This draft *Federal Register* notice contains the latest draft proposed rule language that the NRC staff has publicly released to support interactions with the Advisory Committee on Reactor Safeguards (ACRS). This version is based on reviews by NRC staff and consideration of stakeholder input. The NRC staff expects to adopt further changes in the draft proposed rule language.

This language has not been subject to complete NRC management or legal review, and its contents should not be interpreted as official agency positions. The NRC staff plans to continue working on the draft proposed rule language provided in this document.

Please note that blue text indicates conforming changes to existing rule language in parts other than Part 53.

[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Parts 1, 2, 10, 19, 20, 21, 25, 26, 30, 40, 50, 51, 52, 53, 70, 72,73, 74, 95, 140, 150, 170, and 171

[NRC- 2019-0062]

RIN 3150-AK31

Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is proposing to revise the NRC's regulations by adding a risk-informed, performance-based, and technology-inclusive regulatory framework for commercial nuclear plants in response to the Nuclear Energy Innovation and Modernization Act (NEIMA). The NRC plans to hold

a public meeting to promote full understanding of the proposed rule and facilitate public comments.

DATES: Submit comments by **[INSERT DATE 60 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. Comments received after this date will be considered if it is practical to do so, but the NRC is able to ensure consideration only for comments received before this date. A public meeting will be held on **<INSERT:** Date>.

ADDRESSES: You may submit comments by any of the following methods (unless this document describes a different method for submitting comments on a specific subject):

- **Federal Rulemaking website:** Go to https://www.regulations.gov and search for Docket ID NRC-2019-0062. Address questions about NRC dockets to Dawn Forder; telephone: 301-415-3407; e-mail: Dawn.Forder@nrc.gov. For technical questions contact the individuals listed in the FOR FURTHER INFORMATION CONTACT section of this document.
- **E-mail comments to:** Rulemaking.Comments@nrc.gov. If you do not receive an automatic e-mail reply confirming receipt, then contact us at 301-415-1677.
- **Fax comments to:** Secretary, U.S. NRC at 301-415-1101.
- **Mail comments to:** Secretary, U.S. NRC, Washington, DC 20555-0001, ATTN: Rulemakings and Adjudications Staff.
- **Hand deliver comments to:** 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. (Eastern Time) Federal workdays; telephone: 301-415-1677.

For additional direction on obtaining information and submitting comments, see "Obtaining Information and Submitting Comments" in the SUPPLEMENTARY INFORMATION section of this document.

FOR FURTHER INFORMATION CONTACT: Robert Beall, Office of Nuclear Material Safety and Safeguards, telephone: 301-415-3874; email: Robert.Beall@nrc.gov; or Jordan Hoellman, Office of Nuclear Reactor Regulation, telephone: 301-415-5481; email: Jordan.Hoellman2@nrc.gov. Both are staff of the U.S. NRC, Washington, DC 20555-0001.

SUPPLEMENTARY INFORMATION:

EXECUTIVE SUMMARY:

A. Need for the Regulatory Action

On January 14, 2019, the President signed the NEIMA into law (Pub. L. 115 439). NEIMA section 103(a)(4), directs the NRC to "complete a rulemaking to establish a technology-inclusive, regulatory framework for optional use by commercial advanced nuclear reactor applicants for new reactor license applications." NEIMA defines a "technology-inclusive, regulatory framework" as one that is "developed using methods of evaluation that are flexible and practicable for application to a variety of reactor technologies, including, where appropriate, the use of risk-informed and performance-based techniques."

The current application and licensing requirements in part 50, "Domestic Licensing of Production and Utilization Facilities," of title 10 of the *Code of Federal Regulations* (10 CFR) and 10 CFR part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," were developed over time, and in some cases include requirements specific to large light-water reactors (LLWR) and non-power reactors similar to those for which licenses and permits have been issued by the Commission in the past. These regulations do not fully reflect the range of licensing and regulatory challenges associated with other nuclear reactor technologies. This rulemaking will respond to NEIMA and amend 10 CFR by creating an alternative regulatory framework for licensing future commercial nuclear plants. The new alternative requirements and implementing guidance would adopt technology-inclusive approaches, and include the appropriate use of risk-informed and performance-based techniques, to provide the necessary flexibility for licensing and regulating a variety of technologies and designs for commercial nuclear reactors, including advanced nuclear reactors.

B. Major Provisions

Major provisions of this proposed rule, supported by accompanying guidance, include the addition of the following:

- A new alternative risk-informed, performance-based framework referred to as "Framework A," which includes requirements for licensing and regulating future commercial nuclear plants during the various stages of their life cycle;
- A new technology-inclusive framework referred to as "Framework B," which is similar to the traditional approach to the design and licensing of existing light-water reactors (LWRs) in parts 50 and 52.
- Part 26 placeholder.
- Part 73 placeholder.

C. Costs and Benefits

The NRC prepared a draft regulatory analysis to determine the expected quantitative costs and benefits of this proposed rule and associated guidance as well as qualitative factors to be considered in the NRC's rulemaking decision. The conclusion from the analysis is that this proposed rule and associated guidance would result in net averted costs to the industry and the NRC ranging from $XX million using a 7-percent discount rate to $XX million using a 3-percent discount rate.

The draft regulatory analysis also considered qualitative aspects, such as greater regulatory stability, predictability, and clarity to the licensing process. These benefits would result from XXXX. Another qualitative consideration is promoting a performance-based regulatory framework that specifies requirements to be met and provides flexibility to an applicant or licensee regarding the information or approach needed to satisfy those requirements.

For more information, please see the draft regulatory analysis (available in the NRC's Agencywide Documents Access and Management System (ADAMS) Accession No. MLXXXXXXXXX).

TABLE OF CONTENTS:

I. Obtaining Information and Submitting Comments

A. Obtaining Information

Please refer to Docket ID NRC-2019-0062 when contacting the NRC about the availability of information for this action. You may obtain publicly available information related to this action by any of the following methods:

- **Federal Rulemaking website:** Go to https://www.regulations.gov and search for Docket ID NRC- 2019-0062.

- **NRC's ADAMS:**

You may obtain publicly available documents online in the ADAMS Public Documents collection at https://www.nrc.gov/reading-rm/adams.html. To begin the search, select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. For the convenience of the reader, instructions about obtaining materials referenced in this document are provided in section XVII, "Availability of Documents."

- **NRC's PDR:** You may examine and purchase copies of public documents at the NRC's PDR, Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

B. Submitting Comments

Please include **Docket ID NRC- 2019-0062** in your comment submission. To facilitate NRC review, please distinguish your comments between comments on the proposed rule and comments on the proposed guidance. The NRC cautions you not to include identifying or contact information that you do not want to be publicly disclosed in your comment submission. The NRC will post all comment submissions at https://www.regulations.gov as well as enter the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment into ADAMS.

II. Background

A. NRC Advanced Reactor Readiness

The NRC has been developing licensing approaches for advanced reactors since the Policy Statement on the Regulation of Advanced Reactors, published on July 8, 1986 (51 FR 24643), which the NRC updated on May 9, 2008 (73 FR 26349). The agency's activities related to advanced reactors, including an advance notice of proposed rulemaking titled, "Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors," dated May 4, 2006 (71 FR 26267), were often done in parallel, and sometimes interwoven, with the NRC's efforts to improve risk-informed and performance-based approaches within the agency (e.g., the Commission's policy statement, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," published on August 16, 1995 (60 FR 42622)).

In 2016, the NRC issued its "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light-Water Mission Readiness" (Advanced Reactor Vision and Strategy Document) (ADAMS Accession No. ML16356A670), in response to increasing public interest in advanced reactor designs, including possible legislation then under consideration in Congress. The NRC considered the DOE's advanced reactor deployment goals in the 2016 timeframe and continues to reassess NRC activities to support the DOE's deployment goals. The NRC identified the potential need to initiate and develop a new risk-informed, performance-based, and technology-inclusive regulatory framework. The NRC staff's initial efforts were focused on resolving policy issues and developing guidance for licensing non-LWRs reactor technologies under the existing regulatory frameworks (parts 50 and 52). The NRC staff issues annual Commission papers on the status and progress of the NRC staff's activities related to advanced reactors, including the progress and path forward on each of the related projects (e.g., SECY-22-0008, "Advanced Reactor Program Status," on January 31, 2022 (ADAMS Accession No. ML21337A377)). In 2017, the NRC staff prioritized activities to support the development of technology-inclusive, risk-informed, and performance-based licensing approaches that could be implemented under the existing regulatory framework in parts 50 and 52. One key element of these efforts was the LMP, a cost-shared initiative led by nuclear utilities and supported by DOE. The LMP is a technology-inclusive, risk-informed, and performance-based methodology developed for non-LWR designs that provides a systematic and reproducible process for licensing-basis event (LBE) selection and evaluation, classification of structures, systems, and components (SSCs), and assessment of defense in depth. The LMP refined the DOE's Next Generation Nuclear Plant Program methodologies to reflect interactions with the NRC, to address feedback from industry, and to broaden the scope of the approach to ensure applicability to various non-LWR technologies. The LMP activities led to the publication and submittal of NEI 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," issued August 2019 (ADAMS Package Accession No. ML19241A336).

The NRC endorsed the principles and methodology in Nuclear Energy Institute (NEI) 18-04, with clarifications, in Regulatory Guide (RG) 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and

Approvals for Non-Light-Water Reactors," dated June 2020 (ADAMS Accession No. ML20091L698). The NRC staff described the methodology and its relationship with previous relevant Commission decisions in SECY-19-0117, "Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," dated December 2, 2019 (ADAMS Package Accession No. ML18311A264). These previous Commission decisions include those described 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," dated April 8, 1993 (ADAMS Accession No. ML040210725). The Commission approved the use of the methodology as a reasonable approach for establishing key parts of the licensing basis and content of applications for licenses, certifications, and approvals for non-LWRs in SRM-SECY-19-0117, dated May 26, 2020 (ADAMS Accession No. ML20147A504). While the LMP approach is technology inclusive, the industry and NRC staff limited its initial scope to non-LWRs for efficiency and to issue robust guidance to support near-term applications under the existing regulatory framework, such as the Advanced Reactor Demonstration Projects supported by DOE.

As stated in the part 53 rulemaking plan, SECY-20-0032 (ADAMS Accession No. ML19340A056), the NRC staff developed part 53, and specifically Framework A, building upon recent and ongoing activities such as those described in SECY-19-0117. Such an approach supports implementing the NEIMA requirement to use, where appropriate, risk-informed and performance-based techniques and also capitalizes on previous initiatives by the industry, DOE, and the NRC. Moreover, this approach is consistent with the Commission's direction in the SRM for SECY-10-0121, "Modifying the Risk-Informed

Regulatory Guidance for New Reactors," dated March 2, 2011 (ADAMS Accession No. ML110610166). In that SRM, the Commission stated that "[n]ew reactors with these enhanced margins and safety features should have greater operational flexibility than current reactors. This flexibility will provide for more efficient use of NRC resources and allow a fuller focus on issues of true safety significance." The methodology described in SECY-19-0117, which in turn describes the methodology of the LMP, furthers the expectations described in SECY-10-0121. The methodology in SECY-19-0117 includes using risk-informed safety analyses to identify margins between estimated risks posed by a proposed nuclear plant and established performance standards, including the NRC's safety goals. Both SECY-10-0121 and SECY-19-0117 and the related Commission decisions highlight the role of probabilistic risk assessment (PRA) in risk-informed and performance-based approaches to identifying enhanced safety margins that can be used to justify operational flexibilities. The proposed Framework A is largely based on the methodology described in SECY-19-0117 and includes a prominent role for PRA.

