related (SR) and non-safety-related with special treatment (NSRST) SSCs.

eVinci Microreactor Definitions:

Term	Definition
Anticipated Operational Occurrence	Event sequences with mean frequencies of 1×10^{-2}/facility-year and greater are classified as AOOs. AOOs take into account the expected response of all SSCs within the facility, regardless of safety classification.
Beyond Design Basis Event (BDBE)	Event sequences with mean frequencies of 5×10^{-7}/facility-year to 1×10^{-4}/facility-year are classified as BDBEs. BDBEs take into account the expected response of all SSCs within the facility, regardless of safety classification.
Design Basis Accident (DBA)	Postulated accidents that are used to set design criteria and performance objectives for the design of SR SSCs. DBAs are derived from DBEs based on the capabilities and reliabilities of SR SSCs needed to mitigate and prevent accidents, respectively. DBAs are derived from the DBEs by prescriptively assuming that only SR SSCs classified are available to mitigate postulated accident consequences to within the frequency-consequence (F-C) limits.
Design Basis Event (DBE)	Event sequences with mean frequencies of 1×10^{-4}/facility-year to 1×10^{-2}/facility-year are classified as DBEs. DBEs take into account the expected response of all SSCs within the facility regardless of safety classification.
Licensing Basis Event (LBE)	The entire collection of event sequences considered in the design and licensing basis of the facility, which may include one or more reactor modules. LBEs include AOOs, DBEs, BDBEs, and DBAs.
Non-Safety-Related with Special Treatment (NSRST)	Non-safety-related SSCs that perform risk-significant functions or perform functions that are necessary for defense-in-depth adequacy.
Non-Safety-Related with No Special Treatment (NST)	All SSCs within a facility that are neither safety-related SSCs nor non-safety-related with special treatment SSCs.
Safety-Related	SSCs that are credited in the fulfillment of required safety functions (RSFs) and are capable to perform their RSFs in response to any design basis hazard level.

References

Following is a list of references used throughout this document.

1. EVR-PGD-GX-001, Revision 0, “Acronyms, Abbreviations, and Terms List,” Westinghouse Electric Company LLC.
2. REGDOC-3.6, “Glossary of CNSC Terminology,” Canadian Nuclear Safety Commission, February 2023.
3. REGDOC-2.5.2, Version 2.1, “Design of Reactor Facilities,” Canadian Nuclear Safety Commission, May 2023.
4. NEI 18-04, Revision 1, “Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” Nuclear Energy Institute, August 2019.
5. U.S. NRC and CNSC, “Memorandum of Cooperation on Advanced Reactor and Small Modular Reactor Technologies between the United States Nuclear Regulatory Commission and the Canadian Nuclear Safety Commission,” U.S. Nuclear Regulatory Commission, August 2019 (Available via NRC ADAMS Accession Number ML19275D578).
6. EVR_LTR_230011, NRC, “Notice of Intent for CNSC-NRC Joint Report Reviews in 2023,” Westinghouse Electric Company LLC.
7. EVR_LTR_230012, CNSC, “Notice of Intent for CNSC-NRC Joint Report Reviews in 2023,” Westinghouse Electric Company LLC.
8. CNSC-NRC joint report, “Technology Inclusive and Risk-Informed Reviews for Advanced Reactors: Comparing the US Licensing Modernization Project with the Canadian Regulatory Approach,” Canadian Nuclear Safety Commission, August 2021 (Available via NRC ADAMS Accession Number ML21225A101).
9. REGDOC-3.5.3, Version 3, “Regulatory Fundamentals,” Canadian Nuclear Safety Commission, March 2023.
10. REGDOC-2.4.1, “Deterministic Safety Analysis,” Canadian Nuclear Safety Commission, May 2014.
11. REGDOC-2.4.2, Version 2, “Probabilistic Safety Assessment (PSA) for Reactor Facilities,” Canadian Nuclear Safety Commission, May 2022.
12. Regulatory Guide 1.233, Revision 0, “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,” U.S. Nuclear Regulatory Commission, June 2020.
13. Canadian Nuclear Association (CNA) and Nuclear Energy Institute (NEI) Joint Report: “Canadian and United States Regulatory Cooperation for New Nuclear Deployment: Recommendations for Implementation of the International Regulatory Efficiency Framework,” Nuclear Energy Institute, September 2023.
14. CNSC-NRC, Interim Joint Report, “U.S. NRC – CNSC Memorandum of Cooperation concerning Classification and Assignment of Engineering Design Rules to Structures, Systems and Components,” Canadian Nuclear Safety Commission, June 2023 (Available via NRC ADAMS Accession Number ML23172A201).
15. RD-367, “Design of Small Reactor Facilities,” Canadian Nuclear Safety Commission, June 2011
16. 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” U.S. Nuclear Regulatory Commission.
17. 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission.
18. 10 CFR Part 20, “Standards for Protection Against Radiation,” U.S. Nuclear Regulatory Commission.
19. RA-S-1.4-2021, “Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants,” American Society of Mechanical Engineers, February 2021.
20. Regulatory Guide 1.247, Trial, “Acceptability of Probabilistic Risk Assessment Results for Non-Light Water Reactor Risk-Informed Activities,” U.S. Nuclear Regulatory Commission, March 2022.
21. NEI 20-09, Revision 0, “Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard,” Nuclear Energy Institute, March 2021.

22. International Atomic Energy Agency (IAEA) Safety Reports Series No. 46, “Assessment of Defence in Depth for Nuclear Power Plants,” International Atomic Energy Agency, 2005.
23. “Nuclear Safety and Control Act (NSCA),” Canada’s System of Justice, October 2023.
24. “Generic criteria and operational intervention levels for nuclear emergency planning and response,” Health Canada, 2018.
25. 51 FR 30028, “Policy Statement: Safety Goals for the Operation of Nuclear Power Plants; Policy Statement; Correction and Republication,” U.S. Nuclear Regulatory Commission, August 21, 1986.
26. EPA-400/R-17/001, “PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents,” U.S. Environmental Protection Agency, January 2017.

1.0 Introduction

Westinghouse Electric Company LLC (Westinghouse) has initiated review activities for the **eVinci™** microreactor with both the Canadian Nuclear Safety Commission (CNSC) and the U.S. Nuclear Regulatory Commission (NRC). In Canada, the Vendor Design Review (VDR) was initiated, and several Part 1 focus area reports have been provided for CNSC review. In the United States, white papers and topical reports have been submitted for feedback on the overall eVinci microreactor methodologies. Westinghouse plans to continue pre-application engagement in both countries to support licensing under the Canadian and U.S. regulatory frameworks.

To support these pre-application activities, Westinghouse finds it beneficial to have several eVinci microreactor topics reviewed jointly. As identified in the "Memorandum of Cooperation on Advanced Reactor and Small Modular Reactor Technologies between the Canadian Nuclear Safety Commission and the United States Nuclear Regulatory Commission" (Reference 5), and consistent with the objective to collaborate on pre-application reviews for advanced reactors, Westinghouse is submitting this report to the CNSC and the NRC for a joint review. This report is Westinghouse's first submission for CNSC and NRC joint review, as described in EVR_LTR_230011 to the NRC, "Notice of Intent for CNSC-NRC Joint Report Reviews in 2023" (Reference 6), and EVR_LTR_230012 to the CNSC, "Notice of Intent for CNSC-NRC Joint Report Reviews in 2023" (Reference 7). This pre-application activity takes place under the CNSC and NRC Memorandum of Cooperation (Reference 5).

1.1 Background

The joint CNSC and NRC report, "Technology Inclusive and Risk-Informed Reviews for Advanced Reactors: Comparing the U.S. Licensing Modernization Project with the Canadian Regulatory Approach," (Reference 8) (referred to as the "CNSC and NRC joint Licensing Modernization Project (LMP) report" herein) was previously issued. This report compares the LMP methodology documented in Nuclear Energy Institute (NEI) 18-04 (Reference 4) with CNSC REGDOCs, including the CNSC's regulatory philosophy documented in REGDOC-3.5.3 (Reference 9), the deterministic and probabilistic analysis approaches documented in REGDOC-2.4.1 (Reference 10) and REGDOC-2.4.2 (Reference 11), and the design approach documented in REGDOC-2.5.2 (Reference 3).

The CNSC and NRC joint LMP report identifies similarities in overarching safety goal philosophy with respect to ensuring that events with high consequences have very low frequencies of potential occurrence and that events with high frequencies of potential occurrence have very low consequences. The CNSC and NRC joint LMP report identifies further similarities in the development of qualitative and quantitative safety goals and objectives and a layered approach to design that ensures defence-in-depth (DID).

