Non-Light Water Review Strategy

Staff White Paper

September 2019

This draft staff white paper has been prepared and is being released to support ongoing public discussions

This paper has not been subject to NRC management and legal reviews and approvals, and its contents should not be interpreted as official agency positions.

Non-Light-Water Review Strategy

Table of Contents

1 Introduction

1.1 Background

The U.S. Nuclear Regulatory Commission (NRC) is responsible for reviewing applications for permits, licenses, certifications, and approvals for the design, construction, and operation of advanced reactor technologies. In 2016, the staff developed a vision and strategy to assure that the NRC is ready to review potential applications for non-light water reactor (non-LWR) technologies effectively and efficiently. The staff described this vision and strategy in "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness" [1].

To achieve the goals and objectives stated in the NRC's vision and strategy, NRC staff developed implementation action plans (IAPs), issued in July 2017. The IAPs identify the specific activities the NRC will conduct in the near-term (within 5 years) [2], mid-term (5-10 years) [3], and long-term (beyond 10 years) timeframes [3]. The near-term IAP identifies six strategies to help achieve success in the review of applications for permits, licenses, certifications, and approvals in the near term. NRC staff has ongoing activities in all six strategies. The staff reports on the status of the vision and strategy and IAPs in an annual SECY paper, SECY-19-009, "Advanced Reactor Program Status" [4]. The focus of this Advanced Reactor Review Strategy is on implementation of IAP Strategy 3, which is to develop NRC staff guidance for an adaptable, technology-inclusive, risk-informed, and performance-based (TI-RIPB) approach to licensing advanced reactors.

NRC staff and external stakeholders collaborate on various activities under IAP Strategy 3. Since 2017, the staff has engaged on the Licensing Modernization Project (LMP) being led by Southern Company, coordinated by the Nuclear Energy Institute (NEI), and cost-shared by the U.S. Department of Energy (DOE). The NRC staff published draft regulatory guide (DG) 1353, "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," [5] in the Federal Register on May 3, 2019, for public comment. This DG endorses, with clarifications, the principles and methodology in NEI 18-04, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," [6] as one acceptable method for determining the appropriate scope and level of detail for parts of applications for licenses, certifications, and approvals for non-LWRs[1]. The NRC staff also drafted a SECY paper requesting the Commission's review and approval of the overall approach described in NEI 18-04 and DG-1353. Ultimate implementation and finalization of the DG as a regulatory guide (RG) will be informed by Commission direction received on the SECY paper.

Additionally, in January 2019, the U.S. Congress enacted the Nuclear Energy Innovation and Modernization Act (NEIMA) [7], which in part directs NRC to complete a rulemaking that establishes a TI-RIPB regulatory framework by 2027. NRC staff is working to develop a rulemaking plan and schedule for this undertaking.

[1] The scope of the LMP is specifically for non-LWRs. The NEIMA broadly defines ''advanced nuclear reactor'' as a nuclear fission or fusion reactor. For the purposes of this document, advanced reactor includes a non-LWR. Both terms are used as appropriate throughout this document.

The Non-Light-Water Reactor Review Strategy provides NRC staff guidance for reviewing non-LWR applications submitted before 2027. It is intended to complement these and other IAP Strategy 3 efforts to describe approaches to effectively evaluate licensing basis information provided in applications supporting advanced reactor technologies that may differ in scope, organization, and level of detail from other applications for other reactor technologies.

1.2 Purpose of the Non-Light-Water Reactor Review Strategy

Based on engagement with prospective applicants, the NRC staff is preparing for the submission of non-LWR applications. This document has been prepared by NRC staff to support the near-term reviews of applications for non-LWR designs that are submitted prior to the development of the technology-inclusive, risk-informed, and performance-based regulatory framework in 2027, as required by NEIMA.

Several vendors have indicated that they plan to implement LMP to develop the licensing basis for their applications. Additionally, although NEI 18-04 has not been endorsed by the Commission, concepts in the LMP may inform the development of the TI-RIPB regulatory framework. As such, processes from the LMP are referenced throughout this document. However, applicants are not required to utilize LMP and may instead use another methodology (e.g., traditional deterministic approach, maximum hypothetical accident[2]) to analyze non-LWR performance and develop a licensing basis. The Non-Light-Water Reactor Review Strategy provides NRC staff an approach to reviewing the licensing basis information of a non-LWR application independent of the specific design or methodology used. As such, this review guide is limited in scope. NRC staff should continue to consult other established guidance documents, as applicable, to complete reviews of non-LWR applications. Specific topics addressed in this review guide are:

<u>Section 2.0 Non-LWR Vendor Approaches to Developing the Licensing Basis</u> – This section describes the various approaches to developing the licensing basis for non-LWRs designs. This section also provides background and references for pre-application interactions, contents of applications, and development of safety evaluation reports.

<u>Section 3.0 Acceptance Criteria and NRC Staff Review Approach</u> – This section describes the scope and focus of the staff's technical review and discusses, in general terms, the acceptance criteria that could be considered by NRC staff during the technical review of a non-LWR application. This section also provides guidance for the analysis and evaluation of the integrated system design and provides expectations for probabilistic risk assessments for non-LWRs.

<u>Section 4.0 References</u>

<u>Section 5.0 Acronyms</u>

<u>Attachment 1 - Analysis of Applicability of NRC Regulations for Non-LWRs</u> – This attachment provides a list of current NRC regulations applicable to LWRs and the NRC staff's expectations for the applicability of these requirements to anticipated non-LWR designs based on technical

[2] In this context, "maximum hypothetical accident" refers to a conservatively assessed, deterministic accident with consequences that bound the full spectrum of accident conditions for the plant and is not necessarily a credible event.

and design information currently available. Prospective applicants should engage with the NRC staff to determine the specific applicability of requirements to a particular design or technology.

2 Non-LWR Vendor Approaches to Developing the Licensing Basis

2.1 Background

Non-LWR vendors may use a variety of approaches to develop their design. The approach chosen will impact the licensing strategy. In order to assist non-LWR vendors, the NRC developed its "Regulatory Review Roadmap for Non-Light Water Reactors" [8] which discusses the variety of options the NRC has available for performing regulatory reviews of non-LWR designs. These options include the more formal reviews of final designs in an application for a permit, license, certification, or approval, as well as the less formal reviews of pre-application information. Understanding what options are available and how to choose the best option may be challenging for a designer, especially one that is less familiar with the NRC's regulatory framework and associated review processes. Ultimately, the design strategy and licensing approach selected by a non-LWR vendor will inform the scope, organization, and level of detail necessary within an application. Consequently, the scope, organization, and the level of detail, including the amount of supporting documentation needed to support findings, of the NRC staff's safety evaluation report (SER) will be informed by the presentation of information in the application.

2.2 Pre-Application Interactions

Pre-application interactions are an important tool for the NRC staff to plan reviews of non-LWR designs. While voluntary, pre-application interactions are encouraged by the Commission as part of the NRC "Policy Statement on Advanced Reactors" [9]. These interactions with prospective applicants may be initiated once a prospective applicant has indicated sufficient commercial intent, organizational capacity, design maturity, and expectation of an application submittal to support commencement of meaningful regulatory discussions with NRC staff. Prospective applicants may indicate intent to prepare and submit an application by submitting general design information, licensing plans, and a schedule for submitting documents for NRC review in either a voluntary letter or a response to a Regulatory Information Summary (RIS) (e.g., RIS 17-08, "Process for Scheduling and Allocating Resources for Fiscal Years 2020 through 2022 for the Review of New Licensing Applications for Light-Water Reactors and Non-Light-Water Reactors," dated December 21, 2017 [10]). The major factors that affect the effectiveness of pre-application activities include the maturity of the design and associated analyses, and the timing and frequency of a prospective applicant's engagement with the NRC prior to submitting an application.

Early pre-application engagement may provide the NRC staff insight on how to best prepare to efficiently and effectively review a submittal. For example, white papers, topical reports, technical reports, or other types of documents may be submitted to the NRC for various levels of review during the pre-application period. Documents such as these will assist the NRC staff in understanding the design as early as possible and could provide the applicant with formal regulatory finality or informal NRC staff feedback on technical, policy, and licensing topics. Some key areas related to the licensing basis of the facility that could be addressed during pre-application interactions include:

- Basic understanding of the technology and general design

- Radionuclide inventories and key parameters (e.g., burnup, temperature, pressure, etc.) related to release fractions
- Operating conditions and safety functions needed to maintain key parameters
- Possible threats to operating conditions and safety functions
- Principal design criteria and applicable or non-applicable regulatory requirements, including those requiring an exemption
- Novel design (e.g., fuel qualification) or policy (e.g., emergency planning zones) issues
- Mutual understanding of the decision-making criteria and the integrated approach used for design/licensing
- Methodologies, analytical methods, and validation bases for demonstrating the safety case
- Risk insights from a probabilistic risk assessment (PRA) including risk-significant functions; licensing basis events (LBEs); systems, structures, and components (SSCs); and calculated risk estimates

Pre-application submittals such as white papers, topical reports, technical reports, or other types of documents allow the NRC staff to plan for the review of the application. For example, early availability of preliminary PRA results will assist the NRC staff in understanding the overall plant risk including anticipated LBEs, the preliminary safety/risk categorization strategy for the SSCs, and approaches to defense-in-depth (DID). If the applicant intends to use innovative, or novel, design features (such as passive systems, reliance on inherent features, or simplified control features), early identification of these features to the NRC will facilitate timely identification and resolution of any unique regulatory topics.

2.3 Overview of Design Strategies

An applicant may consider different design strategies in the development of its application. Selected strategies will impact the level of pre-application engagement and information contained in submittals to the NRC. The staff does not require any specific strategy or approach to developing information for an application. However, regardless of the strategy used, an applicant will need to use a well-defined process to develop its safety case and licensing basis.

2.3.1 Licensing Modernization

The LMP's objective is to develop a TI-RIPB approach for the selection of LBEs; the classification of SSCs and associated special treatments; and the determination of DID adequacy for non-LWRs. The approach described in NEI 18-04 and proposed to be endorsed in DG-1353 is consistent with the Commission's staff requirements memorandum (SRM) approving the recommendation in SRM-SECY-03-0047, "Policy Issues Related to Licensing Non-Light-Water Reactor Designs," [11] to allow the use of a probabilistic approach to identify events provided there is sufficient understanding of plant and fuel performance, and to use engineering judgment to address uncertainties. The approach described in NEI 18-04 focuses on safety functions and the identification of the SSCs needed to fulfill those functions. Frequency-consequence (F-C) targets define the SSC capabilities and reliabilities required. The LMP is an iterative process involving assessments and decisions on key SSCs; operating parameters; and programmatic controls to ensure a reactor can be deployed with no undue risk to public health and safety. As written LMP is only applicable for non-LWR designs, and not advanced reactor designs in general (see footnote 1).

2.3.2 Other Approaches

Applicants may choose to follow an alternate approach for determining LBEs and classifying SSCs. This approach could be based on a maximum hypothetical accident, similar to what is described in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," [12], resemble a more traditional deterministic approach, or some other approach, as long as an applicant provides an adequate justification. Regardless of the approach used by the applicant, the staff expects the submittal to identify and evaluate LBEs and classify SSCs with a clear basis.

This LBE selection will be reviewed by the staff, and ultimately will be used to verify the classification SSCs. The NRC review scope for SSCs, as described in Section 3 of this document, is largely predicated on an acceptable LBE selection. As such, a thorough understanding of the plant risks (e.g., as provided through a robust PRA) coupled with early staff engagement is important in achieving an efficient and effective review.

2.4 Informing the Content of Applications

The development of a safety analysis report begins with documenting the basic reactor physical and operating characteristics, including, but not limited to:

- specific technology
- power level
- materials
- moderator
- coolant
- neutron energy spectrum
- thermodynamic cycle including parameters of the cycle and energy balance
- fuel type
- passive versus active safety systems
- selection of design codes for major SSCs
- operations and maintenance philosophy
- other high-level design decisions driven by the top-level requirements

The foundational information for a safety analysis report also includes a comprehensive set of plant-level and system-level functional requirements.

Due to the variety of non-LWR designs being developed, the staff finds it more appropriate to define a technology-inclusive methodology for non-LWRs than to develop prescriptive content guidance as was developed for LWRs and documented in RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," [13]; and RG 1.206, "Applications for Nuclear Power Plants" [14]. Currently, the staff is starting to work with industry on Phase 2 of the LMP, the Technology-Inclusive Content of an Application Project (TICAP). The purpose of this effort is to develop a methodology for using the output from NEI 18-04 to develop an application (e.g. safety analysis report) that contains a level of detail that is commensurate with the risk significance of the SSCs. This methodology can be used by non-LWR applicants in preparing the content of an application, which is usually documented in preliminary or final safety analysis reports.

Consistent with these efforts, designers may choose to organize the information in a safety analysis report in various formats. The general methodology to determine the appropriate content and level of detail for safety functions, SSCs, and programmatic controls remains valid no matter how the information is organized within a safety analysis report.