As discussed in section B of this section, the NRC conducted extensive public outreach on early versions of the rule text. Those early versions provided an approach for licensing, which formed the basis for what would be Framework A in the proposed rule. However, commenters indicated that some designers may find the role of PRA contemplated by Framework A unduly restrictive because of the simplified scope of their designs or because their business plans contemplated marketing in countries that would require a different, more deterministic, safety analysis. In light of these concerns, the NRC also developed an alternate approach to licensing, which would be Framework B in the proposed rule. Framework B largely replicates yet modifies the existing licensing approach in parts 50 and 52 to be technology neutral. While including consideration of insights from risk assessments, Framework B does not require a PRA to be used to the extent contemplated by Framework A. Instead Framework B would require applicants to use risk insights from a PRA, or alternative evaluation for risk insights (AERI), in a confirmatory role to the largely deterministic safety analysis and as a possible tool to identify safety margins to justify operational flexibilities. This approach is consistent with how part 52 currently utilizes risk insights. Additionally, the approach to licensing in Framework B, which would require applicants to develop and use principal design criteria similar to appendix A in part 50, coincides with existing international standards for designing and licensing advanced reactors.

B. Public Comments on Part 53 Preliminary Proposed Rule Language

On November 6, 2020, the NRC published a *Federal Register* notice (85 FR 71002) describing plans for the periodic release of preliminary proposed rule language, meetings with stakeholders, and the ability of stakeholders to provide input during the development of this proposed rule. Sections of the preliminary proposed rule language were subsequently iteratively released, and the NRC held numerous public meetings to discuss the preliminary language and obtain input from stakeholders. On December 10, 2021, the NRC published a second *Federal Register* notice (86 FR 70423) announcing that the development of the proposed rule and related interactions with stakeholders were being extended until August 31, 2022.

By the close of the public stakeholder interactions on August 31, 2022, the NRC staff had held 17 public meetings since September 2020. The NRC staff also held 16 public meetings with the Advisory Committee on Reactor Safeguards (ACRS). By the close of the public comment period on the preliminary proposed rule language, 126 comment letters were received on the preliminary proposed rule language. Of these 126 comment letters, 21 were from non-governmental organizations, 31 were from the public, 1 was from Congress, and the remaining 73 letters were from NRC licensees, the NEI, and other industry groups. The letters from stakeholders provided various points of view and suggestions for clarifications, additions, and deletions to the preliminary proposed rule language. Copies of these letters may be viewed and downloaded from the Federal eRulemaking Web site *http://www.regulations.gov,* under docket number NRC–2019–0062. The inputs received were considered in the development of this proposed rule. However, as described during the various public interactions related to this rulemaking and in supporting documents, the NRC will not formally disposition the questions and suggestions related to the preliminary language as will be done for public comments received following the publication of this proposed rule.

III. Discussion

A. Objective and Applicability

The NRC is proposing to add a new, alternative part to its regulations that would set out risk-informed, technology-inclusive frameworks for the licensing and regulation of commercial nuclear plants. These new approaches would: (1) continue to provide reasonable assurance of adequate protection of public health and safety and the common defense and security; (2) promote regulatory stability, predictability, and clarity; (3) reduce requests for exemptions from the current requirements in parts 50 and 52; (4) establish new requirements to address non-LWR technologies; (5) recognize technological advancements in reactor design; and (6) credit the possible response of some designs of commercial nuclear plants to postulated accidents, including slower transient response times and relatively small and slow release of fission products. The proposed rule would add 10 CFR part 53, “Risk-Informed, Technology-Inclusive Regulatory Frameworks for Commercial Nuclear Plants,” part 26, subpart M, “FFD Programs for Facilities Licensed Under Part 53,” § 73.100, “Technology-inclusive requirements for physical protection of licensed activities at commercial nuclear plants against radiological sabotage,” § 73.110, “Technology-inclusive requirements for protection of digital computer and communication systems and networks,” and § 73.120, “Access authorization,” and make conforming changes throughout 10 CFR chapter I.

B. Need for Changes to the Existing Regulatory Framework

The NRC has long recognized that the licensing and regulation of a variety of nuclear reactor technologies would present challenges given the existing regulatory framework has evolved primarily to address the LLWR designs that comprise the current operating fleet. The NRC has had many interactions with designers of various new reactor technologies, commonly referred to as advanced reactors, and developed policies and guidance to support the potential licensing of new and different types of reactor facilities some of which may not utilize LLWR designs. The NRC first published its policy statement on the regulation of advanced reactors in the *Federal Register* on July 8, 1986 (51 FR 24643), with an objective of providing all interested parties, including the public, with the Commission’s views concerning the desired characteristics of advanced reactor designs. The NRC described its early efforts to establish a technology-inclusive approach to the regulation of nuclear reactors in an advance notice of proposed rulemaking titled, “Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors,” dated May 4, 2006 (71 FR 26267). The

NRC acknowledged in its “Report to Congress: Advanced Reactor Licensing,” issued August 2012, that while the safety philosophy inherent in the current regulations applies to all reactor technologies, the specific and prescriptive aspects of those regulations clearly focus on the current fleet of LLWR facilities. The NRC’s 2016 Vision and Strategy report for non-LWRs identified a potential rulemaking to establish a risk-informed, performance-based, and technology-inclusive regulatory framework for advanced nuclear reactor licensing.

Congress similarly recognized the potential benefits of developing a regulatory infrastructure to support the development and commercialization of advanced nuclear reactors. Consequently, Congress passed NEIMA in late 2018 and signed it into law in January 2019. The NEIMA directed the NRC to undertake this rulemaking, which establishes a technology-inclusive regulatory framework for optional use by applicants for new commercial nuclear reactors.

The requirements in part 53 would support the wide variety of potential commercial nuclear reactor technologies. As noted in this discussion, the current regulatory framework in parts 50 and 52 was developed for light-water reactors and includes some provisions specific to light-water reactor technologies. Although the NRC can license other reactor technologies under this framework by using existing regulatory flexibilities and the exemption process there is significant interest in developing a regulatory framework that is flexible enough to accommodate multiple technologies and robust enough to ensure a level of safety equivalent to parts 50 and 52. However, doing so presents significant regulatory challenges, particularly since part 53 has only been under development for a few years whereas the existing regulatory framework developed over the course of decades. To address these challenges, the NRC drew on well-developed approaches to licensing to produce two technology-neutral and robust regulatory frameworks, described in Sections III.C, “10 CFR Part 53 Frameworks,” and III.D, “Subpart A – General Provisions.”. In terms of providing a regulatory framework that is risk-informed and performance-based, part 53 would provide options for the roles of several risk assessment techniques and design approaches. The proposed regulatory framework that would use PRAs to assess risks, help establish technical requirements, and manage operations is referred to as “Framework A,” which would be established primarily in subparts B through K of part 53. Framework A builds on the approach to licensing described in the LMP, which was developed as a cost-shared initiative led by nuclear utilities and supported by the U.S. Department of Energy. Industry guidance developed under the LMP[1] was endorsed by the NRC for use by non-light water-reactor applicants using the licensing processes in parts 50 and 52—that guidance would not yet be made applicable to part 53 applicants and licensees.[2] Nonetheless, Framework A leverages the LMP, which is a technology-inclusive approach to licensing that leverages insights from a detailed PRA to provide applicants with significant design and operation flexibilities.

However, not all applicants plan to develop and use the PRA needed for Framework A. Consequently, the NRC also developed an alternative framework, referred to as “Framework B,” which would be established primarily in subparts N through U of part 53. This framework would include deterministic and risk-informed acceptance criteria similar to those in parts 50 and 52 but would better address a variety

[1] NEI 18-04, Revision 1, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development" ADAMS Accession No. ML 19241A472).

[2] RG 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors,” June 2020 (ADAMS Accession No. ML20091L698.)

of commercial reactor technologies that may be licensed following this rulemaking. Framework B also builds on international guidance (i.e., International Atomic Energy Agency [IAEA]) for nuclear reactor licensing.

C. 10 CFR Part 53 Frameworks

This rulemaking consists of several major components, including a new part, part 53, "Risk-Informed, Technology-Inclusive Regulatory Frameworks for Commercial Nuclear Plants," to be added to 10 CFR, revisions for parts 26 and 10 CFR part 73, "Physical Protection of Plants and Materials," and conforming changes throughout 10 CFR.

Proposed § 53.000, "Purpose," would describe the purpose of part 53 and would be equivalent to § 50.1, "Basis, purpose, and procedures applicable." Proposed § 53.010 would provide requirements for the use of either of the two optional frameworks presented throughout part 53. Proposed § 53.010 is similar to § 2.2, "Subparts," in that it would direct an applicant into the applicable subparts for each framework. Proposed subpart A and subparts B through K of part 53 would define "Framework A" and are organized to provide high-level performance criteria and to specify requirements to demonstrate compliance with those performance criteria throughout major stages of the life cycle of commercial nuclear plants. This organization supports a systems engineering type approach to the design, licensing, operation, and ultimately decommissioning of future commercial nuclear plants. Organizing requirements in this manner also supports performance-based approaches where programs and monitoring during the operations phase could be used to confirm predictions and possibly compensate for uncertainties associated with reactor technologies, materials, and other innovations that currently lack operating experience. The performance-based approach proposed in “Framework A” of part 53 also includes a flexible and graded approach to regulatory controls based on the role of a particular structure, system, and component (SSC), human action, or program in limiting the risk of an immediate threat to public health and safety or maintaining the overall risks to the public below accepted standards through balanced measures to prevent and mitigate possible events.

Proposed subpart A and subparts N through U of part 53 define “Framework B” and include technology-inclusive requirements similar to the traditional requirements in parts 50 and 52 that were developed primarily for LWR designs. Framework B would maintain the traditional role of specific design rules, including use of the single failure criterion as a tool in the reactor safety review process, and deterministic approaches to define LBEs and performance requirements for SSCs. The traditional approach in Framework B of this part would require applicants to define principal design criteria similar to those required by appendix A to part 50 for LWRs and presented in RG 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors” for select non-LWR technologies.

Proposed subpart A is the only proposed subpart entirely common to both frameworks. Several subparts within proposed part 53 include proposed requirements that would be equivalent under either framework. In these cases, the proposed requirements in Framework B generally reproduce the proposed regulations from Framework A, with appropriate differences to account for differences in the frameworks, including framework-specific references. Specifically, subpart O largely reproduces subpart E, subpart P largely reproduces subpart F, subpart Q largely reproduces subpart G, subpart S largely reproduces subpart I, subpart T largely reproduces subpart J, and subpart U largely reproduces subpart K. Subpart F contains several sections that apply to both frameworks rather than reproducing those sections in each framework. In contrast, subparts B, C, D, and H in Framework A and subparts N and R in Framework B are largely unique to each framework.

[Parts 26 and 73 place holder.]