The CNSC and NRC joint LMP report also identifies key differences between the CNSC and NEI 18-04 risk-informed methodologies associated with postulated initiating event (PIE)/licensing basis event (LBE) frequency categorization, dose acceptance criteria and consequence targets, detailed probabilistic risk assessment (PRA)/probabilistic safety assessment (PSA) requirements, and overall risk metrics.

The LMP methodology documented in NEI 18-04 is endorsed by the NRC in Regulatory Guide (RG) 1.233 (Reference 12). CNSC and NRC regulatory documents frequently use similar terms for events and structure, system, and component (SSC) classification in different ways or use different terms for the same concept. The Glossary of Terms section herein includes key terms that have different or similar definitions from CNSC regulatory documents and those being used for the eVinci microreactor.

1.2 Purpose

The eVinci microreactor project is utilizing the NRC-endorsed NEI 18-04 (Reference 4) in the development of the supporting safety case. Westinghouse has also performed an assessment of the Canadian regulatory framework and concluded there are significant similarities, allowing most of the development work to be leveraged for deployment in Canada. However, the regulatory frameworks differ in the following areas: LBE frequency categorization, event frequency measurement, and dose acceptance criteria. This report summarizes Westinghouse's assessment of the two regulatory frameworks and proposes an approach to

address these differences while maintaining a singular reactor design capable of being deployed in Canada and the United States.

The purpose of this report is for Westinghouse to request CNSC and NRC feedback on its proposed approach for using the integrated, risk-informed, and performance-based LMP methodology documented in NEI 18-04 for development of the design, analysis, and safety case for the eVinci microreactor. [

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Feedback from the CNSC will be used to support Westinghouse's VDR effort for the eVinci microreactor. It will also support future work to develop licence applications in Canada that reference the eVinci microreactor design, including licence to prepare site (LTPS), licence to construct (LTC), and licence to operate (LTO). Feedback from the NRC will support Westinghouse's development of a standard design certification application (DCA) for review and approval by the NRC that may be referenced by future license applicants in the United States.

An ancillary purpose of this report is to provide information to the CNSC and the NRC on specific approaches Westinghouse is taking to inform and advance regulatory cooperation, as discussed in the Canadian Nuclear Association (CNA) and NEI joint report, "Canadian and United States Regulatory Cooperation for New Nuclear Deployment: Recommendations for Implementation of the International Regulatory Efficiency Framework" (Reference 13).

The CNA and NEI issued a joint report in September 2023 (Reference 13) that provides recommendations for improving collaborative efforts between the CNSC and NRC within the overall context of international regulatory efficiency. Westinghouse supports these efforts and believes that submittal of this report specifically advances the goals and recommendations discussed under Goal 2, "Regulatory Cooperation Agreements," and Goal 5, "Design Standardization" (Reference 13). Westinghouse aligns with the following statement from the CNA/NEI report and considers it important to the goal of developing a standardized eVinci microreactor design for Canada and the United States:

> Design standardization is thus enabled when the national regulator ensures that only the appropriate scope necessary for safety decisions is required to be in the design approval, and when the requirements and expectations between two or more regulators are extensively compatible. Here compatibility avoids the summation of the most conservative requirements between multiple countries but does not necessarily mean that the regulators must change their requirements to be identical.

1.3 Scope

This Westinghouse report relies on and expands upon the detailed comparison of the CNSC regulatory framework and the LMP methodology provided in the CNSC and NRC joint LMP report (Reference 8), specifically in relation to the regulatory limits for radiation exposure and dose acceptance criteria that have been determined by each regulator to ensure public health and safety. Westinghouse has also examined the CNSC and NRC interim joint report, "Classification and Assignment of Engineering Design Rules to Structures, Systems and Components" (Reference 14), to assess its relevance in the development of this report.

This report provides a summary and overview of the similarities and differences between the CNSC risk-informed regulatory framework and the LMP methodology previously discussed in the CNSC and NRC joint LMP report. It also describes the process for addressing the differences to establish the safety case for the eVinci microreactor to be licensed in Canada and the United States. The report also includes an overview of the eVinci microreactor design and facility description.

Appendix A of this report provides a summary level discussion of the approaches to nuclear regulation by the CNSC and NRC through establishment of qualitative safety goals and objectives, dose acceptance

limits, and regulatory requirements. Appendix B provides an overview of the LMP methodology, which is proposed by Westinghouse for the development of the eVinci microreactor design, analysis, and safety case under both CNSC and NRC regulatory frameworks.

1.4 Request for Regulators

Westinghouse requests that joint review of this report facilitate discussion regarding use of the LMP methodology documented in NEI 18-04 (Reference 4) for development of the design, analysis, and safety case for the eVinci microreactor for licensing in Canada and the United States.

Based on review of this report and ongoing pre-licensing discussions, Westinghouse is requesting regulator feedback and observations on the approach and information discussed herein. In addition, Westinghouse requests feedback from the CNSC and NRC on the following specific questions:

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- Has the CNSC or NRC identified any issues or concerns (e.g., any apparent challenges for implementation) related to the use of the LMP methodology for the design, analysis, and development of the safety case for the eVinci microreactor?

1.5 Applicable Regulatory Documents

Several regulatory documents associated with the development of a reactor safety case are applicable to this report. These include regulatory documents that provide requirements and guidance for the selection and categorization of PIEs/LBEs, classification of SSCs, risk significance assessments, safety analysis of PIEs/LBEs and their consequences, and DID adequacy assessments. Specific CNSC and NRC regulatory documents related to the Westinghouse proposed approach for using the LMP methodology are described in the following subsections.

1.5.1 Key Canadian Licensing and Guidance Documents

REGDOC-2.5.2, Version 2.1, “Design of Reactor Facilities” (Reference 3)

This regulatory document supersedes both REGDOC-2.5.2, Version 1.0, “Design of Reactor Facilities: Nuclear Power Plants,” which was published in May 2014, and RD-367, “Design of Small Reactor Facilities” (Reference 15), which was published in June 2011. REGDOC-2.5.2, Version 2.1 includes the following relevant requirements and guidance:

- Section 2.2.1 documents requirements and guidance relevant to dose acceptance criteria.
- Section 2.2.2 documents qualitative and quantitative safety goals and objectives.
- Section 5.1 documents requirements and guidance relevant to safety classification of SSCs.
- Section 9 documents the framework for the use of alternative approaches.

REGDOC-2.4.1, “Deterministic Safety Analysis” (Reference 10)

Part I of REGDOC-2.4.1 contains the CNSC’s requirements and guidance for performance of a deterministic safety analysis (DSA) for nuclear power plants. Part II of REGDOC-2.4.1 contains specific requirements and guidance for deterministic safety analysis for small reactor facilities.

Part II of REGDOC-2.4.1 includes the following relevant requirements and guidance:

- Section 6 documents requirements and guidance relevant to a graded approach.
- Section 7 summarizes the expectations for the deterministic analysis.
- Sections 8.1 and 8.2 document requirements and guidance for identifying events to be analyzed, the scope of the event analysis, and the classification of events based on frequency ranges.

- Section 8.3 documents requirements and guidance associated with acceptance criteria for event analysis.
- Section 8.4 documents requirements and guidance for the methods and assumptions used in the for deterministic safety analysis.

REGDOC-2.4.2, "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants" (Reference 11)

This regulatory document contains the CNSC's requirements for a PSA. CNSC requires a Level 1 and Level 2 PSA, but not a Level 3 PSA. The term PRA is used in the NRC regulatory framework and is considered equivalent to the PSA term used by the CNSC.

REGDOC-3.5.3, "Regulatory Fundamentals" (Reference 9)

This regulatory document is applicable to all CNSC regulatory activities. It contains no requirements for licensees or applicants but describes CNSC's regulatory approach and philosophy. The assessment of licensing submissions would be performed by CNSC staff using the regulatory approach described in this regulatory document. CNSC's regulatory approach is risk-informed and includes the option of using performance-based methods.

1.5.2 Key U.S. Licensing Regulations and Guidance Documents

Title 10 of the Code of Federal Regulations (CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Reference 16)

10 CFR 50.34, "Contents of applications; technical information," provides regulations for information to be included with a license application, including an assessment of the radiological consequences of postulated events to meet prescribed dose limits. Although the eVinci microreactor is intended to be licensed under 10 CFR Part 52 (Reference 17), the frequency-consequence (F-C) target curve is based on the dose limits associated with 10 CFR 50.34 that are also prescribed in 10 CFR 52.47 for a DCA and in 10 CFR 52.79 for a combined license application.