For example, a non-LWR vendor will need to provide information to address controls and barriers that can prevent and mitigate damage to a reactor that would result in fission product releases from threats and events. This Non-Light-Water Reactor Review Strategy focuses primarily on the controls and barriers related to Plant Internal Events and External Events. The "bow-tie" diagram in Figure 1 provides an example of how controls and barriers can prevent and mitigate damage to a reactor that would result in fission product releases from threats and events. In the case of fission product migration due to a threat/event, the diagram also shows how controls and barriers can be used to mitigate (or recover) to prevent consequences.

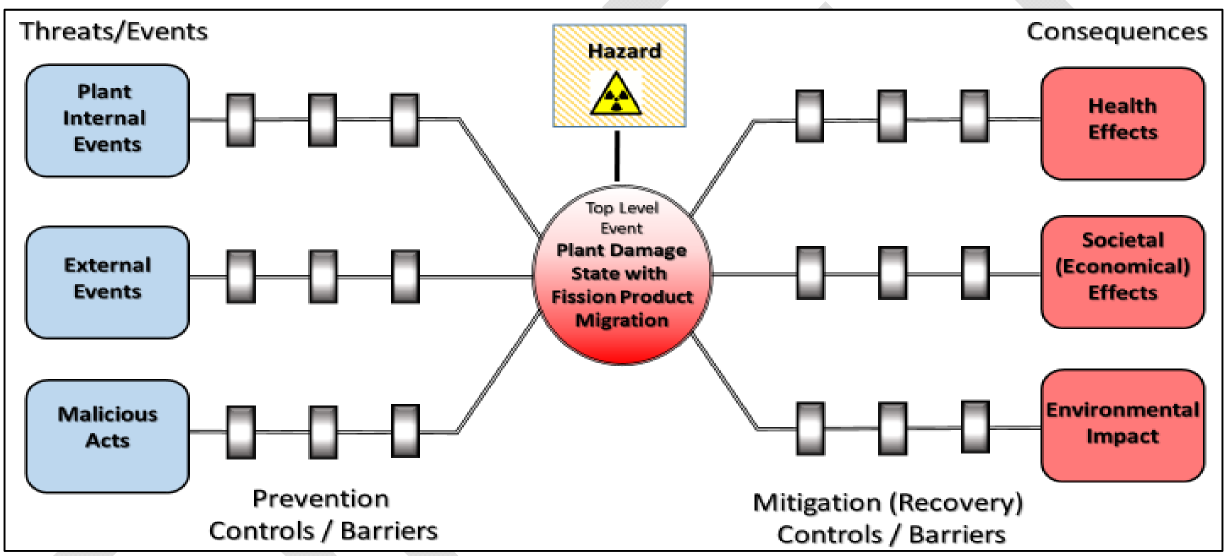

Figure 1: "Bow-tie" diagram

Figure 2 provides an example of NRC and industry initiatives related to addressing technical areas associated with prevention and mitigation.

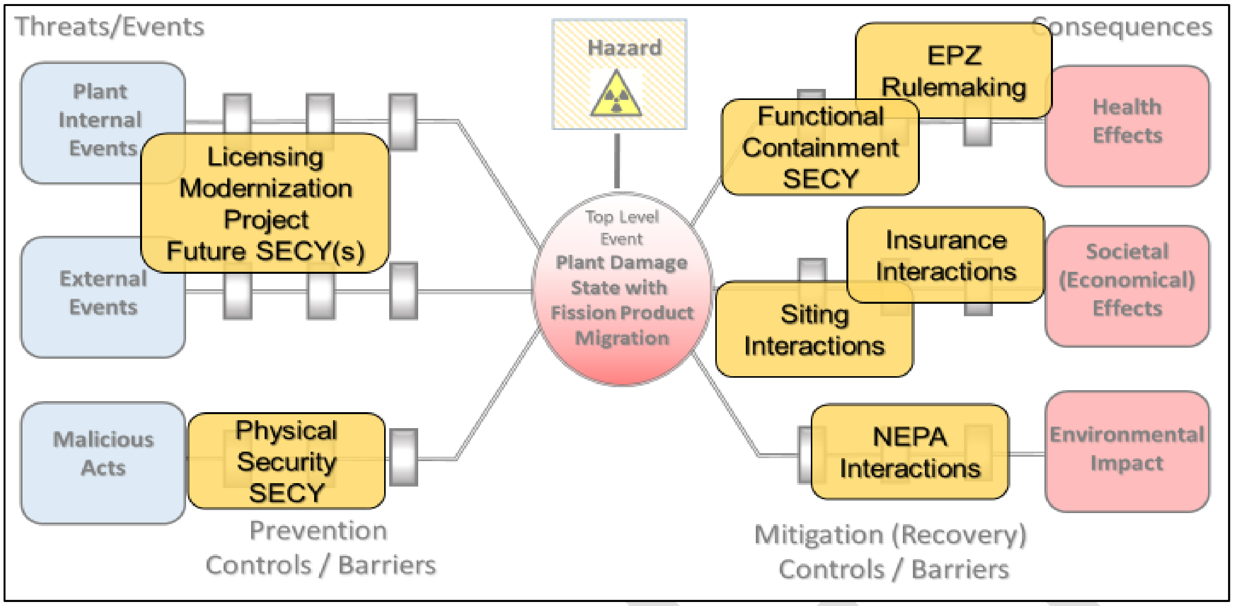

Figure 2: "Bow-tie" diagram with examples of NRC and industry initiatives.

The remainder of Section 2 of this document describes some of the technical areas that need to be addressed in an application to evaluate LBEs and verify the classification of SSCs. For ease of discussion, the following descriptions refer to the traditional chapter-level format of RG 1.206.

2.4.1 Basic Reactor Characteristics Application Information

Many of the basic reactor characteristics have traditionally been described in Chapters 4, 5, and 6 of safety analysis reports of applicants following the guidance in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," [15] (SRP). These chapters address the reactor, including fuel and reactivity control systems, the reactor coolant and connecting systems, backup cooling systems, and functional barriers for retaining radionuclides within the facility. The information in these chapters largely addresses the fundamental safety functions[3] of a design (e.g., reactivity control, core heat removal, radioactive material retention). Other information to be provided includes the descriptions of the fuel or fuel system boundary and primary system in terms of the limits on operation (e.g., values or ranges of values for key parameters) to prevent failures or degradation, or to remain within the bounds of testing or qualification of related SSCs. These limits on operation thus establish the safety functions needed to prevent damage to barriers to the release of radionuclides (e.g., functions to maintain integrity of fuel cladding, coatings, or other fuel system boundary). This information is needed for the development of accident source terms for individual non-LWR designs.

2.4.2 Licensing Basis Events Application Information

The analysis of anticipated operational occurrences (AOOs), design basis events (DBEs), and beyond DBEs (BDBEs) plays an important role in defining safety functions, classifying SSCs,

[3] Fundamental safety functions are those safety functions common to all reactor technologies and designs. This conforms with the internationally recognized definition in IAEA SSR-2/1, "Safety of Nuclear Power Plants: Design," [16]

and evaluating DID adequacy. The safety analysis report describes the analysis results for event sequences and related organization into event sequence families. Applicants following the SRP may present the results from a PRA in Chapter 19 of the safety analysis report or in a new section added to Chapter 15. Alternatively, an application may develop a new chapter to include the analysis of AOOs, DBEs, and BDBEs using the selected methodology.

Deterministic evaluations and PRA inform design decisions and ultimately support the safety arguments presented in applications for permits, licenses, certifications, and approvals. Any interrelationships between the LBEs and the derivation of both plant capabilities and programmatic controls need to be reflected in the layout of the safety analysis report. In addition to plant response information on SSC capabilities typically provided in deterministic evaluations, the description of AOOs, DBEs, and BDBEs needs to include or point to key information such as uncertainties and measures to ensure assumed SSC capabilities and reliabilities. Regardless of the method chosen to determine LBEs and classify SSCs, the staff expects the applicant to discuss how the events selected were arrived at and what SSCs are required to prevent and mitigate against these events. The staff will then review the methodology used to determine the LBEs and classify SSCs using engineering judgment and available resources (e.g., published standards or articles) if the methodology used has not been previously approved by the staff.

Deterministic evaluations have typically been described in in Chapter 15 of safety analysis reports prepared using the SRP. Addressing design basis accidents (DBAs) in a separate section or chapter from the other LBEs could support maintaining the distinction between the deterministic analyses that credit only safety-related SSCs and the assessments of the remaining LBEs. A separate chapter might also help with the development of Technical Specifications and other elements of the licensing-basis documentation that are traditionally related to safety-related SSCs. Descriptions related to the derivation of design basis external hazards and protection of safety-related SSCs from design-basis external hazards are usually provided in Chapters 2 and 3 of safety analysis reports prepared using the SRP.

2.4.3 Structures, Systems, and Components

Safety analysis reports for operating LWRs include chapters that contain detailed descriptions of SSCs that support safety functions. Examples include chapters on instrumentation and control systems, electrical power systems, and cooling water systems. Additional chapters in LWR safety analysis reports are dedicated to power conversion systems. The various system descriptions for LWRs are appropriate, given the importance of support systems for active safety systems and the potential for support or secondary plant systems to cause a plant transient that challenges the fuel cladding or other barrier to the release of radionuclides.

For a non-LWR applicant, a description of ancillary plant systems or the interface between the ancillary and primary facility systems should focus on any safety functions being supported and possible contributions to initiating events. This should include information describing how the failure of the ancillary plant systems impact any SSC performing a necessary safety function[4]. These analyses and assessments should provide insights into the appropriate level of detail

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[4] Necessary safety functions are those functions that must be accomplished to ensure that the facility is operated without undue risk to public health and safety.

needed to describe parts of a facility outside the primary systems, which are typically described in Chapters 7-12 of a safety analysis report prepared using the SRP. In some instances, the level of detail necessary in the safety analysis report for ancillary plant systems in non-LWR designs may be significantly less than that provided for LWRs due to the use of passive safety systems, reliance on inherent features, and increased thermal capacities of reactor systems, which reduce sensitivities to plant upsets. Other appropriate information to be included in an application includes SSC classification and any special treatments identified to address the safety or risk significance of the ancillary SSCs identified via insights from the PRA or evaluation of DID adequacy.

The level of detail for ancillary SSCs may also reflect potential performance-based approaches within applications for permits, licenses, certifications, or approvals. Guidance for NRC staff reviews of non-LWRs encourages the staff to consider performance-based approaches, which can likewise be used to inform the appropriate level of detail in applications. "Introduction - Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Small Modular Reactor Edition," [17] includes the following guidance on the use of performance-based approaches as part of an integrated review for small modular reactors:

> Second, the framework incorporates an integrated review approach by using the satisfaction of selected requirements to provide reasonable assurance of some aspects of SSC performance (for example, performance-based acceptance criteria related to SSC capability, reliability, and availability). Examples of requirements that could be applied for this purpose include 10 CFR Part 50, Appendix A (general design criteria, overall requirements, criteria 1 through 5), 10 CFR Part 50, Appendix B (quality assurance program), 10 CFR 50.49 (electric equipment environmental qualification program), 10 CFR 50.55a (code design, in-service testing and in-service inspection programs), 10 CFR 50.65 (maintenance rule), Technical Specifications (TSs), Availability Controls for SSCs subject to Regulatory Treatment of Non-Safety Systems (RTNSS), the Initial Test Program (ITP), and ITAAC. In preparing the safety evaluation for the application, the staff may use the satisfaction of these selected requirements to augment or replace, as appropriate, technical analysis and other evaluation techniques to obtain reasonable assurance that the performance-based acceptance criteria are satisfied. Under the framework, the staff also has the flexibility to use these selected requirements to demonstrate satisfaction of design-based acceptance criteria for the SSCs with low risk significance. The staff will verify the demonstration of the design-basis capabilities of SSCs that are important to safety as part of the ITAAC completion review prior to plant operation.

2.4.4 Defense in Depth

Current guidance for safety analysis report format and content for LWRs (e.g., RG 1.206) does not include a specific section for evaluating DID adequacy. However, the importance of a DID adequacy evaluation may justify a separate section or a chapter in safety analysis report (e.g., the addition of a chapter titled, "Evaluations of Defense-in-Depth"). The format and content of the new chapter could follow the systematic assessment methodology described in NEI 18-04 and document integrated decision-making process panel (IDPP) decisions. For applicants that choose not to use NEI 18-04, an application chapter on DID may or may not be needed depending on the approaches taken. For a traditional deterministic approach, for example, DID is not needed consistent with RG 1.70 and RG 1.206. For this approach, DID is built into the collection of individual chapters (e.g., using layers of defense).

2.4.5 Other Application Information

As shown in Figure 2, the evaluation of LBEs, classification of SSCs, and identification of controls and barriers credited to prevent and mitigate damage to a reactor may impact the analysis of other technical areas such as emergency planning, physical protection, siting, and containment. It is up to the applicant to determine the best method to convey how this information supports the safety case to the NRC. For example, some of this information may be presented in separate chapters of an application or provided in supporting documents such as Topical Reports or Technical Reports.

An application for a non-LWR may also include a section or chapter addressing the applicability and implementation of regulatory requirements, including any planned or requested deviations or exemptions.