In addition, this proposed rule would make conforming changes throughout 10 CFR chapter I, such as adding "and part 53" where appropriate to account for the addition of the proposed part 53.

D. Subpart A – General Provisions

Subpart A would provide the general provisions applicable to all applicants and licensees under either of the optional frameworks (Framework A and Framework B) that would be established in part 53 for the issuance, amendment, and termination of licenses, permits, certifications, and approvals for commercial nuclear plants licensed under Section 103 of the Atomic Energy Act, as amended (AEA) and title II of the Energy Reorganization Act of 1974 (88 Stat. 1242). Subpart A would include purpose, scope, definitions, written communications, employee protections, completeness and accuracy of information, exemptions, standards for review, jurisdictional limits, attacks and destructive acts, and information collection requirements.

The requirements in subpart A would be largely equivalent to the general requirements in part 50 that are applicable to all part 50 applicants and licensees (specifically, §§ 50.1 through 50.13) but would reference the corresponding regulations in part 53 in place of references to part 50.

Definitions

Due to fundamental differences in the methodologies in Frameworks A and B, as described in section III.C.1 of this preamble, the proposed rule would include three definition sections in §§ 53.020, 53.024, and 53.028 to distinguish among terms common to both frameworks, terms specific to Framework A, and terms specific to Framework B, respectively. Section 53.020 would define terms common to both frameworks and would include terms such as: commercial nuclear plant, commercial nuclear reactor, defense in depth, design features, event sequence, licensing basis information, manufactured reactor, manufactured reactor module, normal operation, PRA, quality assurance, safety function, and site characteristics. In general, the definitions of most of these terms in § 53.020 would be technically equivalent to the corresponding terms defined in §§ 50.2, 52.1, and other existing regulatory definitions; NEI 18-04, as endorsed by RG 1.233; or American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Risk Assessment Standard (RA-S)-1.4-2021, as proposed to be endorsed by RG 1.247, or would be used consistent with how the terms are used under the existing regulatory framework. This is intended to provide clarity and consistency in terminology among all licensing frameworks where possible and to utilize past and ongoing NRC initiatives to support the licensing of new reactors. Specific deviations from existing definitions are further explained in the following paragraphs.

Regarding the definition of “commercial nuclear plant” and “commercial nuclear reactor” in proposed § 53.020, the NRC initially considered establishing the scope of part 53 as being for “advanced nuclear plants,” as indicated in preliminary proposed rule language released for discussion. The preliminary proposed rule language defined “advanced nuclear plant” as “a utilization facility consisting of one or more advanced nuclear reactors [as defined in NEIMA] and associated co-located support facilities, which may include one or more reactor modules, [using nuclear fission, nuclear fusion, or accelerator-driven reactor technologies] that are used for producing power for commercial electric or other commercial purposes.” The NRC considered the feedback received and determined that the NEIMA definition would be difficult for the NRC to implement. Additionally, there would be no technical reason to so limit the applicability of part 53, because the rule would provide at least an equivalent level of safety compared to licensing under parts 50 and 52, consistent with Commission policy, for all reactor technologies. Instead, to be technology inclusive, the NRC proposes to allow use of part 53 by any “commercial nuclear plant,” which is defined in the proposed rule to mean “a facility consisting of one or more commercial nuclear reactors and associated co-located support facilities, including the collection of buildings, radionuclide sources, and SSCs for which a license(s) is being sought under part 53, that are used for producing power for commercial electric power or other commercial purposes.” The phrase “commercial purposes” as used in this definition includes commercial purposes such as providing process heat for a variety of industrial applications (e.g., desalination, oil refining, hydrogen production).” The definition of “commercial nuclear plant” in turn refers to a “commercial nuclear reactor,” which is defined based on the definition of “nuclear reactor” in § 50.2, with the exception that the phrase “in a self-supporting chain reaction” was removed to not preclude the applicability of part 53 to future accelerator driven systems. Relatedly, “utilization facility” is also defined in § 53.020 based on the definition of that term in § 50.2 but is also revised to refer to a “commercial nuclear plant” as defined in § 53.020.

In part 53, the NRC proposes to define the term “manufactured reactor” to mean “the essential portions of a nuclear reactor that are manufactured under a ML and subsequently transported and incorporated into a commercial nuclear plant under a combined license.” The term “manufactured reactor module” would “mean a manufactured reactor loaded with fuel prior to transport to a licensed location for installation and commercial operation." These definitions are provided because the NRC recognizes that some future reactor deployment models include the concept of fabricating a number of small reactors (i.e., microreactors) in a single facility prior to their deployment at a site under a combined license (COL). The NRC also recognizes that some of these deployment strategies include loading fuel at the manufacturing facility, which these definitions intend to accommodate provided that the module is configured with at least two independent mechanisms, each of which is sufficient to prevent criticality during its loading and storage. The storage, movement, and loading of fuel into the manufactured reactor module within the manufacturing facility would need to comply with the requirements of part 70. Any such manufactured reactor module in which these mechanisms have been installed would not be considered a utilization facility as defined in section 11cc. of the AEA or § 53.020 until it is installed in its final place of use and the NRC has found that both the inspections, tests, analyses, and acceptance criteria (ITAAC) in the manufacturing license (ML) and the ITAAC in the COL that authorized reactor construction are met.

In the preliminary proposed rule language, the NRC initially included a requirement establishing "safety functions," in subpart B that would require applicants under Framework A to define additional safety functions supporting the primary safety function of limiting the release of radioactive materials during LBEs. Therefore, NRC did not originally include a definition of "safety functions" for Framework A in the preliminary proposed rule language. Throughout the development of Framework B, the NRC did not include a requirement to establish safety functions because they are implicitly captured through the requirements for principal design criteria. The NRC presented this proposal during public meetings and ACRS briefings, which generated significant discussion about the differences between the frameworks with respect to "safety functions." The NRC considered the feedback received and included in the proposed rule a definition of safety function applicable to both frameworks in proposed § 53.020 to align the two frameworks so that safety functions are technology-inclusive requirements. For this reason, "safety function" would mean, in part, "a purpose served by a design feature, human action, or programmatic control to prevent or mitigate unplanned events and thereby satisfy requirements in part 53 for limiting risks to public health and safety." As discussed in section IV of this document for Framework A, subpart B, § 53.230 would establish a "primary" safety function and then would require applicants to define additional safety functions necessary to retain radioactive materials during LBEs. It would then state that safety functions are required to demonstrate compliance with the safety criteria in subpart B. Safety functions can be performed by any combination of design features, human actions, or programmatic control and can be specified at the plant level or at the level of a particular barrier or system. Plant-level safety functions, such as reactivity control, fluid (heat removal) systems, and reactor containment, are addressed under the existing regulatory framework for LWRs through the organization of the general design criteria (GDC) in appendix A to part 50. A barrier or system-level safety function could be a specific design criterion stating that a containment heat removal system must be provided to maintain containment pressure and temperature within acceptable limits.

The NRC proposes to include a definition of "consensus code or standard" in part 53 that is based on the use of these terms in the National Technology Transfer and Advancement Act of 1995 (NTTAA) (Public Law 104-113) and the Office of Management and Budget (OMB) Circular No. A-119, “Federal Participation in the Development and Use of Voluntary Consensus Standards and in Conformity Assessment Activities.” As required by NTTAA, the NRC: (i) consults with voluntary consensus standards bodies; (ii) participates with voluntary consensus bodies in the development of consensus standards; and (iii) uses consensus standards as a means to carry out NRC’s policy objectives. In part 53, the NRC is not proposing to incorporate by reference specific codes and standards as is done under the existing regulations in § 50.55a, “Codes and standards,” because some codes and standards are light-water-reactor-specific. Part 53 would require that design features must be designed using generally accepted consensus codes and standards but would not incorporate the specific code or standard into the NRC’s regulations. During public meetings, significant discussions with stakeholders indicated that future reactor designers were interested in the use of international consensus standards that have not yet been endorsed by the NRC. The definition proposed in part 53 would allow for the use of international codes and standards not previously used in NRC licensing but recognizes that the use of any consensus code or standard would ultimately need to be found acceptable by the NRC through either generic efforts to endorse a code or standard or on an application-specific basis during an individual licensing review.

The framework-specific definitions in §§ 53.024 and 53.028 of subpart A would define terms that are either applicable under only one of the frameworks or defined differently in Frameworks A and B based on the use of a risk-informed, performance-based methodology in Framework A compared to the traditional licensing approach used in Framework B. In general, this would include terms related to event selection and identification, equipment classification, and the way special treatment is applied to equipment identified as risk- or safety-significant. The proposed definitions of "construction" are also framework-specific – they would cover the same concept but be applied to a slightly different scope of activities based on how structures, systems and components (SSCs) are classified under each framework. In Framework A, the definition of "construction" is based on the definition in § 50.10 but modified to apply to safety-related and non-safety-related but safety-significant (NSRSS) SSCs identified by the analysis requirements to ensure the safety criteria are met. In Framework B, the definition of "construction" is equivalent to the definition in § 50.10.

Definitions Applicable to Framework A

Section 53.024 would add definitions that would be applicable to Framework A, such as terms related to event selection (LBEs, design-basis accidents [DBAs], anticipated event sequences, unlikely event sequences, and very unlikely event sequences); equipment classifications (safety-related, NSRSS, and non-safety-significant SSCs); performance metrics (safety criteria and functional design criteria); and special treatment.

The regulation would define "safety criteria" in terms of the performance-based metrics that would be provided in §§ 53.210 and 53.220. The term "functional design criteria" would be defined as metrics for the performance of SSCs. These are new terms that have not previously been defined or used in NRC regulation.

The term "safety-related SSCs" would refer to "those SSCs that are relied upon to demonstrate compliance with the safety criteria in § 53.210 and warrant special treatment." The term "non-safety-related but safety-significant SSCs" would mean those SSCs that are not safety-related because they do not perform any function necessary to demonstrate compliance with § 53.210 but warrant special treatment because they are relied on to achieve adequate defense in depth or perform risk-significant functions. The term “special treatment” would be defined as requirements, such as quality assurance and programmatic controls identified for each design feature to ensure that the safety criteria are satisfied and the safety functions are fulfilled. These requirements would also ensure that safety-related and NSRSS SSCs will provide defense in depth, or perform risk-significant functions under the service conditions and with the reliability assumed in the analysis required in proposed subpart C. The term “non-safety-significant SSCs” would mean those SSCs that are not safety-related or NSRSS.