10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Reference 17)

10 CFR Part 52 provides regulations for the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities. 10 CFR 52.47, "Contents of applications; technical information," provides regulations for information to be included with an application for standard design certification, including an assessment of the radiological consequences of postulated events to meet prescribed dose limits that are identical to the 10 CFR 50.34 limits.

10 CFR Part 20, "Standards for Protection Against Radiation" (Reference 18)

10 CFR Part 20 provides radiation protection regulations. Subpart D, "Radiation Dose Limits for Individual Members of the Public," provides radiation dose limits from licensed operation of nuclear power plants for individual members of the public.

NEI 18-04, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development" (Reference 4)

NEI 18-04 describes the LMP methodology endorsed by the NRC in RG 1.233 (Reference 12). NEI 18-04 outlines a methodology for use by reactor developers to select LBEs, classify SSCs, determine special treatments and programmatic controls, and assess the adequacy of a design in terms of providing layers of defense (i.e., DID).

RG 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors" (Reference 12)

RG 1.233 provides NRC endorsement of the LMP methodology as one acceptable method for informing the design, analysis, and safety case, and for determining the appropriate scope and level of detail for parts of applications for licenses, certifications, and approvals for non-light water reactors (non-LWRs).

RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants" (Reference 19)

This American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) non-LWR PRA standard is a voluntary consensus standard that provides requirements for a comprehensive probabilistic radiological risk assessment. It addresses all radiological sources, hazards, plant operating states (POSs), and levels of analysis (e.g., from initiating event to radiological consequence) of PRA for non-LWRs. The NRC has endorsed use of this standard on a trial use basis by applicants for non-LWRs with emphasis for those using the NRC-endorsed LMP methodology.

RG 1.247, "Acceptability of Probabilistic Risk Assessment Results for Non-Light Water Reactor Risk-Informed Activities" (Reference 20)

RG 1.247 provides the NRC endorsement, for trial use and with exceptions, of the ASME/ANS non-LWR PRA standard RA-S-1.4-2-2021 (Reference 19) as one acceptable approach for determining whether a design-specific or plant-specific PRA used to support an application is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making.

Included in this regulatory guide is guidance on the performance of PRA peer reviews and an endorsement of NEI 20-09 (Reference 21), in its entirety and without exceptions, as a means of satisfying the peer review requirements in the ASME/ANS non-LWR PRA standard. Consistent with the scope of the ASME/ANS non-LWR PRA standard, NEI 20-09 (Reference 21) addresses PRA peer reviews for a non-LWR PRA that considers all radiological sources, hazards, POSs, and levels of PRA analysis.

2.0 Summary of the eVinci Microreactor Design and Facility Description

The eVinci microreactor is a 15 MW_t thermal neutron spectrum reactor that delivers high temperature heat from the reactor core through heat pipes and a primary heat exchanger (PHX) to an open-air Brayton power conversion system (PCS). The reactor system (RXS) design is shown in Figure 2-1.

The reactor core is enclosed within a canister filled with an inert gas just above atmospheric pressure to protect reactor components from oxidation while enhancing heat transfer. The core design consists of graphite blocks with repeated, segmented, hexagonal unit cells oriented horizontally along the length of the core. The unit cells contain channels for fuel, burnable absorbers, alkali metal heat pipes, and shutdown rods.

The reactor uses high-assay, low-enriched uranium (HALEU) tristructural isotropic (TRISO) fuel. The core is surrounded by a thick radial reflector that houses the control drums. The core alone, without the radial reflector, is subcritical, requiring the radial reflector to achieve criticality. Shielding is used to attenuate gamma and neutron radiation to protect site personnel and the public during operation and transportation. The PCS receives reactor heat from the PHX and converts it from 15 MW_t to 5 MW_e (nominal) with an open-air Brayton cycle.

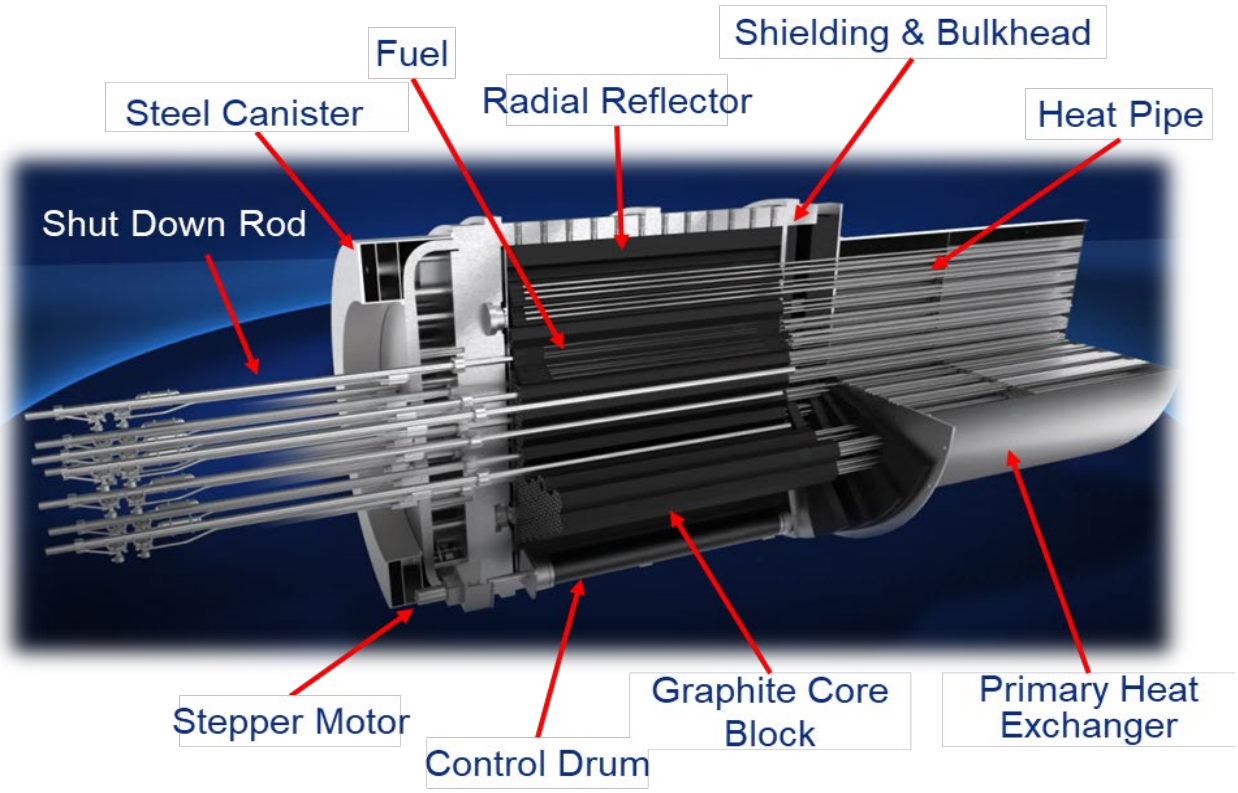

Figure 2-1 eVinci Microreactor Cutaway

The canister system does not function as a pressure vessel but instead as an element of the functional containment. During normal operation, the canister is pressurized just above atmospheric pressure with helium to eliminate oxidation of core components and increase thermal gap conductance. The design of the microreactor allows for decay heat removal through the core block, radial reflector, canister system, and shielding. Several layers of the TRISO fuel and the canister together represent the barriers that exist

to preclude the release of fission products to the environment and collectively represent the functional containment.

Reactivity control is accomplished using control drums located on the periphery of the core and burnable absorbers in the core. Reactivity is monitored using the power range and source range neutron detectors. Shutdown can be achieved by two diverse and independent means: the shutdown rods and the control drums. Additional shutdown rods are used to address hypothetical accident conditions associated with the transportation of a fueled reactor and maintain a subcritical reactor during transportation.

The reactor is installed in a transportation cask for transportation. The secondary system (i.e., the PCS) and support systems, including instrumentation, control, and electrical (ICE) systems, are transported in separate shipping containers. The shipping containers can be transported to remote locations via truck, rail, or waterway.