2.5 Development of Safety Evaluation Reports

The NRC staff documents its review of applications in SERs. The objective of the staff's SER is to explain how the staff reached a reasonable assurance finding that the proposed actions associated with the application will not pose undue risk to the health and safety of the public. The reasonable assurance finding supports the Commission's considerations in determining whether a construction permit or operating license under 10 CFR part 50, or early site permit, combined license, or manufacturing license under 10 CFR part 52 will be issued to an applicant, as described in 10 CFR 50.40, "Common standards." As a risk-informed regulator considering risk insights (i.e., results and findings from a risk assessment) as well as other principles (e.g., DID and safety margin), the NRC staff's review scope and level-of-detail in the SER should generally be commensurate with risk or safety significance of the technology under evaluation. As such, the staff's SER should reflect how the staff has used risk insights in determining the scope and depth of the review.

As a modern, risk-informed regulator, the NRC staff has initiated an effort to provide NRC staff with high-level expectations and guidance on developing high-quality SERs to improve the safety focus, efficiency, and effectiveness of these documents. There is an effort underway to provide NRC staff with high-level expectations on developing high-quality SERs. To document its review of non-LWR applications, the NRC staff is expected to follow the expectations in the following draft document, "Expectations for Safety Evaluation Reports" [18].[5]

Section 3 of this Non-Light-Water Reactor Review Strategy describes how the staff would review a submittal. The level of detail in the staff's SER will be based on the content of the application and safety significance of the review areas. As such, the level of review and documentation necessary for the staff to make its safety findings for an application may vary.

[5] The ongoing effort in this area may result in a future Office Instruction or guidance document.

3 Review of Submittal (Implementation of LMP or Other Design Strategy)

The NRC staff's review of an application for a permit, license, certification, or approval facilitates a finding of reasonable assurance of adequate protection by evaluating the information provided by the applicant against the Commission's regulations. Use of the processes described in the LMP (i.e., NEI 18-04) by an applicant can help facilitate a risk-informed, performance-based licensing process; however, use of the LMP is not required. Non-LWR applicants may choose to follow an alternate approach to LBE selection and SSC classification, such as a maximum hypothetical accident analysis, a more traditional deterministic approach, or some other approach, given an applicant provides an adequate justification.

In review areas that are not technology-dependent, the staff will likely utilize existing guidance to assist in the performance of the review. Discussions in this document mainly pertain to how the staff reviews design-related regulatory requirements (i.e., those in 10 CFR 50.34 and associated requirements). Based on the importance of assessing overall plant risk, NRC staff review will focus on the following required content areas of an application, as applicable:

- Identification and bases for the principal design criteria (PDC) of the facility (see Section 3.1)
- Selection of LBEs (see Section 3.2)
- SSC classification, including consideration of DID (see Section 3.3)
- Design bases of SSCs and their relationship to the PDC (see Section 3.4)
- Analysis and evaluation of the integrated design and performance of SSCs (see Section 3.5), including:
    - determination of margins during normal operations and AOOs,
    - adequacy of safety-related SSCs for accident mitigation, and
    - adequacy of plant response to BDBEs
- PRA (see Section 3.6)
- Other requirements including siting, emergency preparedness, physical security, etc. (see Attachment 1 – Analysis of Applicability of NRC Regulations for Non-Light Water Reactors).

The content identified in the list above conforms to regulatory requirements. Other specific elements reviewed by the staff that are implicit in justifying aspects of the above list and discussed in the section to follow include:

- Analytical methods and associated testing/validation
- Consequence assessment and source terms
- Necessary safety functions
- Consideration of DID
- Any special treatment associated with a particular SSC needed to augment design basis requirements for that SSC

Risk information provided in the licensing application and during pre-application interactions will inform the scope and depth of the staff's review and establish the acceptance criteria for SSCs credited in fulfilling necessary safety functions, performing risk-significant functions, or providing DID.

A review of the content described in this section of the Non-Light-Water Reactor Review Strategy is expected to be highly integrated due to the inter-dependencies among the subject areas. A discussion of these review dependencies is included in the sections below.

In addition to considerations for LBE selection and SSC classification, the applicants should evaluate the applicability of the NRC's regulations to specific designs. The NRC staff recognizes that the applicability of certain regulations developed for LWRs may need to be analyzed to determine the appropriates for non-LWRs. As such, the staff has prepared "Attachment 1 - Analysis of Applicability of NRC Regulations for Non-Light Water Reactors," which presents the NRC staff's analysis of those regulations anticipated to be generally applicable to all non-LWR applicants. Table 1 of this attachment provides additional context for areas where exemptions may be expected for non-LWR designs. Tables 2, 3, and 4 present a non-exhaustive list of regulations to be considered by non-LWR designers, with expected applicability for each regulation in the table. This attachment and the tables contained within do not constitute an exhaustive set of applicable regulations and potential exemptions. Applicants may request exemptions from the NRC's regulations on a case-by-case basis depending on the nature of the specific design and independent determination of regulatory applicability.

3.1 Principal Design Criteria

10 CFR 50.34(a)(3)(i), 10 CFR 52.47(a)(3)(i), 10 CFR 52.79(a)(4)(i), 10 CFR 52.137(a)(3)(i), and 10 CFR 52.157(a) require that proposed PDC be included in an application for a construction permit (CP), design certification (DC), combined license (COL), standard design approval (SDA), or manufacturing license (ML). The PDC establish the necessary design, fabrication, construction, testing, and performance of safety-significant[6] SSCs. When applicable, the staff will review the applicant's PDC against the guidance provided in RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors" [19]. Initial staff review will focus on the PDC to determine whether they are the appropriate set of PDC for the design using RG 1.232 and the GDC to determine whether the proposed PDC adequately capture the necessary safety requirements for the design. The staff will review the applicant's PDC to ensure that the necessary safety functions and DID considerations are adequately addressed.

The NRC staff expects prospective non-LWR applicants will review the GDC pertaining to LWRs provided in Appendix A to 10 CFR Part 50 and the guidance in RG 1.232 to develop their PDC and ensure that necessary safety functions and SSCs are covered under the selected PDC. In each case, it is the responsibility of the designer or applicant to provide the PDC for the design, supporting information that justifies to the NRC how the design meets the PDC submitted, and how the PDC provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. In addition, the designer or applicant needs to consider and justify if additional PDC must be identified in the interest of public safety. For example, additional PDCs may be needed due to the use of passive or inherent design features.

The review of PDC has the following review interfaces:

[6] In this context, safety-significant SSCs are safety-related SSCs as classified by the applicant or designer, SSCs that meet the definition of safety-related in 10 CFR 50.2 as applicable to the design, or those SSCs are classified as risk significant (those that perform functions that are risk-significant) or required to support DID considerations.

- The PDC establish high-level requirements for the design basis of SSCs (Design Basis of SSCs is discussed in Section 3.4)
- The PDC establish acceptance criteria used in the evaluation of the integrated design (Analysis and Evaluation of Integrated Design is discussed in Section 3.5)
- PDC development is informed by risk-insights that may be obtained through a PRA (PRA is discussed in Section 3.6).

3.2 Selection of Licensing Basis Events

LBEs constitute the entire collection of event sequences identified in the design and licensing basis of the plant, which may include one or more reactor modules. LBEs include normal operation, AOOs, DBEs, DBAs, and BDBEs. The NRC staff recognizes that there are other processes that could be used to select the set of LBEs and classify SSCs in accordance with the requirements of 10 CFR 50.34 (or other like requirements in 10 CFR Part 52). For instance, an applicant could propose a maximum hypothetical accident with SSC classification based off that event. Regardless if an applicant uses NEI 18-04 or a different process, the NRC staff will review the methodology used by the applicant to select the LBEs and classify SSCs.

The review of LBE selection has the following review interfaces:

- LBE selection determines which event scenarios are evaluated and the associated acceptance criteria in the evaluation of integrated plant design (Analysis and Evaluation of Integrated Design is discussed in Section 3.5)
- LBE selection may be informed by the event sequence frequency determined from a PRA (PRA is discussed in Section 3.6)

3.3 Classification of SSCs

Classification of SSCs is based on the role performed by the SSCs in preventing or mitigating against the LBEs and their consequences. SSCs required to prevent or mitigate against DBAs and their consequences and meet the acceptance criteria (as applicable to the event in question) should be classified as safety-related. This document uses two additional categories of SSCs, non-safety-related[7] with special treatment (NSRST) and non-safety-related, although other approaches (such as that defined in 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors") may be used. For the purposes discussed here, NSRST SSCs are those that perform risk-significant functions or functions required to support DID adequacy beyond those deemed safety-related. The NRC staff will review the SSC classification process based on the roles performed by the SSCs.

The review of SSC classification has the following review interfaces:

- SSC classification is informed by the LBE selection process (LBE Selection is discussed in Section 3.2)
- SSC classification informs design considerations (e.g., augmented quality) in the design basis of the SSC (Design Basis of SSC is discussed in Section 3.4)

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[7] In the context of this document, the term "non-safety-related" matches the classifications used in DG-1353 and NEI 18-04; that is, "non-safety-related" means SSCs not relied on to perform a safety function (i.e., the total set of SSCs, excluding those classified as safety-related).

- SSC classification informs the treatment of the associated systems in the evaluation of the integrated system design (Analysis and Evaluation of Integrated Design is discussed in Section 3.5)
- SSC classification is informed by the risk profile, which could be established by a PRA (PRA is discussed in Section 3.6)

3.4 Design Basis of SSCs

The NRC staff uses various guidance documents to perform safety reviews of LWRs subject to the requirements of 10 CFR Parts 50 and 52. The primary goal of existing NRC LWR guidance, such as the SRP, is to assure the quality and uniformity of staff LWR safety reviews and increase the transparency of the NRC's regulatory processes so that stakeholders and the public understand NRC review process. Early LWR designs were licensed prior to the establishment of the GDC and SRP. Experience from these reviews informed the development of the GDC and SRP. Similarly, it is expected that the reviews of the first non-LWR technologies will inform the development of future licensing and guidance development. However, to the extent practicable, the NRC staff will leverage applicable licensing experience and engineering and analysis tools to augment of its reviews of non-LWR applications.

The review of the design basis of SSCs has the following review interfaces:

- The design basis of SSCs provides the information describing the SSC performance and classification used in the evaluation of integrated facility design (Analysis and Evaluation of Integrated Design is discussed in Section 3.5)
- The design basis of SSCs provides the information describing the SSC performance and reliability, which may be used as input in a PRA (PRA is discussed in Section 3.6)
- The high-level requirements for the design basis of SSCs are provided by the PDC (PDC are discussed in Section 3.1)
- Design considerations to be included in the design basis of SSCs, and level of detail expected in the SSC description, is informed by SSC classification (SSC classification is discussed in Section 3.3)

3.4.1 Introduction

The design process and related development of licensing-basis information for LWRs are iterative processes involving assessments and decisions on key SSCs; operating parameters; and programmatic controls to ensure a reactor can be operated with no undue risk to public health and safety. This risk arises from the potential release of radioactive materials during normal operation and plant upset conditions. The approach proposed in this Non-Light-Water Reactor Review Strategy does not change NRC requirements for reactor applications or applicants but strives to more clearly define and focus the review of the facility licensing basis on the most risk-significant areas.

One measure of radiological risk to public health and safety can be represented in terms of the consequences – that is, the inventory of radioactive materials and the fraction of that inventory that might be released as a result of a given event. Traditionally, the approach used to assess the potential consequences posed by LWRs has been to select several well understood stylized events to define requirements for SSCs serving as barriers to the release of radioactive materials and protecting such barriers by controlling reactor heat generation and reactivity and providing cooling. These safety functions remain an important consideration for all nuclear power reactors, including non-LWRs, although the requirements for the SSCs that address the safety functions and how they factor into the ultimate goal of controlling the release of radioactive material will vary depending on the facility design.

Additional requirements for LWRs have been identified as a result of operating experience and insights from PRAs. Most of the NRC's requirements, guidance documents, studies, and other activities related to nuclear power plants have focused on LWR technologies and specific design attributes and behaviors related to water coolant, zirconium alloy fuel cladding, and other characteristics of LWRs. These LWR-centric requirements and approaches to regulatory reviews do not readily translate to a licensing framework for non-LWR technologies, which use different coolants, fuel forms, and safety system designs.

Nevertheless, non-LWR reactor technologies are still required to meet the top-level requirements defined in 10 CFR Part 50 for production and utilization facilities, one of which is a dose requirement for nuclear power reactors. Non-LWRs will still need to demonstrate that radioactive materials are retained, through some combination of barriers, control of heat generation and reactivity, and removal of heat.

As described in the Commission's Policy Statement on the Regulation of Advanced Reactors, published in the Federal Register (73 FR 60612) on October 14, 2008, advanced reactors are expected to have enhanced margins of safety, use simplified safety systems, provide increased thermal margins, and demonstrate other attributes associated with advanced reactor technologies. Incorporating these attributes into a non-LWR design may promote more efficient and effective design reviews and define a level of detail for applications commensurate with the risks posed by the non-LWR design. Applications will still need to address those features and programmatic controls necessary to retain radionuclides and to protect barriers from events and related plant conditions. Applications and NRC staff reviews will continue to need information describing the fuel or fuel system boundary, primary system, and other barriers in terms of the limits on operation (e.g., values or ranges of values for key parameters) to prevent failures or degradation, or to remain within the bounds of testing or qualification of related SSCs. These limits on operation will, in turn, establish the needed safety functions to prevent damage to barriers to the release of radionuclides (e.g., functions to maintain the integrity of fuel cladding, coatings, or other fuel system boundaries).