The term “licensing-basis events” would refer to a collection of event sequences considered in the design and licensing of a commercial nuclear plant, including anticipated event sequences, unlikely event sequences, very unlikely event sequences, and DBAs (similar to its meaning in the LMP). The terms “design-basis accidents,” “anticipated event sequences,” “unlikely event sequences,” and “very unlikely event sequences” would be defined to be different types of “licensing-basis events” and would also be largely equivalent to the LMP’s definitions of DBAs, anticipated operational occurrences, design-basis events, and beyond-design-basis events, respectively. The term “design-basis accidents” would be defined as postulated event sequences that are used to set functional design criteria and performance objectives for the design of safety-related SSCs through deterministic analyses.” DBAs would be derived from the unlikely event sequences by prescriptively assuming that only safety-related SSCs are available to mitigate postulated accident consequences. The term “anticipated event sequence” would be defined as “event sequences expected to occur one or more times during the life of a commercial nuclear plant and would take into account the expected response of all SSCs within the plant, regardless of safety classification.” Within the LMP methodology, event sequences with mean frequencies of 1×10-2/plant-year and greater would be classified as anticipated event sequences. The term “unlikely event sequences” would be defined as “event sequences that are not expected to occur in the life of a commercial nuclear plant, are less likely than anticipated event sequences, but are infrequent rather than rare. Within the LMP methodology, infrequent event sequences with mean frequencies of 1×10-4/plant-year to 1×10-2/plant-year would be classified as unlikely event sequences. ”Very unlikely event sequences” would be less likely to occur than unlikely event sequences. Within the LMP methodology, rare event sequences with frequencies of 5×10-7/plant-year to 1×10-4/plant -year would be classified as very unlikely event sequences. While the proposed terminology for these event sequences would create some differences between Framework A and the LMP, Framework A would use new terms for these event sequences specifically to avoid conflicts with terms already used within part 50 and part 52 to represent slightly different concepts. Further, some stakeholder comments demonstrated confusion related to the history of beyond-design-basis accidents terminology and these definitions seek to clarify the event categories in Framework A. The sections of this preamble related to subparts B and C provide additional discussion of licensing-basis events.

Definitions Applicable to Framework B

Section 53.028 would add definitions that would be applicable to Framework B. The term “anticipated operational occurrence” would be the term for an event class unique to Framework B, within part 53. This term would be comparable to the term “anticipated event sequence” in Framework A; however, “anticipated operational occurrence” is used in Framework B to mirror the traditional licensing approach under part 50 and part 52 and to allow Framework B applicants and licensees to use certain part 50 provisions where applicable that use this event terminology. Additionally, use of “anticipated operational occurrence” in Framework B, as opposed to “anticipated event sequence,” recognizes that the frequency component of this type of event is not used in Framework B. The definition for “anticipated operational occurrence” would be equivalent to that in appendix A to part 50, with changes that would eliminate existing examples of anticipated operational occurrences that are not technology inclusive. Similarly, the definition for “construction” would be equivalent to that in § 50.10(a) with conforming changes made for cross references in Framework B. The definitions for “design bases” and “reactor coolant pressure boundary” in Framework B would be equivalent to their definitions in § 50.2.

The term “functional containment” would be defined in § 53.028 to support its use throughout Framework B by non-LWR applicants and licensees that may need or elect to use a functional containment approach in lieu of the traditional leak-tight containment structure used by LWRs. The proposed definition for functional containment is consistent with the definition for the same term from SECY-18-0096, “Functional Containment Performance Criteria for Non-Light-Water-Reactors.”

The term safety-related SSCs is used in both frameworks with different definitions. The proposed rule would provide a specific definition in § 53.028 for use of this term in Framework B. In Framework B, the term would be used in a more deterministic sense consistent with the existing regulatory frameworks in part 50 and part 52. The definition of this term has been bifurcated in Framework B to ensure technology inclusiveness and consistency with the existing regulatory frameworks. For light-water-reactors, the definition of this term would be equivalent to the definition in § 50.2. For non-LWRs, a new portion of the definition would be introduced that is broader to accommodate a wider range of technologies. This portion of the definition would denote that safety-related SSCs for non-LWRs are those SSCs that are used to mitigate the consequences of a DBA, including those SSCs that may be relied on as part of a functional containment. Additionally, the definition of safety-related SSCs is also made more technology inclusive for non-LWRs by structuring item (1) in the definition not in terms of the "reactor coolant pressure boundary," which is not fully technology inclusive, but rather in terms of "the capability to perform safety functions…including cooling to maintain the integrity of required systems and barriers."

Other General Provisions

Section 53.040 would govern written communications and how application submittals and other required submittals must be submitted to the NRC. These requirements would be equivalent to those in § 50.4.

Section 53.050 would establish requirements for deliberate misconduct and enforcement action that a licensee, an applicant, or a licensee's or applicant's contractor or subcontractor may be subject to if any one of them engages in deliberate misconduct. These requirements would be equivalent to those in § 50.5.

Section 53.060 would prohibit discrimination against an employee of a holder or applicant for an NRC license, permit, or design approval, or a contractor or subcontractor of a holder or applicant for an NRC license, permit, or design approval for engaging in certain protected activities. Section 53.060 also would prescribe a procedure for seeking a remedy for an employee who believes he or she has been discriminated against for engaging in such protected activities. These requirements would be equivalent to those in § 50.7.

Section 53.070 would govern the completeness and accuracy of information provided to the NRC. These requirements would be equivalent to those in § 50.9.

Section 53.080 would govern exemptions from the requirements of the regulations in this part. These requirements would be equivalent to those in § 50.12.

Section 53.090 paragraphs (a), (b), (c), and (d) would establish requirements for standards that the NRC would consider in determining that a construction permit (CP), operating license (OL), early site permit (ESP), COL, or ML under part 53 would be issued to an applicant. These requirements would be equivalent to those in §§ 50.40, 50.42, 50.43 and 50.22, respectively. Requirements equivalent to those in § 50.41 and 50.21 would not be included in part 53 because they apply to Class 104 licenses, and part 53 would not apply to those licenses.

Section 53.100 would require that no license issued under part 53 would cover activities which are not under or within the jurisdiction of the United States. These requirements would be equivalent to those in § 50.53.

Section 53.110 would state that licensees and applicants would not be required to provide design features or other measures for the specific purpose of protection against the effects of attacks and destructive acts directed against the facility or deployment of weapons incident to U.S. defense activities. These requirements would be equivalent to those in § 50.13.

Section 53.115 would establish requirements for rights related to special nuclear material (SNM). These requirements would be equivalent to those in § 50.54(b) and (c).

Section 53.117 would establish requirements for license suspension and rights of recapture of the material or control of the facility in a state of war or national emergency declared by Congress. These requirements would be equivalent to those in § 50.54(d).

Section 53.120 would establish requirements for information collection requirements and OMB approval. These requirements would be equivalent to those in § 50.8.

IV. Framework A

New Requirements in 10 CFR Part 53

Subpart B – Technology-Inclusive Safety Requirements

Proposed subpart B, "Technology-Inclusive Safety Requirements," would provide technology-inclusive safety criteria that would serve as performance standards for the subsequent performance-based requirements used throughout Framework A of part 53. Subsequent subparts would define how specific activities during various stages of the life cycle of a commercial nuclear plant support satisfying these high-level performance standards. The performance standards in subpart B would also establish a means to determine appropriate regulatory controls for SSCs, human actions, and programs in the following subparts in Framework A. For example, the classification of safety-related SSCs would be built upon the proposed safety criteria in § 53.210, "Safety criteria for design-basis accidents." The requirements for those SSCs would then be further defined in the design and analysis requirements in subpart C, "Design and Analysis Requirements." The activities for manufacturing, constructing, and maintaining the safety-related SSCs would be governed by subpart E, "Construction and Manufacturing Requirements," and subpart F, "Requirements for Operations." Requirements for NSRSS SSCs warranting special treatment would likewise be determined in accordance with § 53.220, "Safety criteria for licensing-basis events -basis accidents," and § 53.460, "Safety categorization and special treatment," in subpart C. Requirements related to the

NSRSS SSCs would be distinguished from the requirements for safety-related SSCs throughout Framework A within part 53, with more flexibility afforded to applicants and licensees on how NSRSS SSCs would be used in the design and maintained during plant operations.

Section 53.200 would provide the overall safety objectives for Framework A of part 53. These objectives would be to ensure the following: (1) commercial nuclear plants are designed, constructed, operated, and decommissioned to limit the possibility of an immediate threat to the public health and safety, and (2) additional measures are taken as may be appropriate when considering potential risks to public health and safety. The first safety objective would be taken, in part, from the long-standing standards used for determining the content of technical specifications (TS) (see Technical Specifications, Final Rule (60 FR 36953, 36955) [quoting the Atomic Safety and Licensing Appeal Board in Portland General Electric Co. (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979)]). The first safety objective would also support establishing a common performance standard for the plant SSCs categorized as safety related and for the human actions and programmatic controls needed to address DBAs. The use of a safety objective rooted in established standards would help maintain a consistency across Framework A, from the classification of safety-related SSCs to the content of TS that control those safety-related SSCs during the operation of a commercial nuclear plant. The second safety objective would consider potential risks to public health and safety beyond immediate threats—an approach that would ensure that commercial nuclear plants licensed under part 53 are at least as safe as those previously licensed by the NRC per the Commission’s “Policy Statement on the Regulation of Advanced Reactors.”

The collective set of performance-based requirements in Framework A would support the NRC findings required to grant an application for a utilization facility under Section 182 of the AEA that the utilization of SNM will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. This construct would be similar to existing NRC regulations and guidance, which the Commission has said on many occasions, do not specifically define “adequate protection” but where compliance with NRC regulations and guidance may be presumed to assure adequate protection at a minimum. The requirements throughout Framework A that support demonstrating compliance with the second safety objective would be similar to current regulations that both contribute to assuring adequate protection of public health and safety and are necessary or desirable to promote the common defense and security or to protect health or to minimize danger to life or property under Section 161 of the AEA.

Consistent with historical practice, Sections 182 and 161 of the AEA are cited as authorizing legislation within this proposed rule. However, specific language from the AEA would not be incorporated into the safety objectives or safety criteria in part 53. This is because, again consistent with historical practice, the NRC would not be defining “adequate protection” through the individual safety requirements in part 53. Rather, Framework A would enable the NRC to make its required findings under the AEA by providing sufficient performance standards, safety criteria, and related requirements on how applicants must demonstrate compliance with subpart B and other subparts. An important example is that design features that would be required under § 53.400 for LBEs would not only need to be provided to address the safety criteria, but the performance of those design features would need to be demonstrated through analysis, appropriate test programs, prototype testing, operating experience, or a combination thereof for a wide range of conditions predicted to exist throughout the plant's lifetime. Compliance with these regulations and the supporting guidance would be presumed to assure adequate protection at a minimum and to promote the common defense and security, protect health, and minimize danger to life or property.

Section 53.210 would provide safety criteria for DBAs that would be required to be identified under § 53.240 and analyzed in accordance with § 50.450(f) in subpart C of Framework A. Subsequent sections in Framework A would require that the SSCs relied upon to demonstrate compliance with the criteria in § 53.210 be classified as safety related. The use of safety-related SSCs and the 25 roentgen-equivalent man (rem) reference values for potential radiological consequences would align with traditional deterministic approaches for LWRs from §§ 50.34, 52.79, and 100.1 to establish a minimum level of safety that limits the possibility of an immediate threat to public health and safety. A footnote similar to that included in § 50.34(a)(1)(ii)(D)(*1*) and § 52.79(a)(1)(vi)(A) would be included in § 53.210 to explain that the use of the 25 rem value would not be intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this proposed section as a reference value, which would be used in the evaluation of plant design features with respect to DBAs to verify that the proposed designs would provide assurance of low risk of public exposure to radiation in the event of an accident. The inclusion of the safety criteria for DBAs in subpart B provides a logical structure supporting the identification and treatment of safety-related SSCs and establishing the corresponding functional design criteria for those SSCs.