The site will be prepared prior to shipment of the reactor and support systems. Prior to the reactor arriving to the site, construction and installation activities will commence and will continue after the reactor's arrival to the site. Any necessary criticality testing will be performed after site construction and installation of the reactor. The site layout and connection between containers are designed to enable quick deployment. An illustration of the site layout is shown in Figure 2-2.

Figure 2-2 eVinci Microreactor Site Layout Rendering

Limited onsite staff is needed to perform the necessary site activities such as operations, maintenance, and security. A remote monitoring station will be used to allow remote personnel to monitor reactor power operations.

A replacement reactor will be shipped to, and installed at, the site as the operating reactor reaches its end of fuel life. Once the primary reactor reaches its end of fuel life, it is shut down and the replacement reactor will begin operation and become the new primary reactor. The shutdown reactor is allowed to cool before being transported off site for refurbishment and refueling or for decommissioning. Spent fuel is not required to be stored on site.

3.0 Westinghouse Approach to Implementation of LMP Methodology

For development of the design, analysis, and safety case for the eVinci microreactor under both the CNSC and NRC regulatory frameworks, Westinghouse proposes use of the LMP methodology. [

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This section provides a discussion related to implementation of aspects of the integrated LMP methodology that are key to similarities and differences with the CNSC regulatory framework.

3.1 Similarities between CNSC and NRC Regulatory Frameworks

The CNSC and NRC joint LMP report (Reference 8) identified similarities in overarching safety goal philosophy in ensuring that events with high consequences have very low frequencies of potential occurrence and that events with high frequencies of potential occurrence have very low consequences. The joint LMP report identified further similarities in the development of qualitative and quantitative safety goals and objectives and a layered approach to design that ensures DID. Such similarities are discussed in the following subsections.

3.1.1 Safety Goals and Objectives

The approaches used in the CNSC and NRC regulatory frameworks are both based on similar high-level qualitative safety goals and objectives:

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

These high-level goals pragmatically translate to demonstrating that:

1. More frequently occurring plant events have minor potential consequences.
2. Events with severe potential consequences have a very low frequency of occurrence.

These qualitative safety goals are both supported by quantitative safety goals that are expressed in terms of radiological risks.

3.1.2 Fundamental Safety Functions

The key fundamental safety functions established in both the CNSC and NRC frameworks to ensure protection of the health and safety of the public are essentially identical: (1) control of reactivity, (2) heat removal from the fuel, and (3) confinement of radioactive material.

REGDOC-2.5.2 (Reference 3) also lists additional safety functions that are necessary for safe operation and to respond to PIEs, such as, shielding, control of operational and accidental discharges of hazardous substances, and monitoring of safety-critical parameters to guide operator actions. These functions are also considered in the design of the eVinci microreactor.

3.1.3 Postulated Initiating Events/Licensing Basis Events

The PIE/LBE categorization used in the CNSC and NRC regulatory frameworks are similar and based on frequency of occurrence. These events are categorized in similar categories described as follows:

- Anticipated Operational Occurrences (AOOs): AOOs are events that are more complex than operational maneuvers with the potential to challenge the safety of the reactor and which might reasonably be expected to occur during the lifetime of the facility.
- Design Basis Accidents (DBAs)/Design Basis Events (DBEs): DBAs/DBEs are events that are not expected to occur during the lifetime of a facility but, in accordance with the principle of DID, are considered in the design of a facility.
- Beyond Design Basis Accidents (BDBAs)/Beyond Design Basis Events (BDBEs); BDBAs/BDBEs are extremely rare events. The CNSC follows the International Atomic Energy Agency (IAEA) approach and a subset of BDBA is established as design extension conditions (DECs). Both BDBEs and DECs are considered in the design.

Table 3-1 summarizes the comparison between PIE/LBE frequencies for the Canadian and U.S. approaches.

Table 3-1 Comparison of CNSC and LMP PIE/LBE Frequency Ranges

CNSC		LMP	
PIE Category	Frequency (per reactor-year)	LBE Category	Frequency (per plant-year)
AOOs	> $1x10^{-2}$	AOOs	> $1x10^{-2}$
DBAs	$1x10^{-2}$ to $1x10^{-5}$	DBEs (includes DBAs)	$1x10^{-2}$ to $1x10^{-4}$
BDBAs (includes DECs)	< $1x10^{-5}$ (Lower limit not defined)[2]	BDBEs[1]	$1x10^{-4}$ to $5x10^{-7}$

Notes:
1. Event sequences with lower frequencies than indicated here are retained in the PRA/PSA to ensure there are no cliff-edge effects, per NEI 18-04, Section 3.2.1 (Reference 4).
2. The CNSC does not set a lower frequency limit for BDBAs/DEC but requires the licensee or applicant to set a cut-off frequency such that events with lower frequency than the cut-off contribute negligible risk, per REGDOC-2.4.1, Section 8.2.2 (Reference 10).

As shown in Table 3-1 above, event sequences in the frequency range > $1x10^{-2}$ are considered AOOs in both the CNSC risk-informed approach and in the LMP methodology.

3.1.4 Safety Analyses

Deterministic Safety Analysis: The CNSC and NRC joint LMP report concluded that, in general, both the CNSC and NRC regulatory frameworks for performing DSA use similar assumptions regarding dose calculations for DBAs where the committed whole-body dose for an average member of the critical groups who are most at risk at or beyond the site boundary are calculated for a period of 30 days after an event that results in a release of radiation. Also, both regulatory approaches use a similar source term concept for the analysis. The source term calculation includes best estimate analysis of fission product release resulting from the specific accident sequences being evaluated using best estimate models.

Probabilistic Safety Analyses: In both the CNSC and NRC regulatory frameworks, PSA or PRA is a key element in the overall technology-inclusive and risk-informed approaches.

While the current approach being used for safety analyses by Westinghouse for the eVinci microreactor is based on the NRC-endorsed LMP methodology, [

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3.1.5 Safety Classification of SSCs

For the safety classification of SSCs, it is expected that the CNSC and NRC approaches will yield similar results. Both frameworks begin the safety classification by completing a functional analysis that identifies all safety functions needed to prevent or mitigate the postulated initiating events or event sequences. This functional analysis may include use of a combination of deterministic and probabilistic methods. Use of the LMP methodology establishes three safety classifications with a graded approach focused on safety and risk significance: (1) safety-related (SR), (2) non-safety-related with special treatment (NSRST), and (3) non-safety-related with no special treatment (NST). Similarly, in the CNSC framework, SSCs are classified as important to safety (equivalent to SR and NSRST SSCs, per NEI 18-04) or not important to safety (equivalent to NST, per NEI 18-04) using a consistent and clearly defined classification methodology to design, construct, and maintain those SSCs such that their quality and reliability is commensurate with the classification.

3.1.6 Defence-in-Depth

Evaluation of DID adequacy in the CNSC and NRC frameworks are very similar. Both approaches to DID adequacy are generally consistent with the concept of levels of defence described in IAEA Safety Reports Series No. 46, “Assessment of Defence in Depth for Nuclear Power Plants” (Reference 22). In addition, DID adequacy is evaluated for design, programmatic, and procedural elements in both frameworks. While the CNSC provides high-level guidance and points to the IAEA safety reports series No. 46 for detailed assessment requirements, the LMP methodology for evaluating DID adequacy consists of a detailed process that is more precisely defined for the programmatic DID evaluation.

3.2 Differences between CNSC and NRC Regulatory Frameworks

This section focuses on the key differences between the CNSC risk-informed regulatory framework and the LMP methodology associated with LBE frequency categorization and event frequency measurement and consequence targets**.**

The LMP methodology uses an F-C target curve to support categorization of LBEs, determine LBE risk significance, and inform classification of SSCs. As part of this process, the F-C target curve reflects the frequency ranges for LBE sequences and is also based on established dose acceptance limits. Integral to the LMP methodology and use of the F-C target curve is the identification of required safety functions (RSFs) and the selection of SSCs necessary to perform these RSFs to mitigate the consequences of DBEs to within the F-C target.

The PIE/LBE categorization used in the CNSC and NRC regulatory frameworks are very similar and generally based on frequency of occurrence. As discussed in Section 3.1.3, the CNSC and NRC joint LMP report (Reference 8) compared PIE/LBE frequency ranges. Table 3-1 provides a comparison of these frequency ranges.