The approach outlined in DG-1353 focuses on a risk-informed, performance-based framework, and is predicated on the designer implementing a PRA of sufficient detail that captures the important aspects of the design. Other approaches that may be used could focus on bounding consequence analysis or holistic consideration of event likelihood and consequences; regardless of the approach used, the process chosen by the applicant should consider risk information appropriately to inform the application. In order to focus the licensing review on the appropriate design features used to mitigate radiological releases, detailed early communications with the NRC regarding the design, especially regarding the PRA, and LBE selection, and SSC classification, will be important to leverage the risk information to scope the review appropriately.

3.4.2 Objectives

The Non-Light-Water Reactor Review Strategy provides NRC staff an approach to reviewing the licensing basis information of a non-LWR application independent of the specific design or methodology used. The following sections provide high-level staff acceptance criteria and establish general expectations for submittals. More specific acceptance criteria could be established for a particular design as part of pre-application interactions with vendors prior to submission of an application.

This review guide describes a graded approach to reviewing SSCs, providing different approaches for reviewing SSCs that are classified as 1) safety-related, 2) non-safety-related, but important for DID or risk measures (i.e., NSRST), and 3) non-safety-related with no special treatment (NST). These SSC classifications roughly correspond to the risk-informed safety class (RISC) definitions in 10 CFR 50.69. As used in this document, safety-related SSCs are similar to RISC-1 SSCs, NSRST SSCs are similar to RISC-2 SSCs, and NST SSCs are similar to RISC-4 SSCs. The following sections establish general review criteria for SSCs in each of the three classification categories described above, with the expectation that more detailed guidance could be established for specific SSCs, as necessary, on a design-specific basis.

The staff expects that an applicant will assign each SSC identified for its facility to one of the three classification categories using risk insights based the results of a PRA (or other analysis methodology) and detailed design analysis. Complete results of the applicant's classification of SSCs should be provided in the permit, license, certification, or approval application. The staff will review these classification results as a part of its review of the application. It is important that the staff complete the initial verification of SSC safety and risk significance early in the application review to enable agreement on SSC classifications with the applicant. NRC staff should promptly review any change the applicant makes to the classification category of SSCs as a result of the staff's review, or for other reasons.

Using a graded approach, the rigor of the staff review of SSCs is commensurate with safety and risk significance. The most detailed review should be applied to SSCs with the highest safety and risk significance. A progressively less-detailed review should be applied to other SSCs as the assigned safety and risk significance declines. Performance-based acceptance criteria – for instance, operating regimes imposed through technical specifications - may be used for SSCs in lieu of or in conjunction with detailed analyses as the safety and risk significance of any given SSC decreases.

3.4.3 Scope

The guidance below provides a systematic process for identifying and categorizing event sequences into different LBEs, assessing the risk significance of SSCs, and determining if special treatments beyond normal industrial practices are needed to ensure SSC performance of safety functions in the prevention and mitigation of LBEs and their consequences.

This guidance may be used independent of an applicant's chosen analysis approach. Regardless of the approach used by the applicant, the submittal should identify and analyze LBEs with a clear basis (e.g., PRA) for the selection of events and classification of SSCs. The staff's review of will focus on the those on those SSCs classified as safety-related, using a graded approach for the review of other SSCs.

The NRC staff review, therefore, is focused on risk-significant SSCs, or those SSCs that prevent more severe scenarios from becoming more frequent. For example, the staff may pay particular attention to an SSC that is needed to keep a beyond design basis event sequence from becoming a design basis event. Further, the NRC staff's review should focus on the key characteristics and attributes of SSCs that support maintaining radiological releases below acceptable limits. Key characteristics and attributes of SSCs that may be the focus of the NRC's staff review could include seismic qualification for SSCs required to function or not fail in the presence of a design basis seismic event; material properties and heat transfer characteristics of the reactor fuel; and core components required to transfer heat.

For SSCs classified as safety-related, the review should be focused on aspects of the SSC that have risk-significance and factor into the safety case for the design. Subsection 3.4.3.1 of this review guide provides high-level review guidance for safety related SSCs. For SSCs that are non-safety-related but are assessed by the applicant to have some level of special treatment needed for DID adequacy or other reasons, high-level review guidance is provided in Subsection 3.4.3.2 of this review guide. The remaining non-safety-related SSCs will have a reduced review scope, commensurate with their risk significance and any impacts they may have on the safety-related SSCs. High-level guidance for these SSCs is provided in Subsection 3.4.3.3 of this review strategy. The staff may develop further design-specific guidance for the review of SSC classification based on pre-application interactions with vendors.

3.4.3.1 Safety-Related SSCs

Upon selection of an appropriate set of LBEs, the designer is responsible for defining the set of safety-related SSCs required to perform necessary safety functions and prevent LBEs with more severe consequences. These are the only SSCs that can be relied on to prevent or mitigate against radiological consequences for the defined set of DBAs, including external events. The staff's review will focus on the characteristics needed to ensure mitigation of consequence within the design basis.

Each SSC classified as safety-related should have functional requirements associated with it; some of these will be captured by the PDC. In additional to the functional requirements, the expectation is that most SSCs will have more defined performance requirements associated with specific design functions. These functions may include the ability to prevent an initiating event from progressing to an accident, to mitigate the consequences of an accident, or both. Additionally, safety-related SSCs should have adequate safety margins derived from the analyses. These should address the difference between the level of performance that is deemed acceptable in the safety analysis, and the level of performance that would lead to damage or adverse consequences for all the LBEs in which the SSC performs a prevention or mitigation function. These safety margins could be imposed through a demonstration of a bounding conservative performance for the DBAs, established by considering the uncertainties in performance and the nature of the associated LBEs, in accordance with the licensing approach used by the applicant.

The applicant's definition of what constitutes safety-related will be an area of focus for the staff, especially if the scope of the definition proposed by an applicant differs significantly from the definitions used traditionally or discussed here. Review of acceptability of that definition by the staff will occur as an integral part of initial review of the PDC and event classification process prior to detailed SSC review.

3.4.3.1.1 Safety-Related SSCs - Areas of Review

Safety-related SSCs should be identified, accompanied by a description of the SSCs, analyses, data, or other pertinent information including a description of their safety functions. Interactions or interdependencies with other SSCs should also be identified. As applicable, the following should be identified for each safety-related SSC:

- <u>Design bases</u> for the SSC, with limiting values for performance conditions and expected operating and off normal system envelopes.
- <u>A description and basic design drawings</u>, with relevant system specifications identified.

- Design evaluation(s) for the SSC during normal operation, AOOs, DBEs, DBAs, and BDBEs within the facility licensing basis. This evaluation could draw on operating experience, prototypic testing, analysis, or any combination of the three.
- Applicable testing, inspection and surveillance plans, and any other programmatic controls to ensure the SSC will operate and perform in accordance with the design specifications.
- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For DC and COL reviews, the staff reviews the applicant's proposed ITAAC associated with the SSCs necessary to ensure the as-built design reflects the system as described in the application.

The relationship between the retention of radionuclides within the fuel or fuel matrix and the related supporting safety functions of controlling the generation and removal of heat is expected to remain a major factor in managing the risks of non-LWR designs. This will, in turn, make the relationship between the fundamental safety functions key to the design and licensing of non-LWRs. The construct discussed here reflects a general model for barriers and passive heat removal systems expected to be used for most non-LWR designs. Radiological source terms associated with spent fuel storage and other plant systems will also need to be evaluated. In addition, other potential mechanisms for degrading barriers (e.g., irradiation, chemical interactions) will need to be addressed for normal operation as well as for potential plant transients and postulated accidents.

3.4.3.1.2 Safety-Related SSCs - Acceptance Criteria

Acceptance Criteria Requirements - The Acceptance Criteria Requirements subsection identifies the applicable NRC requirements including specific regulations, NRC orders, and industry codes and standards referenced by regulations. In cases with SSCs that do not have a performance history, this could also include applicable test data used to qualify particular materials, SSCs, or analytical models. This includes the 10 CFR Part 50 applicability (e.g., dose criterion specified in 10 CFR 50.34), plus any PDC associated with the SSC.

Acceptance Criteria – The acceptance criteria will depend on the specific components identified. In some cases, the acceptance criteria will look very similar to those in NUREG-0800 for SSCs that share functions similar to those in the existing LWR fleet. In most cases, however, this is an area where early engagement with the staff will be helpful so that the staff can develop a set of design-specific acceptance criteria for the expected safety-related SSCs.

In general, the staff looks for a demonstration that the SSC can perform the necessary safety functions (i.e., capability and reliability) as assumed in the defined set of DBAs, given a postulated single failure (when applicable). Further, the staff will make a reasonable assurance determination on the capability of the SSC to perform as assumed in the full spectrum of DBEs. This determination can be demonstrated through several avenues, appropriate to the SSC of interest. Possible avenues include:

- Relevant PDC are met;
- Relevant industry codes and standards are met;
- The design basis envelope of parameters is appropriate, such that the SSC's operational tolerances and functional capabilities remain within the bounds specified in the safety analyses;

- The assumed performance of the SSC is adequately demonstrated by a code that is benchmarked and validated for the relevant phenomena, and/or has been tested in accordance with 50.43(e);
- The design description is sufficiently accurate and complete (between the docketed information and more detailed audited documentation) to fulfill the SSC description;
- Operating experience has been considered, or, for new and novel components, a program will be established to gather the appropriate operational data;
- The SSC is adequately robust for all design basis conditions and events;
- The analysis performed demonstrates the SSC remains within the acceptance criteria specified by the designer, as set forth for the specific SSC (e.g., specified acceptable fuel design limits, stress, pressure) as defined within applicable codes or standards or as determined by appropriate qualification programs.
- Appropriate programmatic controls have been applied to the SSC (e.g., Technical Specifications, ITAAC, Initial Test Program, reliability assurance program, in-service inspection program, surveillance program, etc.)

The staff notes that when providing information on SSCs, designers should clarify the difference between design-based SSC verification characteristics and performance-based verification characteristics. Examples of design-based acceptance criteria include those related to SSC design, qualification, materials selection, and suitability for service conditions. Examples of performance-based acceptance criteria include those related to SSC capabilities, reliability, and availability.

An example of a performance-based acceptance criteria that could be leveraged by the staff as part of the basis for a safety finding is a technical specification. A designer may choose to institute a technical specification on a parameter important in the safety analyses that is controlled by a non-safety-related SSC. The parameter is in a controlled condition of operation, so how the condition is achieved or how the system is used, is not in and of itself a point of concern with the staff. Doing this may allow a designer to more clearly separate the safety-related SSCs from analysis inputs that are important in the safety analysis basis but not governed by the safety-related SSC. As an example, heat sink temperature could be entirely dependent on natural processes or augmented by cooling systems – the controlled range of values would need to align with the initial analysis assumptions, but the method for meeting those values would not necessarily need to be evaluated by the staff provided the limiting conditions for operation were sufficient.

3.4.3.1.3 Safety-Related SSCs - Review Procedures

This section will be addressed on a design- and SSC-specific basis. Many of the general procedures and areas of interest are outlined above in the Acceptance Criteria section.

The NRC staff's review will focus on the features of the SSC prevent and/or mitigate radiological releases, that establish the characteristics of the SSC to prevent and/or mitigate DBEs, and any functions that serve to prevent events either from being more severe such that consequences exceed acceptance criteria or change the frequency of an event from increasing in classification.

3.4.3.1.4 Safety-Related SSCs - Evaluation Findings

Staff findings will be made against the regulations where applicable, and the PDCs more specifically on an SSC-specific basis. The general approach for the staff is to focus on the appropriate areas for the SSCs of interest – that is, those that are involved in preventing or mitigating the subject LBEs and their potential consequences. This could involve key characteristics of the SSCs needed to perform necessary safety functions, attributes that contribute to system reliability and functionality, or systems that are required to operate (or not fail) to mitigate an event.

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's SER. The reviewer also states the bases for those conclusions. For example:

> The staff concludes that the applicant has demonstrated that (*insert appropriate decision here, based on the specific SSC*). Specifically, the applicant has thereby satisfied (*regulatory requirements and/or demonstrated the applicability of referenced tests, analyses and qualification programs*) by showing that (…). Based on the above, the staff concludes there is reasonable assurance that the SSC will function in the fashion described by the applicant to mitigate the consequences of a DBE within the dose limits of 10 CFR 50.34/will prevent the frequency of analyzed events from increasing to the next higher category (e.g., from BDBE to DBE).

3.4.3.2 Non-Safety-Related with Special Treatment (NSRST) SSCs

After selection of an appropriate set of LBEs and definition of the design's safety-related SSCs, the applicant should have then performed a further design review to identify other risk-significant or safety-significant SSCs. These SSCs are classified as NSRST. While the safety-related SSCs have a more clearly delineated definition, NSRST SSCs rely on insights from the designer and design review to identify non-safety-related SSCs that perform risk-significant functions or functions required to support DID adequacy.