Section 53.220 would provide safety criteria for LBEs other than DBAs that would be required to be identified under § 53.240 and analyzed in accordance with § 53.450(e) in subpart C. Whereas § 53.210 and the related requirements for safety-related SSCs would provide that a defined success path exists for DBAs, the safety criteria for LBEs other than DBAs would establish the connection to a broader set of potential internal and external hazards and addresses defense-in-depth matters such as a balanced consideration of prevention and mitigation. The safety criteria in § 53.220(b) would include a cumulative risk measure and support a performance-based approach to developing an appropriate combination of design features and programmatic controls to prevent or mitigate LBEs other than DBAs. It is worth noting that the evaluation of plant risks as represented by a comparison of analysis results to cumulative risk measures would be one of several performance standards used in subpart B. The proposed use of multiple performance standards, including deterministic criteria and defense-in-depth measures, reflects an integrated decision-making process similar to that described in RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 3.

The RG 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors,” would describe an acceptable approach for identifying and analyzing LBEs, including the use of the quantitative health objectives (QHOs) stated in the NRC’s Safety Goal Policy Statement as a cumulative risk measure. The QHOs from the safety goal policy statement, which would form the basis of § 53.220(b), are a well-established cumulative risk measure used in NRC risk-informed decision-making and are a logical performance metric to support the risk management approaches in the various subparts comprising proposed Framework A. The derivation of the values proposed in § 53.220(b) was originally documented in NUREG-0880, “Safety Goals for Nuclear Power Plant Operation.” The Commission stated in the SRM for SECY-10-0121, “Modifying the Risk-Informed Regulatory Guidance for New Reactors,” dated March 2, 2011 (ADAMS Accession No. ML110610166) that “…the existing safety goals, safety performance expectations, subsidiary risk goals and associated risk guidance …, key principles and quantitative metrics for implementing risk-informed decision making, are sufficient for new plants…".

The Commission stated in the introduction of the Safety Goal Policy Statement that the use of the safety goals could lead to a more coherent and consistent regulation of nuclear power plants, a more predictable regulatory process, a public understanding of the regulatory criteria that the NRC applies, and public confidence in the safety of operating plants. Following the issuance of the Safety Goal Policy Statement, the NRC has used the safety goals as a cumulative risk measure to support many decisions involving safety judgments during the licensing and regulation of operating reactors and proposed nuclear reactor designs. As described in NUREG-0880, the values used in § 53.220(b) would be expressed in terms of an average individual biologically in terms of age and other risk factors. Although the QHOs are defined in terms of fatality risks, the Commission continues to make clear that no death attributable to nuclear power plant operation will ever be "acceptable" in the sense that the Commission would regard it as a routine or permissible event. The QHOs as used in this proposed rule would establish acceptable risks, not acceptable deaths.

Applicants under the proposed Framework A could choose to develop and seek NRC approval to use surrogate measures to show that particular designs or plants satisfy the QHO-related safety criteria in proposed § 53.220(b). Such surrogate measures could be used in a manner similar to the use of core damage frequency and conditional containment failure probability for light water reactors within the safety goal evaluation process in NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission."

Section 53.230 would provide safety functions needed to ensure that the safety criteria in §§ 53.210 and 53.220 can be met if a commercial nuclear plant experiences a LBE. In turn, the safety functions would be fulfilled by the design features and programmatic controls specified later in Framework A. Section 53.230 would specify that limiting the release of radioactive materials from the facility is the primary safety function, and therefore, limiting potential offsite consequences (i.e., dose to a hypothetical individual) would be used as the primary performance metric for Framework A throughout part 53. The additional safety functions needed to limit the release of radionuclides may include controlling processes related to reactivity, heat generation, heat removal, and chemical interactions. This proposed rule provides flexibility in how the safety functions supporting retention of radionuclides are identified, fulfilled, and maintained for commercial nuclear plants of varying sizes and technologies expected to be licensed under proposed Framework A of part 53.

Proposed § 53.240 would require applicants to identify and address LBEs. LBEs are unplanned events, resulting from both internal and external hazards, that are used in the design and analyses supporting commercial nuclear plants to ensure estimates of offsite consequences from analyses performed in accordance with proposed § 53.450

are below the safety criteria identified in proposed §§ 53.210 and 53.220 and that SSCs, personnel, and programs address the safety functions from proposed § 53.230. Including a high-level performance requirement related to the identification and analysis of LBEs in subpart B would reflect the historical and continuing importance of evaluating unplanned events as part of the licensing of commercial nuclear plants. Proposed § 53.240 would require identification and analysis of LBEs in accordance with § 53.450, which would require a PRA. One acceptable method of using PRAs to identify and assess LBEs would be the methodology in RG 1.233.

Section 53.250 would establish defense-in-depth requirements based on the longstanding philosophy of providing defense in depth to address uncertainties about the design, operation, and performance of commercial nuclear plants. For example, defense in depth is addressed through layered prescriptive technical requirements (e.g., fuel performance, cladding integrity, reactor coolant system integrity, containment performance, and emergency planning zones) for light-water reactors within parts 50 and 52. In contrast, the flexibility afforded to applicants in how they propose to demonstrate compliance with the high-level safety criteria within Framework A would necessitate this specific requirement to ensure defense in depth is provided to address various uncertainties in human knowledge, analytical models, and plant performance. The requirement in this section would state that no single engineered design feature, human action, or programmatic control, no matter how robust, should be exclusively relied upon to address LBEs other than DBAs. The use of the phrase “engineered design feature” in the description of those items for which defense in depth is required would address the possible crediting of inherent characteristics within the design and analysis for commercial nuclear reactors and the reduced uncertainties associated with such characteristics. While defense in depth would only be assessed for LBEs other than DBAs, the need to ensure dedicated success paths for DBAs would contribute to the overall defense in depth for each commercial nuclear plant under Framework A.

Section 53.260 would govern normal operations and would consist of two requirements. First, it would include metrics that establish a level of safety or a backstop based on current requirements in 10 CFR part 20, “Standards for Protection Against Radiation,” that limits doses to members of the public or contamination of unrestricted areas once a plant is licensed and is operating. Second, it would include requirements equivalent to those in part 20 and § 50.34a specific to the design of nuclear reactors that require ensuring doses will be and are maintained as low as reasonably achievable (ALARA) for normal operations through the use of an appropriate combination of design features and programmatic controls. The ALARA requirements in § 50.34a, appendix I to part 50 for light-water reactor effluents, and part 20 allow applicants and licensees to take into account the state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations when determining the appropriate combination of design features and programmatic controls to limit the release of radionuclides. These same considerations would apply to the proposed requirements in part 53, which would also emphasize that ALARA would be achieved through a combination of design choices and programmatic controls with an appropriate consideration of potential costs. Given the variety of potential advanced reactor technologies, this provision affords applicants the flexibility to propose an appropriate balance between programmatic controls, such as operational programs, and design features to demonstrate that the ALARA principle has been met for their facility. See, e.g., *Michigan v. EPA*, 135 S. Ct. 2699 (2015). The proposed requirements related to ALARA would also align with and help ensure compliance with U.S. Environmental Protection Agency (EPA) regulations in part 190 to title 40 (Protection of Environment).

Section 53.270 would provide for the protection of plant workers. This section would include the part 20 limits on occupational exposures as a way to ensure that protection of plant workers is addressed within the high-level safety criteria for part 53. Similar in content and rationale to the discussion in the preceding section for proposed § 53.260 to maintain doses to the public ALARA, the proposed § 53.270 would require applicants and licensees to identify and provide a combination of design features and programmatic controls to ensure occupational doses are ALARA. The history of occupational exposures experienced at the current generation of nuclear plants has demonstrated both the need for and ability to use combinations of plant design features and programmatic controls to limit doses to plant workers. The need to give special consideration to limiting occupational exposures during all phases of the life cycle of future commercial nuclear plants is especially important given the wide variety of potential reactor technologies and designs that could be licensed under Framework A of part 53.

Subpart C – Design and Analysis Requirements

This subpart would provide requirements for the design of commercial nuclear plants and the supporting analyses, including the analyses of LBEs, to demonstrate that the performance standards in proposed subpart B can be satisfied. The sections within subpart C would reflect the overall hierarchy throughout Framework A, which would cover: (1) plant-level safety criteria (§§ 53.210, 53.220, and 53.470); (2) safety functions

(§ 53.230) needed to demonstrate compliance with the safety criteria; (3) design features (§ 53.400), human actions, and programmatic controls needed to fulfill the safety functions; and (4) functional design criteria (§§ 53.410 and 53.420) that must be defined for each design feature relied on to demonstrate the safety criteria are met. Subpart C would also contribute to the logic and structure of Framework A by distinguishing between: (1) safety-related SSCs, human actions, and programmatic controls needed to protect against DBAs, which are used to satisfy the safety objective of limiting the possibility of an immediate threat to the public health and safety, and (2) NSRSS SSCs and licensee-controlled programs that address LBEs other than DBAs, which generally make up the appropriate measures considering potential risks to public health and safety. Section 53.400 would establish a requirement that design features be provided for each commercial nuclear plant to satisfy the safety criteria and fulfill safety functions from proposed subpart B during LBEs. Other sections in subpart C would, in turn, further address the necessary capabilities and reliabilities for SSCs by establishing functional design criteria, fulfilling design requirements, performing analyses of LBEs, performing other supporting analyses, and categorizing structures, systems and components based on their roles in preventing or mitigating LBEs.

Section 53.410 would require that functional design criteria be defined for design features relied upon to demonstrate that the consequences from DBAs would be below the criteria in § 53.210 through analyses performed in accordance with § 53.450(f). Design features and functional design criteria for unplanned events would be determined or confirmed through analyses that includes both PRAs and deterministic analyses. Other sections within this and other subparts would establish appropriate controls on these design features (e.g., safety classification, protection from external hazards, quality assurance, and TS) to ensure the functional design criteria are satisfied. The performance requirements for the SSCs needed to address DBAs and the corresponding human actions and programmatic controls would contribute to ensuring that a commercial nuclear plant licensed under Framework A would demonstrate compliance with the safety objective that the plant poses no immediate threat to public health and safety.

Section 53.415 would require that safety-related SSCs be protected against or designed to withstand the effects of natural phenomena (e.g., earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches) and human-related hazards (e.g., dams, transportation routes, military and industrial facilities) of levels up to the design-basis external hazard levels as determined under § 53.510 without losing the capability to perform the safety functions stated in § 53.230. As used in this § 53.415 and subpart D, a hazard level would refer to such things as the magnitude and recurrence rate of an earthquake and the resultant ground motions, the height of a flood, the force of hurricane winds, or the concentrations of chemicals resulting from a release from a nearby facility. The design-basis external hazard level is defined in subpart A as "the level of severity or intensity of an external hazard for which the safety-related SSCs are designed to withstand with no adverse impact on their capability to perform their safety functions." This proposed requirement would support either traditional deterministic approaches for determining and protecting against external hazards or probabilistic approaches that have been developed for seismic hazards and are being developed for some other external hazards.