In the CNSC framework, the events with frequencies between $1x10^{-2}$ and $1x10^{-5}$ per reactor-year are defined as DBAs and extend beyond the frequency range defined in LMP for DBEs and into the frequency range defined for BDBEs. The LMP BDBEs have a frequency of occurrence less than $1x10^{-4}$ per plant-year but with a lower bound frequency of greater than $5x10^{-7}$ per plant-year. Also, in the CNSC framework, BDBAs are events with a lower frequency than DBAs and do not include a lower threshold event frequency as in the LMP process. The LMP event sequences with frequencies less than $5x10^{-7}$ per plant-year are retained in the PRA results (i.e., included when determining cumulative risk metrics) and used to confirm there are no cliff-edge effects. In the CNSC framework, the events with frequencies lower than $1x10^{-5}$ per reactor-year are defined as BDBAs/DECs. Westinghouse plans to use frequency ranges based on event occurrences per facility-year rather than per plant-year, as described in NEI 18-04 during implementation of the LMP methodology for the eVinci microreactor, based on the terminology defined for the eVinci microreactor program.

Based on differences noted in the DBA/DBE frequency range for event sequences (i.e., between $1x10^{-4}$ and $1x10^{-5}$ per plant-year), as shown in Table 3-1, there is a potential for PIEs/LBEs to be categorized as DBAs under the CNSC approach that would otherwise be categorized as BDBEs under the LMP

methodology. As a result, the analysis methods and assumptions for those LBEs categorized as high-frequency BDBEs under the LMP methodology (i.e., between $1x10^{-4}$ and $1x10^{-5}$ per plant-year) would potentially need to be more conservative under the CNSC approach.

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The CNSC and NRC joint LMP report identified differences associated with consequence targets used in the LMP F-C target curve and the CNSC dose acceptance criteria. Table 3-2, which was modified from Table 9 in the CNSC and NRC joint LMP report (Reference 8), provides a comparison of these values.

Table 3-2 Comparison of Event Frequencies and Consequence Targets

CNSC			LMP		
PIE Category	**Frequency (per reactor-year)[1]**	**Dose Acceptance Criteria [2]**	**LBE Category**	**Frequency (per plant-year)[1]**	**Consequence Targets [2,3]**
AOOs	> 1×10^{-2}	0.5mSv (0.05 rem)	AOOs	High: > 1×10^{-1} Low: 1×10^{-1} to 1×10^{-2}	1 mSv (100 mrem) 10mSv (1 rem)
DBAs	1×10^{-2} to 1×10^{-5}	20 mSv (2 rem)	DBEs (includes DBAs)	1×10^{-2} to 1×10^{-4}	10mSv to 250 mSv (1 rem to 25 rem)
BDBAs / DEC	< 1×10^{-5} (lower limit not defined)	Limits not defined – apply safety goals	BDBEs	1×10^{-4} to 5×10^{-7}	250 mSv to 7500 mSv (25 rem to 750 rem)

Notes (not in original table in Reference 8):
1. Using the LMP process, the NRC classifies LBEs on a per plant-year basis, as opposed to the traditional per reactor-year basis. The purpose of using per plant-year is to address the event sequences involving multiple reactors (or reactor modules) and other non-reactor radiological sources at a plant.
2. Doses are calculated for a 30-day period for a person located at the site boundary, per REGDOC-2.5.2, Section 2.2.1 (Reference 3) or exclusion area boundary, per NEI 18-04 Section 3.2.1 (Reference 4). For the eVinci microreactor, site boundary and exclusion area boundary are assumed to be equivalent.
3. Per NEI 18-04, the F-C target provides a general reference to assess events, SSCs, and programmatic controls in terms of sensitivities and available margins and should not be considered as a demarcation of acceptable and unacceptable results.

Figure 3-1, which is based on Figure 9 of the CNCS and NRC joint LMP report (Reference 8), is provided to show a comparison of the CNSC PIE/LBE frequency ranges, and the CNSC dose acceptance criteria against the consequence targets in LMP F-C target curve.

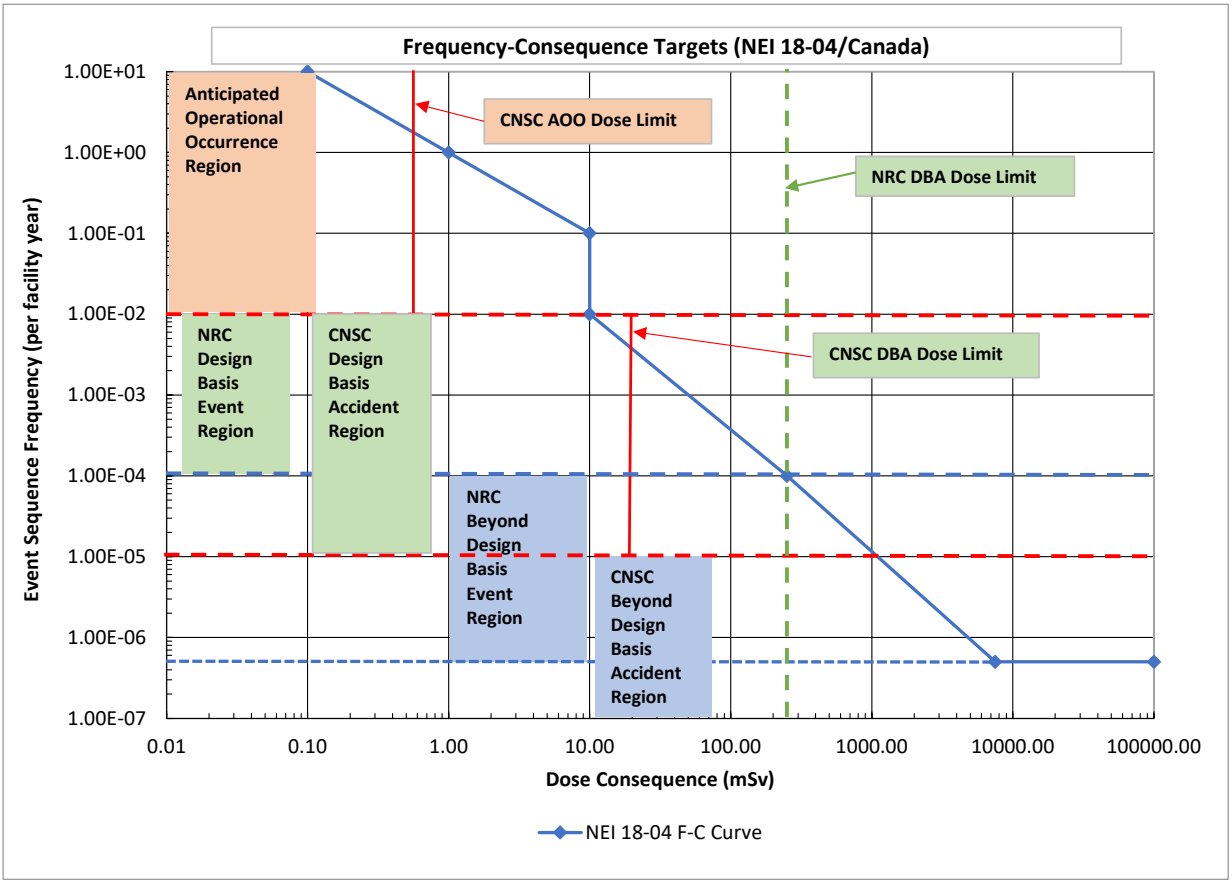

Figure 3-1 Comparison of CNSC PIE/LBE Frequency Ranges and LMP F-C Targets

Both the CNSC and the NRC-endorsed LMP approaches use a similar concept of mechanistic source term with realistic estimates of fission product release for the dose consequence calculation for DBAs. As shown in Table 3-2, the dose criteria associated with AOOs and DBAs/DBEs are different in the two regulatory frameworks.

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3.3 Summary of Comparison of LMP Methodology and CNSC Regulatory Framework

The fundamental objectives of CNSC and NRC for regulation of reactor facilities are in broad agreement and could be summarized as providing reasonable assurance of adequate protection of public health and safety from the risks posed by reactor facilities.

Achieving these fundamental regulatory objectives inherently involves judgement of reasonable assurance of adequate protection. Both the CNSC and NRC set equivalent qualitative health objectives and qualitative safety goals. These qualitative health objectives and safety goals should ensure that facilities:

- Do not add significantly to the risks of accidental death or death from cancer to which individuals are normally exposed.
- Are comparable to risks from other viable forms of energy production.

These qualitative safety goals anchor the requirement to provide adequate protection through comparison with risks normally experienced and accepted in society.