As an example, one aspect evaluating DID is to ensure the design does not rely on a single SSC to perform a necessary safety function. Were that to be the case, staff would expect the designer to consider whether another SSC that can also perform the necessary safety function should be classified as NSRST for the purposes of DID.

Each SSC defined as NSRST has functional requirements associated with it; in general, only those functions related to safety would be subject to NRC review. Those portions and attributes of the SSC and any supporting functional characteristics that make the SSC NSRST (as should be clear from the selection process) will make up the primary focus of the NRC review. The review of these characteristics may look very similar to the safety-related review for a single aspect of the SSC (i.e., leakage, thermal properties, etc.), commensurate with its risk-significance and role in DID. Staff review, however, should be directed at specific, important portions of the SSC rather than the SSC as a whole, and may rely more heavily on a combination of demonstration of margin, design features, and/or performance based, programmatic requirements.

3.4.3.2.1 NSRST - Areas of Review

NSRST SSCs should be identified, accompanied by a description of the systems, components, analyses, data, or other pertinent information for the area. Interactions or interdependencies with other SSCs that have safety or risk significance should also be identified. As applicable, the following should be identified for each NSRST SSC:

- Design bases for the system, with limiting values for performance conditions and expected operating and off normal system envelopes.
- A description and basic design drawings, with relevant system specifications identified.
- Design description and/or evaluation(s) for the SSC during normal operation and during any plant state where the component is relied on in the risk analysis or to perform a role for DID.

In lieu of the more detailed descriptions associated with safety-related SSCs, the aspects of the SSCs that merit “special treatment” should be identified to focus on the relevant aspects of the NSRST SSC, potentially in the following areas or others to demonstrate the SSC in question can perform the function(s) identified:

- Applicable testing, inspection and surveillance plans, and any other programmatic controls to ensure the SSC will operate and perform in accordance with the design specifications.
- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For DC and COL reviews, the staff reviews the applicant's proposed ITAAC associated with the SSCs necessary to ensure the as-built design reflects the system as described in the application.

Compared to safety-related SSCs, the staff expects that NSRST SSCs may draw more heavily on testing, inspection, and surveillance or other performance-based requirements. If the performance objective can be achieved by setting a readily quantifiable standard that has an unambiguous relationship to a readily measurable quantity and could be enforced, the performance objective may specify the intended acceptance criteria rather than prescribe how the objective is to be attained. In particular, components identified as NSRST may fall under the scope of 10 CFR 50.65 (the “maintenance rule”), which requires in part that licensees shall:

> “monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions.”

In particular, 10 CFR 50.65(b)(2) is applicable to non-safety-related SSCs

> “(i) That are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs); or
>
> (ii) Whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related function; or
>
> (iii) Whose failure could cause a reactor scram or actuation of a safety-related system.”

Not all NSRST may fall within the scope set forth in 10 CFR 50.65; however, certain NSRST SSCs, such as those provided to meet DID considerations, may be appropriately evaluated using criteria similar to requirements in 10 CFR 50.65 to determine their adequacy. In particular, performance-based monitoring is one means for the staff to have confidence in the availability of the SSC.

3.4.3.2.2 NSRST - Acceptance Criteria

Acceptance Criteria Requirements - The Acceptance Criteria Requirements subsection identifies the applicable NRC requirements including specific regulations, NRC orders, and industry codes and standards referenced by regulations. This includes the 10 CFR Part 50 applicability, plus any PDC associated with the SSC.

Acceptance Criteria - The acceptance criteria will depend on the specific components identified. In some cases, the acceptance criteria will look very similar to those existing in NUREG-0800 for SSCs that share functions similar to those in the existing LWR fleet, with a reduced scope focused on the aspects of the SSC that resulted in its classification as NSRST. In most cases, however, this is an area where early engagement with the staff will be helpful so that the staff can develop a set of design-specific acceptance criteria for the expected NSRST SSCs.

In general, staff will be looking for a demonstration that the SSC can perform as assumed in the scenarios the SSC is relied on to perform a risk- or safety-significant function. Further, staff will make a reasonable assurance determination on the capability of the SSC to perform as assumed in the full spectrum of LBEs. This determination can be demonstrated through several avenues, appropriate to the SSC of interest. Possible avenues include:

- Relevant PDCs are met;
- Relevant industry codes and standards are met;
- The design basis envelope of parameters is appropriate, such that the SSC’s operational tolerances and functional capabilities remain within the bands specified in the safety analyses;
- The assumed performance of the SSC to perform the specific function that entails special treatment is adequately demonstrated by a code that is benchmarked and validated for the relevant phenomena, and/or has been tested in accordance with 50.43(e);
- The design description is sufficiently accurate and complete (between the docketed information and more detailed audited documentation) to fulfill the SSC description;
- Operating experience has been considered, or, for new and novel components, a program will be established to gather the appropriate operational data;
- The SSC is adequately robust for all design basis conditions and events;
- The analysis performed demonstrates the SSC remains within the acceptance criteria specified by the designer, as set forth for the specific SSC (e.g., specified acceptable fuel design limits, stress, pressure).
- Appropriate programmatic controls for performance-based parameters have been applied to the SSC (e.g., Technical Specifications, ITAAC, Initial Test Program, reliability assurance program, in-service inspection program, maintenance rule, surveillance program etc.).

For NSRST SSCs, while the above review should be more focused on specific risk-significant or DID functions and attributes of the SSCs, the review may be supplemented by:

- detailed review of key PRA assumptions associated with the SSC including sensitivities and uncertainties;
- investigation of any potential adverse systems interactions for the SSC that may interfere with the SSC function; and

- evaluation of the stated reliability and availability of the SSC, given the review will be focused on specific SSC mission functions rather than the more detailed systemic SSC review performed on safety-related SSCs.

3.4.3.2.3 NSRST - Review Procedures

This section will be addressed on a design- and SSC-specific basis. Many of the general procedures and areas of interest are outlined above in the acceptance criteria section. For NSRST SSCs in particular the assumed capabilities and reliabilities of SSCs and the related DID assessments are also likely to be used as part of the justification for the SSC classification and associated credit for prevention or mitigation. Staff review, therefore, will focus on those areas credited by the designer as appropriate.

3.4.3.2.4 NSRST - Evaluation Findings

Staff findings will be made against the regulations where applicable, and the PDC (as appropriate) on an SSC-specific basis. These findings are tailored to the applicable function of the SSC that makes it risk-significant, provides DID, or the adverse impact failure of the NSRST SCC can have on a safety-related SSC.

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support the design bases for the SSC, as appropriate to the risk significance of the SSC. The reviewer also states the bases for those conclusions. The staff expects designers to appropriately consider DID adequacy and risk insights to ensure SSCs are appropriately classified as NSRST (as opposed to safety-related).

3.4.3.3 Non-safety-related SSCs

Non-safety-related SSCs should be identified to the extent practicable, accompanied by a general description of the systems, components, or linkages to other systems. Interactions or interdependencies with other safety-related or NSRST SSCs should also be identified.

3.4.3.3.1 Non-safety-related SSCs - Areas of Review

Designers should describe applicable interactions with safety-related SSCs, as well as the role of the SSCs in the safety case and risk profile of the overall design. For SSCs that are effectively isolated from the design by a higher classification SSC (e.g., a set of safety-related isolation valves that represent the barrier from the primary and secondary coolant), the staff review can be restricted to the component that serves as the barrier. Any contribution to event initiation or impact to a safety-related or NSRST SSC will be reviewed and evaluated by the staff, though likely as a subset of review of a safety-related SSC.

3.4.3.3.2 Non-safety-related SSCs - Acceptance Criteria

<u>Acceptance Criteria Requirements</u> – Non-safety-related SSCs that serve to meet regulatory requirements are reviewed to the extent that the applicant demonstrates that the SSC meets the regulatory requirements.

Sufficient information should be provided such that the staff can determine that no adverse impact is made by non-safety-related SSCs supporting safety-related systems.

3.4.3.3.3 Non-safety-related SSCs - Review Procedures

This section will be addressed on a design- and SSC-specific basis. In general, the staff review of non-safety-related SSCs should be heavily risk-informed, with the primary focus on ensuring the SSC cannot adversely impact an accident sequence (i.e., no adverse impact on safety-related or NSRST SSCs).

3.4.3.3.4 Non-safety-related SSCs - Evaluation Findings

For components that are not safety-related, NSRST, required by regulations, or credited to mitigate a DBE, generally the staff makes no finding beyond any potential impact on SSCs or analyses in the previous categories. For SSCs that are required to meet a regulatory requirement, the staff finding is focused narrowly on the ability of the SSC to meet the requirement, relying on programmatic controls to the maximum extent practical, if applicable. In general, any information about non-safety-related SSCs with impacts on risk- or safety-significant SSCs should be mentioned in the design description for a safety-related or NSRST SSC so that the staff can make its required reasonable assurance safety finding. The staff's review will not focus on these systems except to the extent they interact adversely with safety-related or NSRST SSCs or are required by other regulations. The staff expects designers to appropriately consider DID and risk insights to ensure SSCs are appropriately classified.

3.5 Analysis and Evaluation of the Integrated System Design

10 CFR 50.34(a)(4), 10 CFR 52.47(a)(4), 10 CFR 52.79(a)(5), 10 CFR 52.137(a)(4), and 10 CFR 52.157(f)(1) require that an analysis and evaluation of the design and performance of SSCs be performed to determine (1) the margins of safety during normal operations and AOOs, and (2) the adequacy of SSCs for the prevention and mitigation of the consequences of accidents for a CP, DC, COL, SDA, or ML.

Additional requirements are provided for the consideration of BDBE. Specifically, (1) 10 CFR 50.62, 10 CFR 52.47(a)(15), 10 CFR 52.79(a)(42), 10 CFR 52.137(a)(15), and 10 CFR 52.157(f)(7) require that applicants provide information to describe how an applicant will comply with requirements for reduction of risk from anticipated transients without scram events, and (2) 10 CFR 50.63, 10 CFR 52.47(a)(16), 10 CFR 52.79(a)(9), 10 CFR 52.137(a)(16), and 10 CFR 52.157(f)(5) require that applicants provide a coping analysis, and describe any design features necessary to address station blackout. Furthermore, 10 CFR 52.47(a)(23)[8], 10 CFR 52.79(a)(38), 10 CFR 52.137(a)(23), and 10 CFR 52.157(f)(23) require that applications for LWR[9] designs include a description and analysis of design features for the prevention and mitigation of severe accidents and their consequences. The risk consideration associated with these BDBEs, as well as other low frequency event sequences, will be considered as part of the required PRA.

The review of the analysis and evaluation of the integrated system design has the following review interfaces:

- The analysis and evaluation of the integrated system design are compared against acceptance criteria established by the PDCs (PDCs are discussed in Section 3.1)
- The analysis and evaluation of the integrated system design are performed for event sequences determined by the selection of LBEs (LBE Selection is discussed in Section 3.2)

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[8] In SRM-SECY-15-002, "Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications," [20], the Commission confirmed that the "Commission Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants," [21] should be applied to applications under 10 CFR Part 50 in a manner consistent with 10 CFR Part 52.

[9] The Commission's Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants is not restricted to light-water reactor designs; however, the policy did focus on large LWR designs.

- The analysis and evaluation of the integrated system design give appropriate credit for SSC performance in accordance with SSC classification (SSC Classification is discussed in Section 3.3)
- The analysis and evaluation of the integrated system design are provided information on SSC performance and reliability from the design basis of SSCs (Design Basis of SSCs is discussed in Section 3.4)
- The analysis and evaluation of the integrated system design calculates the consequences of event sequences, which could be used in a PRA (PRA is discussed in Section 3.6)

The scope of the staff's review of this area is further discussed in Section 3.5.1 for normal operations, including AOOs, Section 3.5.2 for DBAs, and Section 3.5.3 for analyses supporting safety-significant SSC performance. Additionally, Section 3.5.4 discusses safety limits associated with fission product barriers and accident source term which either directly establish or inform acceptance criteria for the analysis evaluation of the integrated system design.

3.5.1 Margin of Safety During Normal Operation and AOO

Evaluation of the integrated system design during normal operation and AOOs is performed to (1) determine the margins to the safety limits for the barriers protecting against the release of radioactive material, and (2) ensure that the radiological consequences, if any, satisfy the Commission's regulations. The identification of appropriate AOOs for a particular design may use PRA in a risk informed manner in accordance with NEI 18-04 or by other means, such as a deterministic evaluation. The analyses performed in the evaluation of normal operation and AOOs can be more realistic and are not required to be as rigorous as the analyses performed to evaluate DBAs because (1) normal operation and AOOs are expected to have significant margin to the safety limits for the barriers protecting against the release of radioactivity, (2) the radiological consequences of normal operation and AOOs are expected to be insignificant, (3) the evaluation of normal operation and AOOs are not used to demonstrate the satisfactory performance of necessary safety function (see Section 3.5.2 of this report), and (4) non-safety-related SSCs relied upon during normal operation, including AOOs, are expected to be capable of performing their intended functions as described in this review guide and in accordance with 10 CFR 50.65 and 10 CFR 52.79(a)(15), as applicable. A review of the analyses supporting normal operation and AOOs will be performed using a graded approach to criteria b-d described in Section 3.5.2 of this report. This graded approach will consider the margin to the safety limits for the barriers protecting against the release of radioactivity, and the potential radiological consequences of the scenario analyzed.