Section 53.420 would require that functional design criteria be defined for design features that play a significant role in demonstrating the safety criteria for LBEs other than DBAs are satisfied. The analyses required for this demonstration would be described in proposed § 53.450I, which would require that those events must be identified and assessed using a PRA methodology. The SSCs determined to be safety significant (i.e., either safety-related or NSRSS) would have associated special treatment requirements as specified in § 53.460. Special treatment would be defined in subpart A and generally refers to measures (e.g., quality assurance, testing, monitoring) taken beyond the procurement and installation of commercial grade products to provide confidence that the SSC will demonstrate compliance with the applicable functional design criteria. The inclusion of a systematic approach to identifying the functional design criteria for SSCs and tailoring the special treatments to specific LBEs and safety functions is an important contributor to satisfy the safety objectives in proposed § 53.200. This approach also would allow a more specific role to be defined for the DBA and associated safety-related SSCs in terms of ensuring a commercial nuclear plant poses no immediate threat to public health and safety. With the confidence that other sections in Framework A would require protection against an immediate threat to public health and safety, designers and licensees for commercial nuclear plants would be provided flexibility on how LBEs other than DBAs are either prevented or mitigated and how the cumulative plant risks remain below the safety criteria in § 53.220.

Section 53.425 would establish requirements for design features and related functional design criteria playing a role in limiting the release of radionuclides during normal operations and thereby satisfy the criteria in § 53.260 of subpart B. This section, as well as the subsequent section, § 53.430, for protection of plant workers, would include requirements to establish functional design criteria for SSCs contributing to achieving doses ALARA when considering the state of technology, the economics of improvements in relation to benefits to the public or worker health and safety, and other societal and socioeconomic considerations. These requirements related to keeping public doses and doses to workers ALARA would recognize the roles of both design features and programmatic controls in reaching desired objectives. The development of an integrated approach to maintaining doses to the public and workers ALARA during normal operations would present a particular challenge to applicants seeking only an approval or certification of the design of a commercial nuclear plant. Design features are nevertheless an essential element in limiting doses resulting from normal operations and would need to be considered. A performance-based approach to the design and the associated NRC review of the design can consider proposed combinations of design features and programmatic controls for maintaining doses ALARA for various applications made under proposed subpart H. See, for example, *Michigan v. EPA*, 135 S. Ct. 2699 (2015). In addition, § 53.425(b) would include a performance goal similar to that provided by appendix I to part 50 to assist designers, applicants, and licensees perform the evaluations of possible reductions in public dose from routine effluents when considering costs and other factors. As emphasized in existing regulations in part 50, the design objective of keeping doses to the public from routine plant effluents less than 10 millirem per year should not be construed as a radiation protection standard.

Section 53.440 would address various design requirements that warrant specific mention to ensure that the design features required by § 53.400 demonstrate compliance with the functional design criteria required by §§ 53.410 and 53.420. These requirements would be met through design practices, consideration of testing and operating experience, and various assessments of LBEs and other potential challenges to commercial nuclear power plants. Discussions of some of the key design requirements included in this section would be as follows:

- § 53.440(a): An essential element to ensuring a proposed design can demonstrate compliance with the performance criteria in proposed part 53 would be that the abilities of design features to fulfill their safety functions are demonstrated by a combination of analyses, test programs, prototype testing, and operating experience. This requirement closely aligns with the language in § 50.43(e) and is proposed in part 53 as the same foundational requirement that it has historically been in previous licensing and regulation of nuclear plants.
- § 53.440(b): The design and licensing of commercial nuclear plants can benefit from the use of generally accepted consensus codes and standards. Such codes and standards are a vehicle to ensure sufficient testing and qualification of materials and equipment as well as providing defined processes, specifications, and acceptance criteria for use by designers and suppliers. An example of using generally accepted consensus codes and standards to demonstrate compliance with this requirement would be citing particular consensus codes and standards as a means to demonstrate compliance with the quality assurance requirements in proposed subpart K for safety-related SSCs. The NRC acceptance of consensus codes and standards used in the design and licensing of a specific commercial nuclear plant would be provided through either the NRC's generic endorsement of a code or standard (i.e., through regulatory guidance), including any limitations or conditions, that can be referenced within an application or the review of a referenced code or standard as part of the review of a specific application.

- § 53.440(c): The design requirements in subpart C would require the materials used for safety related and NSRSS SSCs to be qualified for their service conditions over the plant lifetime.
- § 53.440(d): The requirements in § 53.440 would include the need to consider possible degradation mechanisms for materials and equipment to inform both the design process and the development of integrity assessment programs to be executed during plant operations in accordance with subpart F. The inclusion of requirements related to designing and monitoring for possible degradation mechanisms reflects important lessons learned from the history of light-water reactors as well as operating experience with structures and systems in countless other engineering endeavors.
- § 53.440(e) and (f): The design requirements in subpart C would capture specific design requirements similar to existing requirements in parts 50, 52, and 73 for protections against fires and explosions and consideration of safety and security together in the design process per the Commission's "Policy Statement on the Regulation of Advanced Reactors."
- § 53.440(g) and (h): Specific design requirements are being proposed to ensure that commercial nuclear reactors under part 53 have the capability to achieve and maintain subcriticality and long-term cooling. The requirements would be included to address the potential that some reactor designs may be able to achieve a stable end state for the purpose of event analyses but might need further actions to completely shut down and service the facility.
- § 53.440(i): The design, analysis, and development of programmatic controls under part 53 would consider the number of reactor units and other

significant inventories of radioactive materials contributing to the risks to public health and safety. This would reflect the definition of “commercial nuclear plant” in subpart A and reinforce that the evaluation of LBEs is performed on a plant-wide basis. This aspect of Framework A would be different from parts 50 and 52, which generally define safety requirements on the assumption of events involving only individual reactor units.

- § 53.440(j): A design requirement is proposed to provide a technology-inclusive requirement that would be equivalent to the requirements in § 50.150 to address the possible impact of a large commercial aircraft.
- § 53.440(k): The inclusion of a specific proposed requirement to address the risks to public health from potential chemical hazards of licensed material is appropriate given the diversity of reactor technologies and designs that might be licensed under part 53. The requirement in part 53 would be similar to the existing requirements in part 70 of this chapter that address both potential radiological and chemical hazards for licensed materials at fuel cycle facilities.
- § 53.440(l): Provisions are proposed to require that measures be taken during the design of commercial nuclear plants to minimize contamination of the facility and the environment, facilitate eventual decommissioning, and minimize the generation of radioactive waste in accordance with § 20.1406.
- § 53.440(m): A design requirement is proposed to provide a technology-inclusive equivalent to the requirements in § 50.68 by including options for commercial nuclear plants to either have a monitoring system capable of detecting a criticality as described in § 70.24 or to have restrictions on SNM that would prevent inadvertent criticality events.

- § 53.440(n): The design would need to reflect state-of-the-art human factors principles for safe and reliable performance in all settings that human activities are expected for performing or supporting the continued availability of plant safety or emergency response functions.

Section 53.450 would establish analysis requirements and would center upon the use of a PRA in combination with other generally accepted approaches for systematically evaluating engineered systems. The reliance on PRAs as a key component in the proposed analysis requirements for Framework A would reflect the decades of improvements in PRA methodologies and the increasing use of PRA techniques in the design, licensing, and oversight of both operating and future nuclear reactors. The Commission has stated in its Policy Statement entitled, “Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities,” that the use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data, and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy. The inclusion of other generally accepted approaches to be used in combination with PRA would reflect the need to complement PRAs with supporting engineering analyses related to the performance of SSCs as well as using expert panels or other tools to address the limitations of any assessment technique. The need to supplement PRA insights with other engineering approaches and judgments reflects the NRC’s longstanding policy for regulatory decisionmaking to be risk informed but not solely based on numerical results of a risk assessment (i.e., a risk-based approach). Part 53 would maintain a role for NRC’s traditional deterministic approaches (particularly for DBAs) and defense-in-depth philosophy by including specific requirements utilizing these regulatory tools in subparts B and C.

PRA would be used in combination with other techniques in Framework A to identify and categorize LBEs, classify SSCs, and evaluate defense in depth. This foundational role for the PRA necessitates that it would be developed, performed, and maintained in accordance with NRC-approved standards and practices (see § 53.450(c) and (d)). The computer codes used to model the plant response and the behavior of the barriers to the release of radionuclides would need to be qualified for the range of conditions being simulated across a wide range of unplanned events. These analyses would need to use realistic approaches and address uncertainties associated with states of knowledge, modeling, and performance of SSCs.

The categories of LBEs used in Framework A would include anticipated event sequences, unlikely event sequences, and very unlikely event sequences. The unlikely event sequences would include those events with estimated frequencies well below the frequency of events expected to occur during the lifetime of a commercial nuclear plant. An important aspect of the analysis requirements is that, under proposed § 53.450(e), the analyses of LBEs other than DBAs would not only be used to show the performance criteria of § 53.220 are satisfied but to also show that evaluation criteria defined for each licensing-basis event or category of LBEs would also be satisfied. Such evaluation criteria for specific LBEs or categories of LBEs would be defined in terms of limits on the release of radionuclides or maintaining the integrity of one or more barriers used to limit the release of radionuclides and reflect the established graded approach of allowing lesser potential consequences from more frequent events. An example of such evaluation criteria for a range of LBEs is provided in RG 1.233. Another proposed requirement for the proposed § 53.450(e) analyses is that the methodology would need to include a means to identify event sequences deemed risk-significant such that those event sequences can be given special attention within other sections of part 53.

Framework A would maintain an important role for a deterministic-style analysis of DBAs in the performance criteria of § 53.210 and the related analytical requirements in § 53.450(f). The analysis of DBAs would be required to address the event sequences with estimated frequencies below the expected lifetime of a generation of reactors (e.g., event sequences with frequencies down to one in ten thousand years). As proposed in this section, DBAs would need to be analyzed using deterministic methods and ensure a safe, stable end state with reliance upon only safety-related SSCs and human actions, if needed, to be performed by operators licensed under the provisions of §§ 53.760 through 53.795.

While the DBA under Framework A would be similar to the traditional DBA under part 50, there are important distinctions between the overall role of DBAs in part 50 and proposed Framework A. In Framework A, the role of the DBA would be more narrowly focused on selecting safety-related SSCs and determining functional design criteria for those SSCs to ensure the commercial nuclear plant poses no immediate threat to public health and safety. The overall control of risks posed by commercial nuclear plants under Framework A would be provided by the analyses of and measures taken for both DBAs and other LBEs, including very unlikely event sequences. This would contrast with the traditional deterministic approach in part 50 wherein design-basis events such as DBAs were used with the intent to provide bounding assessments, incorporate standard design rules such as assumptions related to single failures, and to define conservative performance requirements for safety-related SSCs. Limitations related to the traditional deterministic approach were addressed in part 50 through case-by-case assessments and specific actions for issues such as anticipated transients without scram and station blackout.

Section 53.450 would also include provisions to ensure that analyses are performed to support the design requirements of § 53.440(e) on fire protection, § 53.440(j) related to aircraft impact assessments, and § 53.425 on controlling effluents and otherwise maintaining the dose to individual members of the public ALARA. The proposed analysis requirements related to fire protection would support either a traditional, deterministic approach or a more risk-informed approach where the risks from fires are addressed within the identification and analyses of LBEs.