Section 3.1 identified similarities and common ground between the LMP methodology and the CNSC framework for the design and safety analysis topics. Section 3.2 identified and discussed key differences between the LMP methodology and the CNSC framework for the design and safety analysis topics. The CNSC and NRC joint LMP report also identified these differences and discussed how the CNSC and NRC approaches are implemented.

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For licensing in the United States, the LMP methodology is endorsed by the NRC in RG 1.233 (Reference 12). [

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4.0 Conclusions

Westinghouse reviewed the CNSC and NRC joint LMP report comparing the CNSC risk-informed regulatory framework with the NRC-endorsed LMP methodology (Reference 8). At the highest level, the CNSC and NRC joint LMP report concludes that the fundamental objectives of the CNSC and NRC for regulation of reactor facilities are in broad agreement and could be summarized as providing reasonable assurance of adequate protection of public health and safety from the risks posed by reactor facilities.

Westinghouse proposes use of the LMP methodology for development of the design, analysis, and safety case for the eVinci microreactor under both the CNSC and NRC regulatory frameworks. While the CNSC and NRC joint LMP report identified similarities and common ground between the LMP methodology and the CNSC frameworks, the joint report also identified some differences in how the approaches are implemented and in the acceptance criteria used and employed. [

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Westinghouse intends to comply with the requirements in the applicable CNSC regulatory documents and codes/standards [

]^{a,c} Specifically, as noted in the CNSC and NRC joint LMP report, the dose acceptance criteria and dose limits are different in the two regulatory frameworks. [

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Westinghouse supports the efforts of CNA and NEI to develop recommendations for improving regulatory cooperation between the CSNC and the NRC, as documented in their September 2023 report (Reference 13). This Westinghouse report on use of the LMP methodology advances specific goals and recommendations discussed under Goal 2, “Regulatory Cooperation Agreements,” and Goal 5, “Design Standardization.” Westinghouse considers the focus on regulatory cooperation agreements and design standardization to be important to its goal for developing a standardized eVinci microreactor design for Canada and the United States.

Appendix A CNSC and NRC Regulatory Framework

This appendix presents a top-down examination of the regulatory framework requirements and guidance for developing the eVinci microreactor design, analysis, and safety case for Canada and the United States. A detailed comparison of the CNSC risk-informed regulatory framework and the LMP methodology was previously performed by the CNSC and NRC and is provided in the joint LMP report (Reference 8). Relevant differences between specific requirements in the CNSC and NRC regulatory frameworks are identified and summarized from the joint report below. Further details for additional background and information, as needed, are available in the CNSC and NRC joint LMP report.

A.1 CNSC APPROACH

Licensing and operation of nuclear facilities and activities in Canada are overseen by the CNSC, which was established under Canada's Nuclear Safety and Control Act (NSCA) (Reference 23). The CNSC regulates all areas related to nuclear energy; however, this report is focused on the licensing for reactor facilities.

Portions of the NSCA directly impact the requirements for design and analysis of reactor facilities, in particular the limitation of risks to a reasonable level. Section 9 of the NSCA states:

> The objects of the Commission are
>
> > (a) to regulate the development, production and use of nuclear energy and the production, possession and use of nuclear substances, prescribed equipment and prescribed information in order to
> >
> > > (i) prevent unreasonable risk, to the environment and to the health and safety of persons, associated with that development, production, possession or use,
> > >
> > > (ii) prevent unreasonable risk to national security associated with that development, production, possession or use, and
> > >
> > > (iii) achieve conformity with measures of control and international obligations to which Canada has agreed; and
> >
> > (b) to disseminate objective scientific, technical and regulatory information to the public concerning the activities of the Commission and the effects, on the environment and on the health and safety of persons, of the development, production, possession and use referred to in paragraph (a).

The CNSC provides regulatory documents to provide guidance and requirements for applicants and licensees. Regulatory documents applicable to this report are documented in Section 1.5. Guidance contained in these regulatory documents exists to inform the applicant, elaborate on requirements, or provide direction to licensees and applicants on how to meet requirements. They also provide more information about how CNSC staff evaluates specific problems or data during their review of licence applications.

CNSC guidance also states that applicants and licensees are expected to review and consider the guidance; however, the regulatory framework provides flexibility for licensees to propose alternative means of achieving the intent of the requirement.

As noted in REGDOC-2.5.2 (Reference 3), the CNSC endorses a general nuclear safety objective to support the NSCA and its associated regulations. This general nuclear safety objective is supported by three complementary safety objectives that deal with radiation protection, the technical safety aspects of the design, and environmental protection. All three safety objectives focus on reasonable protection from radiological consequences during normal plant operations and postulated accident scenarios.

The technical safety objectives are meant to provide all reasonably practicable measures to prevent accidents in the reactor facility, and to mitigate the consequences of accidents if they do occur. This considers all possible accidents considered in the design, including those of very low probability. When these technical safety objectives are achieved, any radiological consequences will be below prescribed

limits, and the likelihood of accidents with serious radiological consequences will be extremely low. The NSCA and these technical safety objectives provide the basis for the CNSC to establish safety goals and dose acceptance criteria. Safety analyses must be performed to confirm that these criteria and goals are met, demonstrating effectiveness of measures for preventing accidents and mitigating radiological consequences of accidents if they do occur.

Section 2.2.2 of REGDOC-2.5.2 (Reference 3) documents the CNSC's Qualitative Safety Goals, which are as follows:

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

CNSC's quantitative safety goals provide numerical targets that meet the qualitative goals. They are given in Section 2.2.2 of REGDOC-2.5.2 (Reference 3) as follows:

- Core Damage Frequency – The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than 10^{-5} per reactor year.
- Small Release Frequency – The sum of frequencies of all event sequences that can lead to any release to the environment that requires temporary evacuation of the local population or a release to the environment of more than 10^{15} becquerels of iodine-131 shall be less than 10^{-5} per reactor year.
- Large Release Frequency – The sum of frequencies of all event sequences that can lead to any release to the environment that requires long-term relocation of the population or a release to the environment of more than 10^{14} becquerels of cesium-137 shall be less than 10^{-6} per reactor year.

In addition to the safety goals described above, to meet the technical safety objectives, Section 4.3.2 of REGDOC-2.4.1 (Reference 10) states that analysis for AOOs and DBAs shall demonstrate that: (1) radiological doses to members of the public do not exceed the established limits and (2) the derived acceptance criteria, established in accordance with Section 4.3.4 of REGDOC-2.4.1 are met. CNSC sets dose acceptance criteria for individual AOO and DBA event sequences at less than or equal to 0.5 mSv for any AOO or 20 mSv for any DBA. CNSC's dose acceptance criteria given in Section 4.3.2 of REGDOC-2.4.1 (Reference 10) and are based on:

- AOO – Half the annual dose limit for normal operation.
- DBA – Set below Health Canada's "Generic criteria and operational intervention levels for nuclear emergency planning and response" (Reference 24), which recommends evacuation of the population if the projected whole-body dose exceeds 50 mSv in seven days.

The dose acceptance criteria are applied across frequency bands for postulated event sequences established in Section 8.2.3 of REGDOC-2.4.1 (Reference 10). These criteria establish frequency ranges that support classification of event sequences into facility states.

A.2 U.S. NRC APPROACH

Licensing and operation of nuclear facilities and activities in the United States are overseen by the NRC. The NRC's mission is to license and regulate civilian use of radioactive materials, to provide reasonable assurance of adequate protection of public health and safety, to promote the common defense and security, and to protect the environment.

NRC's regulations are documented in Title 10 of the Code of Federal Regulations. For licensing of nuclear reactors, certification of designs, and setting of dose limits, the most relevant parts are 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (Reference 16); 10 CFR Part 52, "Licenses,

Certifications, and Approvals for Nuclear Power Plants" (Reference 17); and 10 CFR Part 20, "Standards for Protection Against Radiation" (Reference 18).

Qualitative safety goals are set in 51 FR 30028, "Policy Statement on Safety Goals for the Operation of Nuclear Power Plants" (Reference 25), which states:

> The qualitative safety goals are as follows:
>
> - Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
> - Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

Quantitative safety objectives are also set in 51 FR 30028 (Reference 25), which states:

> The following qualitative safety objectives are to be used in determining the achievement of the above safety goals:
>
> - The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
> - The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes.