3.5.2 Adequacy of SSCs for DBAs

DBA analyses are performed to demonstrate that safety-related SSCs are capable of performing their necessary safety functions. The identification of appropriate DBAs for a particular design may use PRA in a risk-informed manner in accordance with NEI 18-04 or by other means, such as a deterministic evaluation. DBA analyses are expected to be highly reliable (i.e., the results from the analyses are worthy of belief or confidence) because (1) these analyses are used to demonstrate the adequacy of SSCs to perform necessary safety-functions, and (2) DBA scenarios are expected to have more severe consequences than AOO scenarios. The review of these analyses will consider the following items:

a. Mitigation of the DBAs only credits safety-related SSCs

b. The acceptability of the analysis method
    1. The analysis methodology contains the appropriate modeling capabilities
        i. Analysis methodology is capable of modeling the geometry of the system
        ii. Analysis methodology is capable of modeling the material properties associated with the system
        iii. Analysis methodology is capable of modeling the necessary physics associated with relevant phenomena
    2. Adequate verification and validation have been performed
        i. The analysis tool is tested to ensure that modeling capabilities are implemented correctly
        ii. The analysis tool is tested to ensure that modeling capabilities are adequate to predict the required figures of merit
    3. Adequate uncertainty analysis and sensitivity studies have been performed
        i. The uncertainty in the quantified analysis results is sufficiently well understood to establish suitable safety margin
        ii. Sensitivity of the figure-of-merit to input parameters and specific models has been investigated
c. The analysis inputs adequately address parameter uncertainty
    1. Each input parameter is controlled within the bounds of safety analysis through (1) plant operating parameters, (2) reactor trip setpoints, or (3) other physical bounds
d. The results of the analysis meet the acceptance criteria specified by the designer
    1. Analysis shows that thermal limits and radiological consequences do not exceed limits
    2. Fuel and/or fission product barrier failure mechanisms, the associated limits corresponding to those failure mechanisms, and the radiological source term have been identified over the range of applicability.
        i. The identification of failure mechanisms, associated limits, and contribution to source term are supported by credible experimental data
        ii. Data reduction and/or correlation development is performed using acceptable techniques (e.g., standard statistics)

In addressing the items above, the staff will make findings on the lowest level criteria with the understanding the higher-level criteria are satisfied by addressing the lower-level supporting criteria (e.g., Item b is satisfied by addressing b.1 – b.3, and Item b.1 is satisfied by addressing items b.1.i-b.1.iii). The items identified above are informed by the guidance provided in RG 1.203, “Transient and Accident Analysis Methods,” [22] and standard engineering practice[10]. When considering items b and c from the list above, a review of the analyses will also consider the margin to safety limits. For example, the degree of code assessment and uncertainty quantification can be reduced if the design incorporates sufficient margin to bound analysis uncertainties. In addition, the review can consider operational programs to address analysis uncertainties. For example, subcritical testing of the reactor can be performed to verify analysis tool predictions.

[10] Substantial literature is available on the subject of credibility in scientific computing. A particularly well-known source on this topic is *Verification and Validation in Scientific Computing* by Oberkampf and Roy, Cambridge 2010, [23].

3.5.3 Evaluation of Beyond Design Basis Events

BDBE analyses are performed to (1) assess the plant risk against events of very low frequency, (2) inform emergency planning requirements, and (3) assess the DID adequacy of the plant design. The identification of appropriate BDBEs should be performed as part of the plant's PRA or alternate analysis methodology in a risk-informed manner. The analyses performed in the evaluation of BDBEs are expected to be best-estimate because the purpose of these analyses is to provide a realistic assessment of plant risk. The staff's review of BDBE analyses will focus on the accuracy of the input parameters and will utilize a graded approach. This graded approach will consider the impact of the event scenario and associated input parameters on the calculated risk.

3.5.4 Safety Limits of Fission Product Barriers and Accident Source Term

The analysis and evaluation of the integrated system design are used to assess and quantify the potential release of fission products. These evaluations include (1) an evaluation of the fission product barriers against their design limits, and (2) calculations of the radiological consequences associated with the event scenarios. The assessment and quantification of fission product release for event scenarios is essential to the staff's review because (1) fission product barrier performance requirements are included in the PDCs for the facility and establish the acceptance criteria for the integrated system design analyses, (2) radiological consequences for event sequences, in combination with event frequency, are used to assess the plant risk in a PRA, and (3) the potential for fission product release impacts licensing decisions on reactor siting and emergency planning. Due to the significance and broad implications associated with the assessment and quantification of fission product release, there should be a high degree of confidence in the accuracy of the safety limits for fission product barriers and accident source term[11]. When reviewing safety limits for fission product barriers and accident source term, the staff will consider the following:

a. The identification of fuel failure mechanisms, the associated limits, and the contribution to source term are supported by experimental data that cover the needed range of applicability.
b. The performance of fission product barriers credited to prevent and/or inhibit the release of radionuclides are supported by experimental data that cover the needed range of applicability.
c. Experimental data reduction and/or correlation development is performed using standard statistical techniques.

3.6 Probabilistic Risk Assessment

Applicants for nuclear power plants are required to provide a description of the design- or plant-specific PRA and its results as part of an application submitted under 10 CFR Part 52. Specifically, 10 CFR 52.47(a)(27), 10 CFR 52.79(a)(46), 10 CFR 52.137(a)(25), and 10 CFR 52.157(f)(31) prescribe the requirements for DC, COL, SDA, and ML applications, respectively.

Further, the Commission has established policies for applicants for construction permits and operating licenses under 10 CFR Part 50 to perform PRAs. Specifically, SRM-SECY-15-002,

[11] A similar statement is made in the staff's recommendation in SECY-93-092 "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements," [24] regarding the understanding of reactor and fuel performance during normal and off-normal conditions to permit a mechanistic source term analysis.

“Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications,” [20] confirmed the Commission’s expectations that PRAs would be developed and described in new power reactor applications submitted under 10 CFR Part 50. Finally, the Commission’s, “Policy Statement on Safety Goals for the Operations of Nuclear Power Plants,” [25] lays out cumulative risk metrics that are typically demonstrated by conducting PRAs.

For a risk-informed and performance-based licensing approach such as the LMP discussed in NEI 18-04, a PRA has an expanded role in the systematic assessment of risk, including initiating event analysis, accident sequence analysis, and consequence analysis. In LMP, a PRA is an essential tool in the selection of LBEs, the classification and performance criteria for SSCs, and the evaluation of DID adequacy.

The ability of a PRA to support risk-informed regulatory decision making can be measured in terms of its appropriateness regarding scope, conformance with the technical elements (e.g., plant operating state analysis, success criteria development, human reliability assessment, and so on) of a PRA, level of detail, and plant representation. Collectively the NRC generalizes these concepts as the “technical acceptability” of a PRA. Early establishment of PRA technical acceptability by an applicant is essential for efficiency and effectiveness in a risk-informed and performance-based licensing review.

Should an applicant propose the use of another analysis approach, such as a traditional deterministic approach, in lieu of a risk-informed and performance-based approach discussed above, adequate justification should be provided, including any necessary requests for exemptions from applicable regulatory requirements. Any alternative approach to that based on PRA should still demonstrate a systematic assessment of risk, including initiating event analysis, accident sequence analysis, and consequence analysis, selection of LBEs, the classification and performance criteria for SSCs, and the evaluation of DID adequacy. The staff reviewing methodologies other than that based on PRAs may use the acceptance criteria, review procedures, and evaluation findings provided below to the extent that this guidance may be adapted generically to other analysis approaches.

The review of a PRA will have the following interfaces:

- PDC development may be informed by risk insights obtained through a PRA (Development of the PDCs are discussed in Section 3.1)
- LBE selection may be informed by the initiating events and event sequence frequencies determined from a PRA (Selection of LBEs is discussed in Section 3.2)
- SSC classification may be informed by the risk profile established by a PRA (SSC classification is discussed in Section 3.3)
- A PRA uses information on SSC performance and reliability from the design basis of SSCs (Design Basis of SSCs is discussed in Section 3.4)
- Consequences of event sequences can be calculated through the analysis and evaluation of the integrated system design (Analysis and evaluation of the integrated system design is discussed in Section 3.5)

3.6.1 Areas of Review

The areas of review related to a PRA typically include the following:

- Description of the PRA and its results in the application

- Technical acceptability of the PRA
- Application of PRA results and insights
- Quality control applied to PRA development
- PRA maintenance and update process

3.6.2 Acceptance Criteria

Description of the PRA and its results in the application

The applicant describes the PRA and its results in the application and performed sufficiently complete and scrutable analyses. In addition, the PRA results and insights support the application, and the maintenance and update process of the PRA is provided.

Technical acceptability of the PRA

The staff will determine the technical acceptability of a PRA based on the following considerations:

- The applicant demonstrates that the PRA is technically acceptable for the application. For non-LWRs, the guidance in ASME/ANS RA-S-1.4-2013 generally provides an acceptable means to establish PRA technical acceptability. This standard, which was developed for a broad spectrum of non-LWR designs, has been published for trial use and has not yet been reviewed or endorsed by the NRC. Therefore, if used, an applicant should demonstrate that it has appropriately tailored ASME/ANS RA-S-1.4-2013 for the application in accordance with Section 3 of the standard and provided sufficient justification for its use. The applicant should also demonstrate that the appropriate portions of the standard were used in performing a design- or plant-specific PRA, including use of current industry best practices, such as PRA modeling software and thermal-hydraulic modeling codes. For areas that ASME/ANS RA-S-1.4-2013 "Probabilistic Risk Assessment Standard for Non-LWR Nuclear Power Plants," [26] does not adequately address or cover (e.g., other sources of radioactive material, such as the spent fuel system), the applicant should establish additional criteria/guidance to perform the PRA.
- The applicant has addressed key technical elements of Section 4 of ASME/ANS RA-S-1.4-2013 based on the high-level and supporting requirements of the standard as tailored for the application.
- The applicant has established appropriate risk metrics to be used in their PRA.
- The applicant has appropriately addressed the risk associated with any multi-unit or multi-module aspects of the design.
- The applicant has appropriately addressed the reliability of inherent features or passive safety systems that are relied upon for the design or safety case.
- The applicant has addressed the risk associated with different sources of radioactive material both within and outside of the reactor core. Note that the technical requirements in ASME/ANS RA-S-1.4-2013 are limited to sources of radioactive material within the Reactor Coolant System Pressure Boundary (RCPB) as defined in the standard.
- The applicant has identified the appropriate criteria for determining the risk significance of various elements modeled in PRA including initiating events, SSCs, and operator

actions by performing PRA importance analyses based on appropriate importance measures (e.g., Risk-Achievement Worth, Risk Reduction Worth, and Fussell-Vesely).
- The applicant has appropriately analyzed the uncertainties in the PRA and performed sensitivity analyses addressing the uncertainties. The applicant has also identified key assumptions made in the PRA that may influence the uncertainties.
- A peer review has been performed consistent with Section 6 of ASME/ANS RA-S-1.4-2013. The applicant's peer review process is based on a documented procedure utilizing the best practices for PRA peer reviews.

<u>Application of PRA results and insights</u>

The staff will confirm that PRA results and insights have been adequately described and applied as appropriate, which typically include the following:

- Through the development of the PRA, the applicant has identified and addressed potential risk-significant design features and plant operational vulnerabilities. The applicant has reduced or eliminated any significant risk contributors of existing operating plants that are applicable to their design by selecting alternative features, operational strategies, or design options.
- Risk insights based on systematic evaluations of the risk associated with the design are identified including; (1) the design's robustness, levels of DID, and tolerance of severe accidents, and (2) the risk significance of potential human errors associated with the design.
- The applicant's design meets the applicable Commission policy-level goals related to PRA and severe accidents.
- The applicant's design- or plant-specific risk metrics established (e.g., as substitutes to those intended for LWRs) are acceptable in implementing the goals.
- Risk insights are appropriately identified regarding important risk contributors to each of the identified risk metrics.
- The applicant's systematic assessment of all potential internal and external hazards led to a list of LBEs considering their frequency and consequence.
- The evaluation of the risks of the LBEs against frequency and consequence considerations is appropriately performed.
- Information (e.g., uncertainties, risk margins, insights, SSC reliability) is appropriately used for selection of protective strategies and in the evaluation of DID adequacy in a risk-informed and performance-based manner.
- Other programs (e.g., Reliability Assurance Program) that use input from PRA results and insights are appropriately identified and described.

<u>Quality control applied to PRA development</u>

The applicant has established a quality control program[12] and developed the PRA in accordance with the program, which includes that following key areas:

- The applicant used personnel qualified for the analysis.