Section 53.460 would establish criteria for the safety classification of SSCs and determination of appropriate special treatments. As noted in subpart A, the term “special treatment” would be defined to mean those requirements, such as measures taken to satisfy functional design criteria, quality assurance, and programmatic controls, that provide assurance that certain SSCs will provide defense in depth or perform risk-significant functions and that provide confidence that the SSCs will perform under the service conditions and with the reliability assumed in the analysis performed in accordance with § 53.450 to satisfy the safety criteria in § 53.210 and § 53.220. The terminology used in part 53 would include the following categories for SSC classification: (1) safety related, (2) NSRSS, and (3) non-safety significant. Requirements for safety-related SSCs would be defined in other sections of part 53 and would include using TS for controls during operation and the application of quality assurance requirements from subpart K.

Requirements for NSRSS SSCs would include the need to identify appropriate special treatments such as performance measures on reliability. Licensees would generally be afforded more control and flexibility in maintaining and changing special treatment requirements for SSCs categorized as NSRSS. Non-safety-significant SSCs would be expected to be addressed under normal licensee programs for commercial grade equipment and typical industry practices for general plant design and maintenance.

Section 53.470 would allow an applicant or licensee to seek operational flexibilities by adopting more restrictive criteria than those provided in § 53.220 and that might otherwise be used in the analysis of LBEs under § 53.450(e). Such an approach might be taken to ensure sufficient safety margins to gain operational flexibilities in areas such as justifying specific emergency planning zones, siting in relation to population centers, or staffing levels if the methodology being used has not already adopted criteria and included appropriate assumptions related to the given topical area such as emergency planning zones, siting, or staffing. An example is the methodology described in SECY-19-0117, "Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," which could support adopting design objectives to support siting alternatives even though the methodology does not include specific assumptions related to protective actions or population densities. In the example, an applicant or licensee could propose to justify siting proposals by adopting alternate criteria for very unlikely event sequences that would require calculated consequences for an individual at the exclusion area boundary to be less than one rem total effective dose equivalent versus higher acceptance criteria within an NRC accepted methodology that includes dose-related criteria for such events based on limiting the risk of prompt fatalities. This section in Framework A would establish requirements to ensure that, if more restrictive evaluation criteria than those required by a methodology were used to justify operational flexibilities, then the analysis, design features, and programmatic controls would be established and maintained accordingly.

Section 53.480 would establish the seismic design considerations. This proposed section would relate to the safety criteria in subpart B, the analytical requirements related to external hazards in § 53.450, and subpart D, “Siting Requirements.” For licenses issued under Framework A of part 53, this section in subpart C would support a variety of approaches to seismic design. For example, a design for a commercial nuclear plant could show that SSCs are able to withstand the effects of earthquakes by adopting an approach similar to that in appendix S to part 50, or the design process could follow the more recent risk-informed alternatives afforded by standards development organizations, or the design could be done with the full integration of seismic PRAs into the design and licensing of a particular commercial nuclear plant. This section has been developed to accommodate a risk-informed, performance-based seismic design approach similar to that described in the NRC Research Information Letter, 2021-04, Feasibility Study on a Potential Consequence-Based Seismic Design Approach for Nuclear Facilities,” and consensus codes and standard such as the American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI) 43-19, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities,” should these methods be endorsed or otherwise found acceptable in the future by the NRC. The analyses required by § 53.450 would need to address seismic hazards as well as other external hazards. The expected responses of SSCs to a range of seismic events would be considered when ensuring that the safety criteria defined in § 53.220 would be met. The potential SSC responses to seismic hazards could be addressed in the analyses using a fragility model (conditional probability of its failure at a given hazard input level), a high confidence of low probability of failure value, or other method endorsed or otherwise found acceptable to the NRC.

Subpart D – Siting Requirements

Proposed subpart D in Framework A would state requirements for the siting of commercial nuclear plants and would serve the role provided by part 100, “Reactor Site Criteria,” for nuclear reactors licensed under parts 50 and 52. The NRC is proposing to include the siting requirements in this subpart consistent with the overall organization of Framework A by the phases of a project life cycle. Including the siting requirements for commercial nuclear plants to be licensed under Framework A in a dedicated subpart would also reduce the need to revise part 100 to conform to the relationships Framework A would establish among design and analysis, siting, plant operations, and decommissioning. As reflected in proposed § 53.500, the reason for establishing siting requirements would remain the same as it has been historically, which is to ensure that licensees and applicants assess what impact the site environs may have on a commercial nuclear plant (e.g., external hazards) and, conversely, what potential adverse health and safety impacts a commercial nuclear plant may have on nearby populations.

Proposed § 53.510 would require that design-basis external hazard levels be identified and characterized based on site-specific assessments of natural and manmade hazards with the potential to adversely affect plant functions. The site-specific assessments would be used in the proposed § 53.415 that would require that safety-related SSCs be designed to withstand the effects of natural phenomena and man-related hazards of levels or severities up to design-basis external hazard levels. The design-basis external hazard levels for relevant hazards for a site would need to account for uncertainties and variabilities in data, models, and methods used to characterize those hazards. As discussed in RG 1.233, existing approaches and guidance could be used to demonstrate compliance with this requirement. The historical importance of assessing seismic events as risks to commercial nuclear plants and the associated development of risk-informed approaches to address seismic events would be reflected in proposed § 53.480, “Earthquake engineering,” and specific requirements in subpart C. For example, ASCE/SEI 43-19 is a risk-informed, performance-based seismic design standard that supports a graded approach for seismic design by grouping SSCs into different seismic design categories (SDCs) in accordance with their risk significance and could be an acceptable approach under Framework A if the standard is endorsed or otherwise found acceptable by the NRC. As such, the approach described in RG 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” would be an acceptable way to develop site-specific ground motion response spectra under proposed Framework A for SSCs that are categorized as the highest seismic design category (SDC-5).

The evaluation of seismic hazards under subpart D would need to be sufficient to inform a site-specific design (e.g., a CP or custom COL) or confirm the use of a standard design (e.g., a standard design certification) for a subject commercial nuclear plant under § 53.480 and other sections of subpart C. Under a risk-informed approach such as that described in ASCE 43-19 and potentially allowed under proposed Framework A, if the methodology was endorsed or otherwise found acceptable by the NRC, an application would not need to use a single safe shutdown earthquake (SSE) to assess all safety-related SSCs. It could use several design-basis ground motions (DBGMs) to assess SSCs in various SDCs (i.e., one DBGM per SDC). Section 53.510(d) would state that geologic and seismic siting factors an applicant must consider when evaluating siting for a commercial nuclear plant must also include related hazards such as seismically induced flooding and volcanic activity that may affect the design and operation of a proposed commercial nuclear plant for the proposed site.

Section 53.520 would require applicants to identify and assess site characteristics related to topics such as meteorology, geology, hydrology, and other areas in the design and analyses required under subpart C.

Proposed section 53.530 would set requirements for population-related considerations and maintain requirements and definitions similar to those currently in part 100 for an exclusion area, low population zone, and population center distance. The NRC recognizes that some applicants may propose to essentially collapse the exclusion area and low population zone to the site boundary by demonstrating that the calculated consequences of DBAs remain below the proposed dose guidelines used in Framework A, which are the same as those in the existing regulations in parts 50, 52, and 10 CFR part 100, “Reactor Site Criteria.” The proposed definitions in § 53.020 would allow such configurations, assuming they were justified by the design and analyses from subpart C, albeit with a requirement to actually define the exclusion area and low population zone as being at the site boundary. This approach should provide flexibility to justify alternative exclusion areas and low population zones without foreclosing the option for an applicant to define more conventional exclusion areas and low population zones outside of a defined site boundary. The NRC's long-standing preference for siting reactors in areas of low population density would be maintained in Framework A by using the current language from part 100 in proposed § 53.530(c). The NRC currently plans to revise guidance related to population densities surrounding a commercial nuclear plant to reflect Commission direction in SRM-SECY-20-0045, "Population Related Siting Considerations for Advanced Reactors."

Subpart E – Construction and Manufacturing Requirements

The proposed part 53 language would establish construction and manufacturing requirements in subpart E for Framework A and in subpart O for Framework B. The two subparts would essentially be the same but would be included separately within the frameworks to support clarity and ease of use due to the differences in the internal references between Framework A and Framework B. The proposed language for construction-related activities would largely reflect current requirements in part 50 without any fundamental changes. Limited changes would be made in several places, as described in the following paragraphs, to be technology-neutral and for consistency with the organization and language of part 53. The proposed language for requirements for manufacturing activities would largely mirror those for construction-related activities. However, the proposed manufacturing requirements have been updated from the current requirements in subpart F of part 52 to better accommodate the possible factory fabrication of manufactured reactors. The manufacturing of specific components outside the scope of a ML would not be addressed by these proposed subparts.

Sections 53.600 and 53.4100 within Frameworks A and B, respectively, would establish the overall construction and manufacturing requirements for CPs, OLs, COLs, MLs, and limited work authorizations (LWAs), and in doing so would connect the construction and manufacturing requirements to the safety criteria, quality assurance requirements, and other requirements located in other subparts in Frameworks A and B. These requirements would require that construction and manufacturing activities be managed and conducted such that when combined with associated design features and programmatic controls, the constructed plant would satisfy the relevant requirements in subpart B in Framework A and subpart R in Framework B.

Sections 53.605 and 53.4105 in Frameworks A and B, respectively, would establish requirements for the reporting of defects and noncompliances during construction. Both sections would provide equivalent requirements to those in § 50.55(e).

Sections 53.610(a) and 53.4110(a) in Frameworks A and B, respectively, would establish the requirement to have in place a well-defined command and control structure to manage construction activities. The requirements would generally reflect current requirements, with an emphasis on the quality assurance programs demonstrating compliance with the requirements in proposed subparts K and U, which would both be equivalent to appendix B to part 50. The proposed § 53.610(a)(6) would require appropriate programmatic controls to provide special treatment for NSRSS SSCs to align with requirements in other subparts in Framework A. The sections in both frameworks would also refer to other NRC regulations to address matters such as requirements to have a fitness-for-duty program, a radiation protection program if radioactive materials are brought onto the site, and security programs to protect sensitive information and cyber threats.

Sections 53.610(b) and 53.4110(b) would provide requirements governing construction activities, including the equivalent of the requirement in a § 50.10(e) that prohibits starting construction until the NRC has authorized the activities by issuing a CP, COL, ESP, or LWA. Other requirements in these paragraphs would be equivalent to requirements in parts 50 and 52 with appropriate references to other parts for items such as possession of byproduct material or SNM, protecting operating units from construction activities for commercial nuclear plants with multiple reactor units, and having a redress plan in case LWA activities are terminated.

Sections 53.610(c) and 53.4110(c) would address inspection and acceptance activities by including equivalent requirements in part 53 to specific quality assurance criteria in appendix B to part 50 and the part 53 equivalents to ITAAC in part 52 if the construction is performed under a COL.