Dose limits are established in both 10 CFR 50.34(a)(1)(ii)(D) (Reference 16) and 10 CFR 52.47(a)(1)(iv) (Reference 17) as follows:

> […] The evaluation must determine that:
>
> (A) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem[4] total effective dose equivalent (TEDE).
>
> (B) An individual located at 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) would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).
>
> [Footnote [4] omitted]

The dose limit of 25 rem (250 mSv) is specifically established for evaluation of a hypothetical accident with substantial meltdown of the core. As noted in these regulations, the required evaluation assesses safety features of the facility design, including focusing on containment performance with respect to fission product releases, fission product cleanup systems, and containment leakage rates.

The consequence targets/ranges established for AOOs, DBA/DBEs, and BDBEs under the LMP methodology are described in Section 3.2.1 of NEI 18-04 (Reference 4). These consequence targets/ranges are also shown in the F-C target curve that is used as an assessment tool in the LMP methodology and are documented in Figure 3-1 of this report**.** The F-C target curve documented in NEI 18-04 sets the consequence target for lower frequency AOOs to a reference value of 10 mSv (1 rem), which corresponds to the U.S. Environmental Protection Agency (EPA) "PAG Manual: Protective Action Guide" (Reference 26). This is consistent with the goal of avoiding the need for offsite emergency response for any AOO.

Use of the F-C target curve as part of the integrated LMP methodology implements the principle that low frequency events must have minor consequences and high consequence events must have low frequencies. The LMP methodology also includes a new risk target that integrates risk for all event sequences within the licensing basis. This integrated risk target is included in the three cumulative risk

targets described in Task 7b of NEI 18-04, Section 3.2.2 (Reference 4) and specifies that the total mean frequency of exceeding a site boundary dose of 1 mSv (100 mrem) from all LBEs should not exceed 1/facility-year. This new risk target is selected from the annual cumulative exposure limits in 10 CFR Part 20 (Reference 18). The remaining two risk targets are associated with the NRC's Quantitative Health Objectives (Reference 4):

- The average individual risk of early fatality within 1.6 kilometers (1 mile) of the exclusion area boundary from all LBEs shall not exceed 5×10^{-7}/plant-year to ensure that the plant meets the NRC safety goal quantitative health objective for early fatality risk.
- The average individual risk of latent cancer fatalities within 16 kilometers (10 miles) of the exclusion area boundary from all LBEs shall not exceed 2×10^{-6}/plant-year to ensure that the plant meets the NRC safety goal quantitative health objective for latent cancer fatality risk.

Appendix B Overview of the LMP Methodology

The LMP methodology integrates the main tasks performed to support design and safety analysis with the safety case utilizing the concept of DID. The process includes:

- Selection and categorization of LBEs and determination of LBE risk significance
- Classification of SSCs
- Determination of special treatments and programmatic controls
- Assessment of risk and safety significance against a risk-informed F-C target curve and cumulative probabilistic risk targets
- Demonstration of the adequacy of DID

The LMP methodology provides a systematic process to derive the appropriate list of LBEs as one acceptable process to assist with meeting requirements for establishing the safety case. This process begins with use of a F-C target curve to assist in determining the categorization of event sequences based on mean frequencies and a determination of their risk significance as shown below in Figure B-1, which is reproduced from Figure 3-1 of NEI 18-04 (Reference 4). The specific considerations and bases used for the construction of this F-C target curve are described in Section 3.2.1 of NEI 18-04.

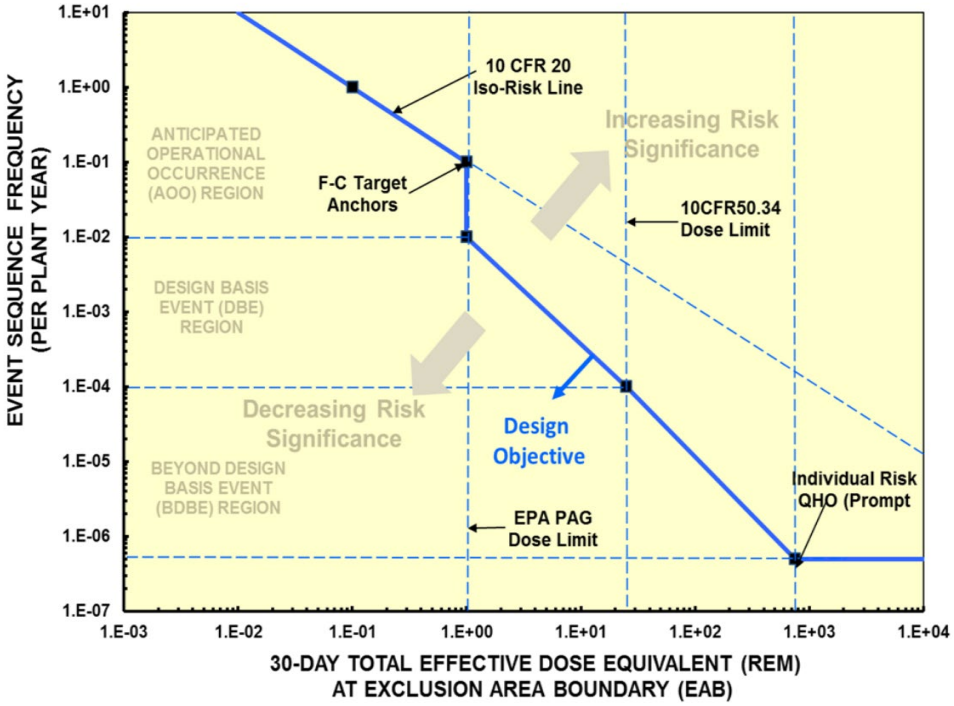

Figure B-1 Frequency-Consequence Target Curve

The purpose of the F-C target curve is to categorize LBEs, assess their risk significance, and determine the resultant safety classification of SSCs. It is not intended to be used to assess compliance with regulatory limits on radiation exposure.

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 non-reactor radiological sources.

The LMP methodology for selecting LBEs is based on the event sequences delineated by a PRA's logic model (e.g., the collection of event trees and fault trees). An event sequence is an initiating event defined for a set of initial plant conditions followed by a sequence of system, safety function, and operator action

failures or successes leading to a specified end state (i.e., prevention of release or release in one of several reactor-specific release categories).

The term “event sequence” is used in the LMP methodology to emphasize that: (1) all sequences delineated by the PRA need to be considered and (2) some sequences delineated by the PRA do not result in a release of radioactive material. LBEs are categorized as AOOs, DBEs, and BDBEs. LBE categories are based on mean event sequence frequency of occurrence per facility-year where a facility may be comprised of multiple reactor modules. LBEs may or may not involve release of radioactive material and may involve two reactor modules or radionuclide sources.

Although the mean values of the frequencies are used to classify the LBEs into AOOs, DBEs, and BDBE categories, when the uncertainty band for the frequency estimates straddles a frequency boundary, the LBE is evaluated in both LBE categories. For example, with respect to application to an eVinci facility, an LBE with mean frequency above 10^{-2}/facility-year and 5th percentile less than 10^{-2}/facility-year is evaluated as an AOO and DBE and an LBE with a mean frequency less than 10^{-4}/facility-year with a 95th percentile above 10^{-4}/facility-year is evaluated as a BDBE and a DBE. An event sequence family with a mean frequency less than $5x10^{-7}$/facility year but with a 95th percentile frequency estimate above $5x10^{-7}$/facility-year is evaluated as a BDBE. Uncertainties about the mean values are used to help evaluate the results against the frequency-consequence criteria and to identify the margins against the criteria.

The process for selecting and evaluating LBEs is illustrated below in Figure B-2, which is reproduced from Figure 3-2 of NEI 18-04 (Reference 4). The uncertainties about these means are considered as part of the DID evaluation in Task 7e.