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[12] The PRA is not subject to the quality assurance requirements of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

- The applicant used procedures that ensure control of documentation, including revisions, and provide for independent review, verification, or checking of calculations and information used in the analyses.
- The applicant documents and maintains records, including archival documentation as well as submittal documentation.
- The applicant uses procedures that ensure that appropriate attention and corrective actions are taken if assumptions, analyses, or information used previously are changed or determined to be in error.

<u>PRA maintenance and update process</u>

The applicant has established and described the approach for maintaining and periodically upgrading the PRA.

3.6.3 Review Procedures

This section will be addressed on a design- and SSC-specific basis. Many of the general procedures and areas of interest are outlined above in the acceptance criteria section.

3.6.4 Evaluation Findings

Based on its review of an applicant's PRA, the staff should be able to make the following evaluation findings related to the adequacy of the analysis:

- The applicant has used the PRA and/or severe accident evaluation to identify and assess the balance of preventive and mitigative features, including consideration of operator actions.
- The applicant has used PRA results and insights to support other programs, such as Reliability Assurance Program.
- The PRA reasonably reflects the as-designed, as-built, and as-operated plant, and the PRA maintenance program will ensure that the PRA will continue to reflect the as designed, as-built, and as-operated plant, consistent with its identified uses and applications.
- The PRA is of the appropriate scope, level of detail, and technical adequacy for its identified uses and applications.
- The applicant has performed adequate systematic evaluations of the risk associated with the design and used them to identify risk-informed safety insights in a manner consistent with the Commission's stated goals.
- In accordance with the Commission's objectives for new reactor designs, the applicant has introduced appropriate and effective design features that contribute to the mitigation of severe accidents.

4 References

1. U.S. Nuclear Regulatory Commission, "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness," December 2016. (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16357A670).
2. U.S. Nuclear Regulatory Commission, "NRC Non-Light Water Reactor Near-Term Implementation Action Plans," July 2017. (ADAMS Accession No. ML17165A069).
3. U.S. Nuclear Regulatory Commission, "NRC Non-Light Water Reactor Mid-Term and Long-Term Implementation Action Plans," July 2017. (ADAMS Accession No. ML17164A173).
4. U.S. Nuclear Regulatory Commission, SECY 19-009, "Advanced Reactor Program Status," January 2019. (ADAMS Accession No. ML18346A075).
5. U.S. Nuclear Regulatory Commission, Draft Guide (DG)-1353, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors," May 2019. (ADAMS Accession No. ML18264A093).
6. Nuclear Energy Institute, NEI-18-04, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development," Revision 1, August 2019. (ADAMS Accession No. ML19241A472).
7. U.S. Congress, "Nuclear Energy Innovation and Modernization Act," January 2019. (https://www-pub.iaea.org/MTCD/publications/PDF/Pub1715web-46541668.pdf)
8. U.S. Nuclear Regulatory Commission, "A Regulatory Review Roadmap for Non-Light Water Reactors," December 2017. (ADAMS Accession No. ML17312B567).
9. U.S. Nuclear Regulatory Commission, "Policy Statement on the Regulation of Advanced Reactors" (73 FR 60612), October 14, 2008. (ADAMS Accession No. ML082750370).
10. U.S. Nuclear Regulatory Commission, RIS 17-08, "Process for Scheduling and Allocating Resources for Fiscal Years 2020 through 2022 for the Review of New Licensing Applications for Light-Water Reactors and Non-Light-Water Reactors," December 2017. (ADAMS Accession No. ML17262B022).
11. U.S. Nuclear Regulatory Commission, SRM SECY 03-0047, "Policy Issues Related to Non-Light-Water Reactor Designs," June 2003. (ADAMS Accession No. ML031770124).
12. U.S. Nuclear Regulatory Commission, NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," February 1996. (https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1537/part1/sr1537p1.pdf)
13. U.S. Nuclear Regulatory Commission, RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," November 1978. (ADAMS Accession No. ML011340122)
14. U.S. Nuclear Regulatory Commission, RG 1.206, "Applications for Nuclear Power Plants," October 2018. (ADAMS Accession No. ML18131A181)
15. U.S. Nuclear Regulatory Commission NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," August 2004. (ADAMS Accession No. ML042080088).

16. International Atomic Energy Agency, SSR-2/1, “Safety of Nuclear Power Plants: Design,” Revision 1, 2016. (https://www-pub.iaea.org/MTCD/publications/PDF/Pub1715web-46541668.pdf)
17. U.S. Nuclear Regulatory Commission, NUREG-0800, “Introduction - Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Small Modular Reactor Edition,” January 2014. (ADAMS Accession No. ML18131A181)
18. U.S. Nuclear Regulatory Commission, “Expectations for Safety Evaluation Reports,” August 2019. (ADAMS Accession No. ML19218A142).
19. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” April 2018 (ADAMS Accession No. ML17325A611).
20. U.S. Nuclear Regulatory Commission, SRM-SECY-15-0002, “Staff Requirements – SECY-15-0002 – Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications,” September 2015, (ADAMS Accession No. ML15266A023).
21. U.S. Nuclear Regulatory Commission, “Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants,” (50 FR 32138) (ADAMS Accession No. ML003711521).
22. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.203, “Transient and Accident Analysis Methods,” December 2005. (ADAMS Accession No. ML053500170).
23. W. Oberkampf and C. Roy, “Verification and Validation in Scientific Computing,” Cambridge University Press, 2010. (https://doi.org/10.1017/CBO9780511760396).
24. U.S. Nuclear Regulatory Commission, SECY 93-092, “Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements,” April 1993. (ADAMS Accession No. ML040210725).
25. U.S. Nuclear Regulatory Commission, “Policy Statement on Safety Goals for the Operations of Nuclear Power Plants,” (51 FR 28044) (ADAMS Accession No. ML).
26. American Society of Mechanical Engineers and American Nuclear Society, Standard ASME/ANS RA-S-1.4-2013, “Probabilistic Risk Assessment Standard for Non-LWR Nuclear Power Plants,” 2013.

5 Acronyms

Nuclear Regulatory Commission (NRC)
non-light water reactor (non-LWR)
implementation action plans (IAPs)
technology-inclusive, risk-informed, and performance-based (TI-RIPB)
licensing modernization project (LMP)
Nuclear Energy Institute (NEI)
U.S. Department of Energy (DOE)
draft regulatory guide (DG)
regulatory guide (RG)
Nuclear Energy Innovation and Modernization Act (NEIMA)
safety evaluation report (SER)
regulatory information summary (RIS)
probabilistic risk assessment (PRA)
licensing basis events (LBEs)
systems, structures, and components (SSCs)
defense-in-depth (DID)
staff requirements memorandum (SRM)
frequency-consequence (F-C)
anticipated operational occurrences (AOOs)
design basis events (DBEs)
beyond DBEs (BDBEs)
NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (SRP)
technology-inclusive content of an application project (TICAP)
integrated decision-making process panel (IDPP)
principal design criteria (PDC)
general design criteria (GDC)
construction permit (CP)
design certification (DC)
combined license (COL)
standard design approval (SDA)
manufacturing license (ML)
non-safety-related with special treatment (NSRST)
non-safety-related with no special treatment (NST)
risk-informed safety class (RISC)
Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)
reactor coolant System pressure boundary (RCPB)

ATTACHMENT 1 – Analysis of Applicability of NRC Regulations for Non-Light Water Reactors

In addition to considerations for LBE selection and SSC classification, applicants should evaluate the applicability of the NRC's regulations to specific designs. The NRC staff recognizes that the applicability of certain regulations developed for LWRs may need to be analyzed to determine the appropriateness for non-LWRs. As such, the staff has prepared this attachment presenting the NRC staff's analysis of those regulations anticipated to be generally applicable to all non-LWR applicants for construction permits and operating licenses under 10 CFR Part 50 and combined licenses and design certifications under 10 CFR Part 52. For simplicity, the staff did not include those regulations only applicable to early site permits, limited work authorizations, standard design approvals and manufacturing licenses. Table 1 of this attachment provides additional context for areas where exemptions may be expected for non-LWR designs. Tables 2, 3, and 4 present a non-exhaustive list of regulations to be considered by non-LWR designers, with expected applicability for each regulation in the table. In Tables 2-4, the expected applicability of a regulation to a non-LWR is indicated by "Y" for Yes, "N" for No, or "*" for those items discussed in more detail in Table 1. Note: Use of LMP could prompt an applicant to request exemptions from certain regulations, such as the 10 CFR 50.2 definition of "safety-related structures, systems and components." This attachment and the tables contained within do not constitute an exhaustive set of applicable regulations and potential exemptions. Applicants may request exemptions from the NRC's regulations on a case-by-case basis depending on the nature of the specific design and independent determination of regulatory applicability.

Table 1 – Areas with expected exemptions

Topical Area	Regulation	Discussion
Fission Product Release	10 CFR 50.34(a)(1)(ii)(D) 10 CFR 52.47(a)(2)(iv) 10 CFR 52.79(a)(1)(VI)	Requires that an applicant shall assume a fission product release from the core into the containment and that the applicant perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents. This language is LWR-centric and the prescriptive nature is not consistent with the commission policy in SRM-SECY-18-0060 that would allow functional containment for fission project retention rather than assuming that the facility would include a traditional pressure retaining containment.
Criticality	10 CFR 50.68 10 CFR 52.47(a)(17) 10 CFR 52.79(a)(43)	Paragraph (b) of 10 CFR 50.68 provides LWR-centric conditions for criticality safety. Non-LWR differs significantly from traditional fuel types used in LWRs and in many cases have higher enrichment.

Topical Area	Regulation	Discussion
		The corresponding regulations in 10 CFR Part 52 that cite 10 CFR 50.68 would be included in the exemption if applicable to the application type.
Pressurized Thermal Shock Events	10 CFR 50.34(b)(9) 10 CFR 52.47(a)(14) 10 CFR 52.79(a)(7)	Most non-LWRs are low pressure (near atmospheric) designs, therefore the conditions associated with a pressurized thermal shock event for an LWR are not expected to be applicable.
Reactor Coolant Pressure Boundary	10 CFR 50.2 (Definitions – Basic Component) 10 CFR 50.2 (Definitions – Safety-related SSCs) 10 CFR 50.36(c)(2)(ii) 10 CFR 50.49(b) 10 CFR 50.65 10 CFR Part 50 Appendix S	The reactor coolant pressure boundary for an LWR provides a fission product retention barrier for the release of radionuclides. However, it is expected that in some non-LWRs, the reactor coolant boundary would not serve this function. Fission product retention is provided by the functional containment. Therefore, the statement in 10 CFR 50.2 (2 instances), 10 CFR 50.49 (b), and 10 CFR 50.65, "The integrity of the reactor coolant pressure boundary" is not applicable and an exemption is expected. In 10 CFR 50.36 (c)(2)(ii), "significant abnormal degradation of the reactor coolant pressure boundary is not applicable and can be replaced by "significant abnormal degradation of the functional containment" via an exemption.
Analysis of SSCs and ECCS Evaluation	10 CFR 50.34(a)(4) 10 CFR 50.34(b)(4) 10 CFR 50.46a 10 CFR 52.47(a)(4) 10 CFR 52.79(a)(5)	The first sentence in each of these citations refers to the analysis and evaluation of the design and performance of SSCs, which is still applicable to non-LWRs. However, the second sentence of each citation states that an analysis and evaluation of the ECCS cooling performance shall be provided in accordance with 10 CFR 50.46, which is not applicable to non-LWRs. Therefore, a partial exemption is anticipated for these requirements. Similarly, 10 CFR 50.46a requires reactor coolant system venting systems, but does not explicitly reference LWRs.
Miscellaneous	Containment Leak rate -10 CFR 52.79 (a)(12) Material surveillance program -10 CFR 52.79 (a)(13) SBO - 10 CFR 52.47(a)(16) and 10 CFR 52.79 (a)(9) ATWS	A subset of regulations in 10 CFR 52.47 and 10 CFR 52.79 do not state that they are applicable to LWRs only; however, they cross-reference other regulations that are applicable to LWRs only, and therefore these regulations should not be applicable to non-LWRs. A literal reading could necessitate an exemption request.

Topical Area	Regulation	Discussion
	- 10 CFR 52.47(a)(15) and 10 CFR 52.79 (a)(42)	

Table 2 – 10 CFR Part 50 Requirements, as applicable to non-LWRs

Regulation	Topic	Presumed applicability to non-LWRs
10 CFR 50.2	Definitions	Y
10 CFR 50.2	Definitions - Controls	Y
10 CFR 50.2	Definitions - Safe shutdown	Y
10 CFR 50.2	Definitions – Reactor coolant pressure boundary	*
10 CFR 50.2	Definitions – Safety-related structures	Y
10 CFR 50.3	Interpretations	Y
10 CFR 50.4	Written communications	Y
10 CFR 50.5	Deliberate misconduct	Y
10 CFR 50.7	Employee protection	Y
10 CFR 50.9	Completeness and accuracy of information	Y
10 CFR 50.10	License required; LWA	Y
10 CFR 50.11	Exceptions and exemptions from licensing requirements	Y
10 CFR 50.12	Specific exemptions	Y
10 CFR 50.13	Attacks and destructive acts by enemies of the United States; and defense activities	Y
10 CFR 50.20	License Classification	Y
10 CFR 50.21	Class 104 licenses for commercial and industrial facilities	Y

Regulation	Topic	Presumed applicability to non-LWRs
10 CFR 50.22	Class 103 licenses for commercial and industrial facilities	Y
10 CFR 50.23	CPs	Y
10 CFR 50.30	Filing of application; oath or affirmation	Y
10 CFR 50.31	Combining applications	Y
10 CFR 50.32	Elimination of repetition	Y
10 CFR 50.33	Applicant Information	Y
10 CFR 50.34(a)(13)	Aircraft Impact	Y
10 CFR 50.34(b)	FSAR	Y
10 CFR 50.34(b)(1)	Site Evaluation (10 CFR Part 100) for CP Applications	Y
10 CFR 50.34(b)(2)	FSAR Description of SSCs	Y
10 CFR 50.34(b)(3)	Kinds and Quantities of Radioactive Materials (10 CFR Part 20)	Y
10 CFR 50.34(b)(4)	Analysis of SSCs and ECCS Evaluation	Y, in part
10 CFR 50.34(b)(5)	Description and Evaluation of Applicable Programs including Research and Development	Y
10 CFR 50.34(b)(6)	Facility Operation Documentation (programs, TS, etc.)	Y
10 CFR 50.34(b)(7)	Technical Qualifications	Y
10 CFR 50.34(b)(8)	Operator Requalification Program	Y
10 CFR 50.34(b)(9)	Description of Pressurized Thermal Shock	*
10 CFR 50.34(b)(10)	Seismic Criteria in Appendix S	Y
10 CFR 50.34(b)(11)	Siting Criteria	Y

Regulation	Topic	Presumed applicability to non-LWRs
10 CFR 50.34(b)(12)	Aircraft Impact	Y
10 CFR 50.34(c)	Physical Security Plan	Y
10 CFR 50.34(d)	Safeguards Contingency Plan	Y
10 CFR 50.34(e)	Protection against unauthorized disclosure	Y
10 CFR 50.34(f)	TMI requirements	N
10 CFR 50.34(g)	Combustible Gas Control	*
10 CFR 50.34(h)	Conformance with SRP	N
10 CFR50.34(i)	Loss of Large Area of Plant	Y
10 CFR 50.34a	Design objectives for equipment to control releases of radioactive material in effluents	Y
10 CFR 50.36	Technical specifications	Y
10 CFR 50.43(e)(1)	Additional standards and provisions affecting class 103 licenses and certifications for commercial power	Y
10 CFR 50.43(e)(2)	Additional standards and provisions affecting class 103 licenses and certifications for commercial power	Y
10 CFR 50.44(d)	Combustible gas control for nuclear power reactors	Y
10 CFR 50.45	Standards for construction permits, operating licenses, and combined licenses	Y
10 CFR 50.46	Acceptance criteria for emergency core cooling systems	N
10 CFR 50.46a	Acceptance criteria for reactor coolant system venting systems	*
10 CFR 50.46a(a)	Acceptance criteria for reactor coolant system venting systems	*
10CFR 50.46a(b)	Acceptance criteria for reactor coolant system venting systems	*

Regulation	Topic	Presumed applicability to non-LWRs
10 CFR 50.46a(c)	Acceptance criteria for reactor coolant system venting systems	*
10 CFR 50.47	Emergency Plans	Y
10 CFR 50.48	Fire Protection	Y
10 CFR 50.49	Environmental qualification of electric equipment important to safety for nuclear power plants	Y
10 CFR 50.50	Issuance of licenses and construction permits	Y
10 CFR 50.51	Continuation of license	Y
10 CFR 50.52	Combining licenses	Y
10 CFR 50.53	Jurisdictional limitations	Y
10 CFR 50.54	Conditions of licenses	Y, as applicable
10 CFR 50.54(a)	Quality Assurance	Y
10 CFR 50.54(j)	Reactivity manipulation	Y
10 CFR 50.54(k)	Operator at the controls	Y
10 CFR 50.54(m)	Staffing requirements	Y
10 CFR 50.54(o)	Primary containment/App J applicability	N
10 CFR 50.54(ff)	Seismic	Y
10 CFR 50.54(hh)	Aircraft Impact	Y
10 CFR 50.55	Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses	Y
10 CFR 50.55a	Codes and Standards	Y, as applicable
10 CFR 50.55a(a)(1)(i)	Codes and Standards	N

Regulation	Topic	Presumed applicability to non-LWRs
10 CFR 50.55a(a)(1)(ii)	Codes and Standards	N
10 CFR 50.55a(a)(1)(iii)	Codes and Standards	N
10 CFR 50.55a(a)(1)(iv)	Codes and Standards	N
10 CFR 50.55a(a)(2)	Codes and Standards	Y
10 CFR 50.55a(h)(3)	Codes and Standards	Y
10 CFR 50.55a(z)	Codes and Standards	Y
10 CFR 50.56	Conversion of construction permit to license; or amendment of license	Y
10 CFR 50.57	Issuance of operating license	Y
10 CFR 50.58	Hearings and report of the Advisory Committee on Reactor Safeguards	Y
10 CFR 50.59	Changes, tests and experiments	Y
10 CFR 50.60	Acceptance criteria for fracture prevention measures for LWRs for normal operation	N
10 CFR 50.61	Fracture toughness requirements for protection against pressurized thermal shock events	N
10 CFR 50.61a	Alternate fracture toughness requirements for protection against pressurized thermal shock events	N
10 CFR 50.62	Requirements for reduction of risk from ATWS events for LWRs	N
10 CFR 50.63	Loss of all alternating current power	N
10 CFR 50.64	Limitations on the use of HEU in domestic non-power reactors	Y
10 CFR 50.65	Maintenance rule	Y
10 CFR 50.66	Requirements for thermal annealing of the reactor pressure vessel	N

Regulation	Topic	Presumed applicability to non-LWRs
10 CFR 50.67	Accident source term	Y
10 CFR 50.68	Criticality accident requirements	*
10 CFR 50.69	Risk-informed categorization and treatment of SSCs	Y
10 CFR 50.70	Inspections	Y
10 CFR 50.71	Maintenance of records, making of reports	Y
10 CFR 50.71(h)(1)	PRA	Y
10 CFR 50.72	Immediate notification requirements for operating nuclear power reactors	Y
10 CFR 50.73	Licensee event report system	Y
10 CFR 50.74	Notification of change in operator or senior operator status	Y
10 CFR 50.75	Reporting and recordkeeping for decommissioning planning	Y
10 CFR 50.76	Licensee's change of status; financial qualifications	Y
10 CFR 50.78	Facility information and verification	Y
10 CFR 50.80	Transfer of licenses	Y
10 CFR 50.81	Creditor regulations	Y
10 CFR 50.82	Termination of license	Y
10 CFR 50.83	Release of part of a power reactor facility or site for unrestricted use	Y
10 CFR 50.90	Application for amendment of license, construction permit, or early site permit	Y
10 CFR 50.91	Notice for public comment; State consultation	Y
10 CFR 50.92	Issuance of amendment	Y

Regulation	Topic	Presumed applicability to non-LWRs
10 CFR 50.100	Revocation, suspension, modification of licenses, permits, and approvals for cause	Y
10 CFR 50.101	Retaking possession of special nuclear material	Y
10 CFR 50.102	Commission order for operation after revocation	Y
10 CFR 50.103	Suspension and operation in war or national emergency	Y
10 CFR 50.109	Backfitting	Y
10 CFR 50.110	Violations	Y
10 CFR 50.111	Criminal penalties	Y
10 CFR 50.120	Training and qualification of nuclear power plant personnel	Y
10 CFR 50.150	Aircraft Impact	Y
10 CFR Part 50 Appendix A	General Design Criteria	N
10 CFR Part 50 Appendix B	Quality Assurance	Y
10 CFR Part 50 Appendix C	Financial Data and Qualifications	Y
10 CFR Part 50 Appendix E	Emergency Planning	Y
10 CFR Part 50 Appendix F	Fuel Reprocessing Plants and Related Waste Management Facilities	N
10 CFR Part 50 Appendix G	Fracture toughness requirements	N
10 CFR Part 50 Appendix H	Reactor Vessel Material Surveillance Program Requirements	N
10 CFR Part50 Appendix I	ALARA	Y, in part
10 CFR Part 50 Appendix J	Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors	N

Regulation	Topic	Presumed applicability to non-LWRs
10 CFR Part 50 Appendix K	ECCS Evaluation Models	N
10 CFR Part 50 Appendix N	Standardization of Nuclear Power Plant Designs	Y
10 CFR Part 50 Appendix Q	Pre-Application Early Review of Site Suitability Issues	Y
10 CFR Part 50 Appendix R	Fire Protection	N
10 CFR Part 50 Appendix S	Earthquake Engineering Criteria	Y

Many of the 10 CFR Part 52 requirements associated with design certifications, combined licenses, standard design approvals, and manufacturing licenses are similar. In order to reduce repetition, Table 3 provides the anticipated applicability for a selection 10 CFR Part 52 requirements. Specifically, the regulations listed in the table below reference the regulations for 10 CFR Part 52, Subpart B, “Design Certifications.” Similar and additional requirements may exist for combined licenses, standard design approvals, and manufacturing licenses. Table 3 also does not address applicable Part 50 requirements, as these are discussed in the tables above.

Table 3 – Selected 10 CFR Part 52 Requirements, as applicable to non-LWR DC applications

Regulation	Topic	Presumed applicability to non-LWRs
10 CFR 52.47(a)(4)	Analysis of SSCs and ECCS Evaluation	Y, in part
10 CFR 52.47(a)(9)	Applicability of SRP	N
10 CFR 52.47(a)(12)	Combustible gas control	Y
10 CFR 52.47(a)(14)	Pressurized Thermal Shock	N
10 CFR 52.47(a)(15)	ATWS	N
10 CFR 52.47(a)(16)	Loss of all alternating current power	N
10 CFR 52.47(a)(17)	Criticality accident requirements	*
10 CFR 52.47(a)(18)	Fire protection	Y, in part
10 CFR 52.47(a)(21)	USI resolution	As applicable
10 CFR 52.47(a)(22)	Operating Experience	Y
10 CFR 52.47(a)(23)	Severe accident considerations	N
10 CFR 52.47(a)(24)	Conceptual design information not part of the certification	Y
10 CFR 52.47(a)(25), (26)	Interface requirements met that are not part of the certification	Y
10 CFR 52.47(a)(27)	PRA	Y
10 CFR 52.47(b)(1)	ITAAC	Y

Regulation	Topic	Presumed applicability to non-LWRs
10 CFR 52.47(b)(2)	Environmental report	Y
10 CFR 52.47(c)(2)	Designs that differ significantly from LWRs must meet 50.43(e)	Y

Table 4 - Other regulations (excluding 10 CFR Parts 50 and 52) that may apply to non-LWRs at some stage in CP/OL/DC/COL/SDA/ML licensing:

Regulation	Topic	Presumed applicability to non-LWRs
10 CFR Part 2	Agency Rules of Practice and Procedure	Y
10 CFR Part 9	Public Records	Y
10 CFR Part 11	Criteria and procedures for determining eligibility for access to restricted data or national security information or an employment clearance	Y
10 CFR Part 19	Notices, instructions and reports to workers: inspection and investigations	Y
10 CFR Part 20	Standards for protection against ionizing radiation	Y
10 CFR 20.1406	Minimization of Contamination	Y
10 CFR Part 21	Reporting of defects and noncompliance	Y
10 CFR Part 25	Access authorization	Y
10 CFR Part 26	Fitness for duty programs	Y
10 CFR Part 30	Rules of general applicability to domestic licensing of byproduct material	Y
10 CFR Part 31	General domestic licenses for byproduct material	Y
10 CFR Part 37	Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material	Y
10 CFR Part 40	Domestic licensing of source material	Y
10 CFR Part 51	Environmental protection regulations for domestic licensing and related regulatory functions	Y
10 CFR Part 54	Requirements for renewal of operating licenses for nuclear power plants	Y
10 CFR Part 55	Operator's licenses	Y

Regulation	Topic	Presumed applicability to non-LWRs
10 CFR Part 70	Domestic licensing of special nuclear material	Y
10 CFR Part 71	Packaging and transportation of radioactive material	Y
10 CFR Part 72	Licensing requirements for the independent storage of spent nuclear fuel and high-level radioactive waste, and reactor-related greater than Class C waste	Y
10 CFR Part 73	Physical protection of plants and materials	Y, as applicable
10 CFR Part 74	Material control and accounting of special nuclear material	Y
10 CFR Part 81	Standard specifications for the granting of patent licenses	Y
10 CFR Part 95	Facility security clearance and safeguarding of national security information and restricted data	Y
10 CFR Part 100	Reactor site criteria	Y
10 CFR Part 110	Export and import of nuclear equipment and material	Y
10 CFR Part 140	Financial protection requirements and indemnity agreements	Y
10 CFR Part 170	Fees for facilities, materials, import and export licenses, and other regulatory services under the Atomic Energy Act of1954, as amended	Y
10 CFR Part 171	Annual fees for reactor licenses	Y