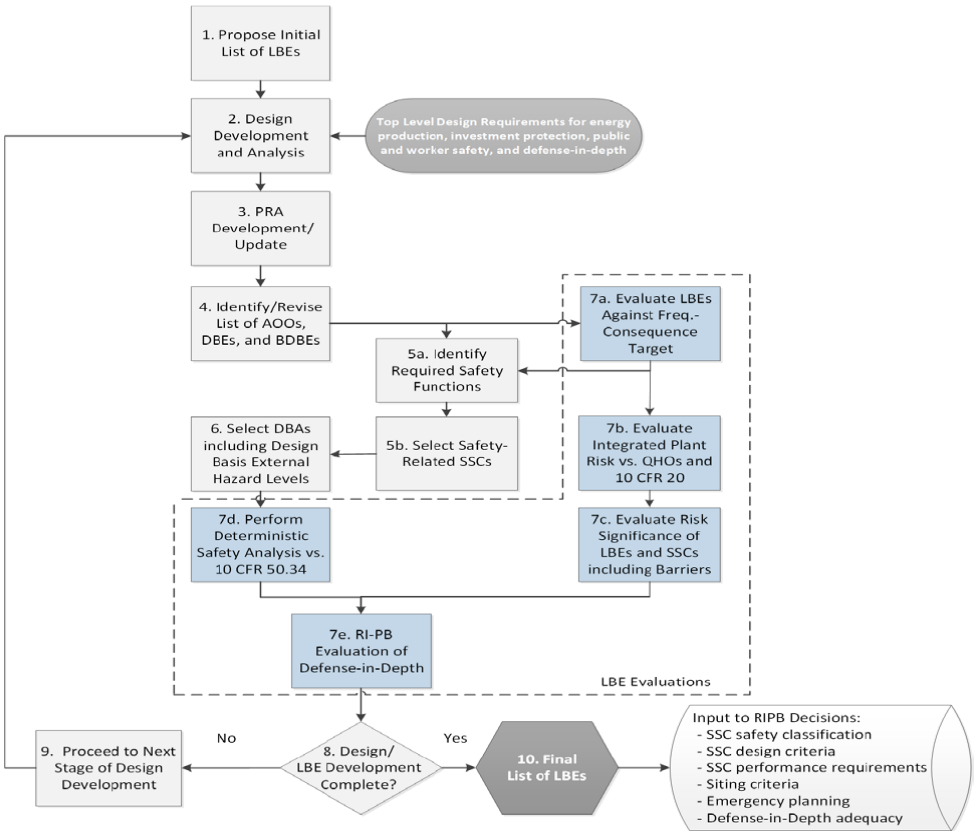

Figure B-2 Process for Selecting and Evaluating Licensing Basis Events

Central to the evaluation of individual LBEs and determination of their risk significance using the F-C target curve is the identification of RSFs and the selection of SSCs necessary to perform these RSFs to mitigate the consequences of DBEs to within the LBE F-C target. These are classified as SR SSCs. Structures and

physical barriers necessary to protect any SR SSCs performing their RSFs in response to any design basis hazards are also classified as SR. Included in this SSC selection is a determination of their assumed capability and reliability to perform RSFs. Further assessment of DID supports the establishment of special treatments to ensure high reliability and capability of the SSCs due to their importance to ensuring DID adequacy, as discussed further below.

SR SSCs are also selected for any RSF associated with any high-consequence BDBEs where the reliability of the SSC is necessary to keep the event in the BDBE frequency region. In addition, since DBAs use conservative assumptions and only rely on SR SSCs to meet the dose limits of 10 CFR 50.34 (Reference 16), SSCs needed to mitigate DBAs are also classified as SR. The remaining SSCs that are not classified as SR are considered in other evaluation tasks as part of Tasks 7b, 7c, and 7e, as outlined in Figure B-2. These other evaluation tasks may determine that SSCs require special treatments to support DID adequacy and are classified as NSRST. Special treatment is defined in NEI 18-04 and refers to those requirements that provide increased assurance beyond normal industrial practices (e.g., commercial grade) that SSCs perform their design-basis functions. Section 4.4.5 of NEI 18-04 (Reference 4) provides a detailed discussion of special treatment requirements for SSCs. Examples of special treatments include enhanced quality assurance requirements, reliability assurance requirements, and application of specific design requirements to support reliability and capability requirements, as determined by a DID adequacy assessment (e.g., seismic). All remaining SSCs are classified as NST.

The process illustrated in Figure B-2 reflects an integrated and iterative process that may result in design changes, reclassification of SSCs, and/or the identification of additional special treatments.

A more detailed, comprehensive, and integrated assessment of DID adequacy is also performed as part of the LMP methodology. The LMP methodology embraces the concept of layers of defense for evaluating the adequacy of DID and provides an approach to establishing DID in design, construction, maintenance, and operation of nuclear facilities.

Establishing DID adequacy involves incorporating DID design features, operating and emergency procedures, and other programmatic elements. DID adequacy is evaluated by using a series of risk-informed and performance-based decisions regarding design, facility risk assessment, and selection; the DID adequacy assessment also evaluates categorization of LBE risk significances, safety classification of SSCs, specification of performance requirements for SSCs, and programs to ensure these performance requirements are maintained throughout the life of the facility (i.e., reliability and capability). The LMP methodology for establishing DID adequacy embraces the concept of layers of defense and uses these layers to identify and evaluate DID attributes.

Elements of the DID adequacy assessment are included in Tasks 7e and 8 as shown Figure B-2. The more detailed, comprehensive, and integrated assessment of DID adequacy is illustrated in Figure B-3, which is reproduced from Figure 5-4 of NEI 18-04.

Although this is the last step in the overall LMP methodology, it illustrates how this process includes and integrates iteration steps for the interrelated processes for selection of LBEs, safety classification, and performance criteria of SSCs. It also demonstrates the evaluation of DID adequacy that leads to convergence of the design and analysis. Use of this integrated process culminates in a safety case that demonstrates effective compliance with the safety goals and objectives, which is confirmed by an assessment of DID.

Section 5 of NEI 18-04 provides further details on this integrated methodology. Sections 3 and 4 of NEI 18-04 provide further details on the processes for selection of LBEs and safety classification and performance criteria of SSCs.

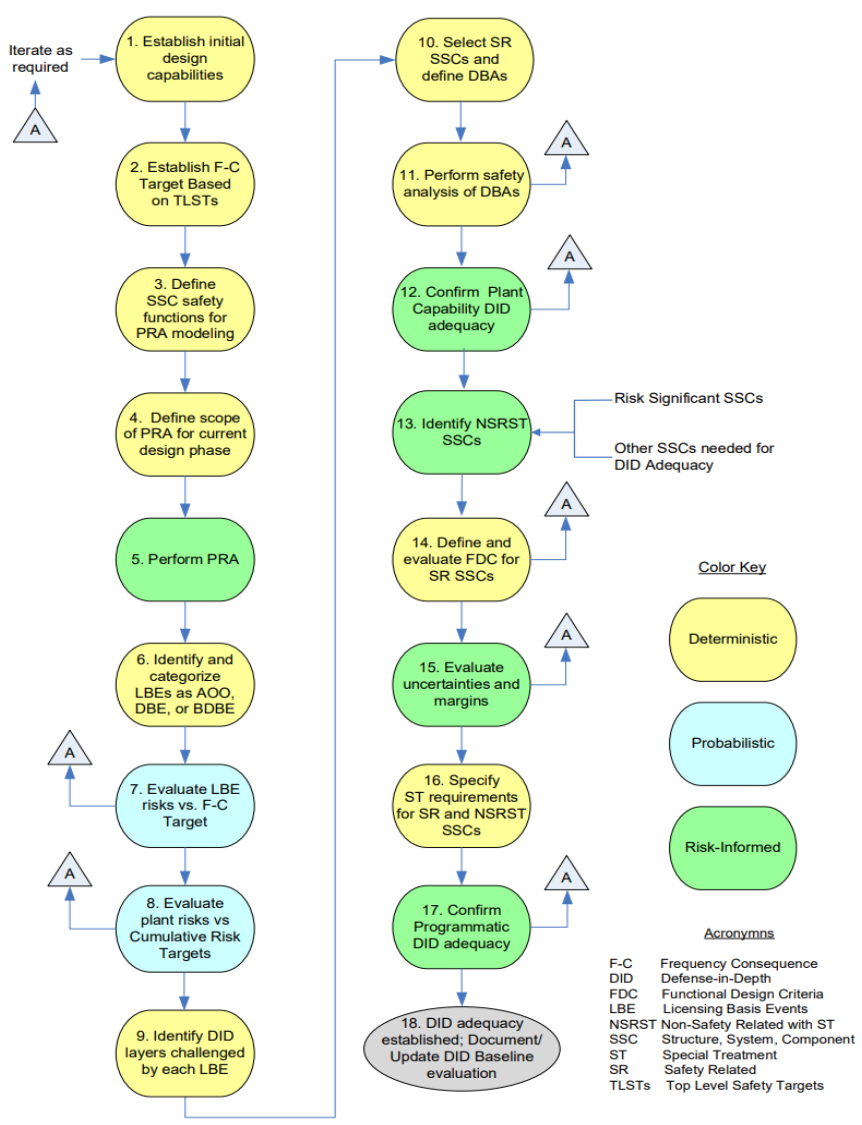

Figure B-3 Integrated Process for Incorporation and Evaluation of DID