Sections 53.620(a) and 53.4120(a) would include proposed requirements covering the activities performed under a ML issued under either Framework A or Framework B. Provisions related to MLs were first adopted by the NRC in 1973, through the addition of appendix M to part 50. The regulation supported the manufacture of a nuclear power reactor to be incorporated into a nuclear power plant under a CP and operated under an OL at a different location from the place of manufacture[3]. The regulations and processes for MLs were changed substantially in the part 52 rulemaking in 2007 (72 FR 49352). The most important shift in the ML concept in that rulemaking was that a final reactor design, which would be equivalent to that required for a standard design certification under part 52 or an OL under part 50, must be submitted and

[3] The NRC issued a manufacturing license to Offshore Power Systems Floating Nuclear Plants 1-8 in 1982 but the project was abandoned in 1984.

approved before issuance of a ML. The rationale for that change was that approval of a final design ensures early consideration and resolution of technical matters before there is any substantial commitment of resources associated with the actual manufacture of the reactor, which will greatly enhance regulatory stability and predictability.

The proposed part 53 sections in subparts E and O for manufacturing and in subparts H and R for licensing matters would maintain equivalent requirements as part 52 for the design and manufacturing processes considered in applying for and complying with a ML. The NRC approval of a standardized design and related manufacturing processes, coupled together with a stable workforce and established procedures, has the potential for maintaining and even improving the quality and consistency of manufacture, as compared to the traditional method of constructing reactors onsite by a variety of contractors and subcontractors.

Both frameworks in part 53 would provide requirements and pathways to obtain MLs covering a manufactured reactor and a manufactured reactor module. These would be defined in § 53.020 as: *"Manufactured reactor* means the essential portions of a nuclear reactor that are manufactured under a ML and subsequently transported and incorporated into a commercial nuclear plant under a combined license" and *"Manufactured reactor module* means a manufactured reactor loaded with fuel prior to transport to a licensed location for installation and commercial operation."

Subparts E and O would include requirements that would apply to portions of a manufactured reactor in recognition that some activities covered by a ML may occur at different fabrication facilities. As with the preceding sections on construction, §§ 53.620 and 53.4120 would establish the requirements to have in place a well-defined command

and control structure, programs, and procedures to manage manufacturing-related activities.

Section 53.620(b) in subpart E and § 53.4120(b) in subpart O would propose requirements for executing the manufacturing activities following receipt of a ML in Frameworks A and B. Much of the specifics on the design and manufacturing processes would be addressed through the applications for and issuance of an actual ML. The importance of the ML is reflected in several of the proposed requirements in §§ 53.620(b) and 53.4120(b) that would refer to complying with the ML, including conducting manufacturing processes within facilities for which the license holder can control activities. The essential role of post-manufacturing inspections would also be incorporated into these proposed sections by requiring inspections and acceptance processes for manufactured reactors, portions of a manufactured reactor, and manufactured reactor modules.

Sections 53.620(c) and 53.4120(c) in Frameworks A and B, respectively, would provide proposed requirements for the control of radioactive materials if the holder of a ML plans to possess and use source, byproduct, or SNM as part of the manufacturing process. By and large, the proposed subparts E and O would refer to NRC regulations in 10 CFR part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material,” 10 CFR part 40, “Domestic Licensing of Source Material,” and 10 CFR part 70, “Domestic Licensing of Special Nuclear Material,” for the requirements on controlling radioactive materials. Several specific requirements to address the potential hazards of radioactive materials are proposed in areas such as having a fire protection program, emergency plan, training programs, and procedures to minimize contamination.

Perhaps the most significant change proposed for MLs in part 53 would relate to § 53.620(d) in subpart E, § 53.4120(d) in subpart O, and the associated licensing provisions in subparts H and R that would allow and establish requirements for the loading of fuel into a manufactured reactor module for subsequent transport and use at a commercial nuclear plant with a COL. The first requirement in the proposed §§ 53.620(d) and 53.4120(d) would establish limitations on when an ML can authorize the loading of fuel. The proposed requirement would include the manufactured reactor module having at least two independent mechanisms that can prevent criticality should conditions result in the maximum reactivity being attained for the fissile material. The NRC has a longstanding practice of requiring defense in depth for preventing accidents in any facility dealing with SNM, including requirements in § 70.64 for fuel cycle facilities to adhere to the "double contingency principle." This proposal would require that the NRC determine that: (1) any such module in which these two mechanisms have been installed is not a utilization facility as defined in section 11cc. of the AEA or proposed § 53.020 until it is installed in its final place of use and the NRC has found that both the ITAAC in the ML are met under § 53.620(f) or § 53.4120(f) and the ITAAC in the COL that authorized reactor construction are met under § 53.1452(g) or § 53.5052(g); and (2) upon a Commission finding with respect to a particular module that the ITAAC are met in accordance with a COL and § 53.1452(g) or § 53.5052(g), that the module is a utilization facility and all COL provisions and regulations applicable to the type of commercial nuclear plant for which the Commission has made the finding apply to that module.

If the condition for authorizing the loading of fuel is satisfied, the proposed §§ 53.620(d) and 53.4120(d) would specify additional requirements to address the fuel loading operations. The additional proposed requirements would require that the following be in place prior to the receipt of SNM: (1) radiation monitoring instrumentation and alarms; (2) measures to prevent criticality accidents satisfy the requirements in §§ 70.61 and 70.64 and to detect potential criticality accidents in accordance with § 53.440(m); (3) appropriate procedures, equipment, and personnel qualified for the fuel loading; (4) physical security programs; (5) and material control and accounting programs. The proposed regulations in part 53 covering the activities related to the storage, movement, and loading of fresh fuel into a manufactured reactor module in the manufacturing facility would refer to the applicable regulations in part 70. The proposed §§ 53.620(d) and 53.4120(d) would also call for the loading or unloading of fresh fuel into a manufactured reactor module and any changes to the configuration of reactivity-related systems to be performed by a certified fuel handler demonstrating compliance with the requirements in subpart F.

Sections 53.620I in subpart E and 53.4120(e) in subpart O would propose to limit the transport of a manufactured reactor or major portions of a manufactured reactor to only the site of a licensee with a COL that authorizes the construction of a commercial nuclear plant using a manufactured reactor under the specific ML. This proposed requirement is similar to the limitations in § 52.153, with the difference being that part 53 would propose to allow the installation of a manufactured reactor at the site of a COL but would not propose to support installation at the site of a CP. The possible combination of a manufactured reactor and the licensing option of CP and OL seems unlikely and would complicate matters by requiring the introduction of ITAAC into the licensing provisions for CP and OL. Additional proposed paragraphs in §§ 53.620(e) and 53.4120(e) would provide requirements for protecting manufactured reactors or major portions thereof during transport to the site of the commercial nuclear plant. The proposed loading of

SNM into a manufactured reactor module introduces additional complexities into the transportation of a module and would be addressed in the proposed part 53 by including references to the transportation and security requirements in 10 CFR part 71, "Packaging and Transportation of Radioactive Material," and part 73.

Sections 53.620(f) and 53.4120(f) in Frameworks A and B, respectively, would include proposed requirements for the acceptance and installation of a manufactured reactor or manufactured reactor module at the site of a commercial nuclear plant. The proposed requirements would reference the construction requirements in §§ 53.610 and 53.1410 to govern the integration of the manufactured reactor or manufactured reactor module into the construction of a commercial nuclear plant. Other proposed requirements in the sections would address required receipt inspections, ITAAC defined in the ML and COL, and verification that interface requirements between the manufactured reactor or manufactured reactor module and the balance of the commercial nuclear plant have been met.

Subpart F – Requirements for Operation

Proposed subpart F would provide the requirements for the operations phase of a commercial nuclear plant to ensure that the safety criteria in subpart B are satisfied throughout the plant's lifetime and during all modes of normal operation and unplanned events. Section 53.700 would provide the overall objectives and general organization of subpart F, which would be to establish requirements during operations for: (1) plant SSCs; (2) plant personnel; and (3) plant programs.

Proposed § 53.710 would provide the requirements for maintaining capabilities, availability, and reliability of SSCs to support demonstrating compliance with the safety criteria and design requirements for unplanned events that are described in proposed subparts B and C. The basic structure of this proposed section would be that controls for safety-related (SR) SSCs are provided by TS and controls for NSRSS SSCs are required to be addressed with licensee-controlled documents and procedures.

The general content and control of TS under the proposed Framework A would be similar to the requirements in parts 50 and 52. The proposed requirements for TS would include limits on the inventories of radioactive materials, plant operating limits, and specific requirements for each safety-related SSC, including limiting conditions for operation (LCO) and required surveillances. The proposed requirements for TS would also include a section on important design elements, which is similar to design features in § 50.36, and a section for administrative controls. A provision addressing the development and submittal of TS to address decommissioning activities would also be included in the proposed subpart G.

The proposed requirements for TS under Framework A would not carry over safety limits or associated limiting safety system settings from § 50.36, which contains TS requirements for operating reactors under parts 50 and 52. As discussed in SECY-18-0096, systematic assessments and more mechanistic approaches to evaluating source terms support an alternative approach to establishing barrier-based safety limits. An example provided in that paper is a comparison of (1) the traditional specified acceptable fuel design limits support protecting a specific barrier from potential failure mechanisms (e.g., departure from nucleate boiling to protect fuel cladding) and (2) the specified acceptable system radionuclide release design limit (SARRDL) concept, which limits the possible increase in circulating radionuclide inventory during normal operations or an anticipated operational occurrence as part of an integrated or “functional containment” approach. Additional discussion of the use of SARRDL in the design and licensing of advanced reactors is provided in RG 1.232. The SARRDL could be addressed as an operating limit within this proposed construct of requirements for TS.

The proposed requirements for TS under Framework A would also not include the criteria for identifying LCOs, which, like safety limits, is also included in § 50.36. Instead, consistent with subparts B and C, the TS requirements in subpart F of Framework A would define TS LCOs as providing limits on safety-related SSCs. The safety-related SSCs protect against an immediate threat to public health and safety to demonstrate compliance with the safety criteria in the proposed § 53.210. In the proposed construct for Framework A, risk significant SSCs would be addressed through a combination of TS for the safety-related SSCs and establishment and monitoring of performance standards for NSRSS SSCs.

In addition to addressing TS for safety-related SSCs, proposed § 53.710 would require appropriate controls be developed and implemented for NSRSS SSCs. Examples include appropriate surveillances and controls established through reliability assurance programs. Configuration management and other special treatments would provide that the capabilities, availabilities, and reliabilities of NSRSS SSCs are maintained consistent with the underlying risk assessments while providing flexibility to licensees through maintaining the management functions within licensee-controlled programs. Controls on NSRSS SSCs are appropriate as part of the overall performance-based approach within proposed Framework A. Additionally, these controls justify proposed changes in Framework A from the traditional or deterministic approaches in parts 50 and 52 in areas such as replacing the single-failure criterion with a probabilistic (reliability) criterion (See SRM-SECY-03-0047, “Policy Issues Related to

Licensing Non-Light-Water Reactor Designs," dated June 26, 2003). This approach could also support the incorporation of risk insights and analytical margins to gain operational flexibilities in areas such as siting, emergency planning zones, and staffing requirements described in subsequent sections of proposed subpart F.

Proposed § 53.715 would provide the requirements for developing and implementing a program to do the following: (1) control maintenance activities; (2) take appropriate corrective action when performance issues are identified; (3) conduct routine evaluations of effectiveness; and (4) assess and manage risks resulting from maintenance activities. These proposed requirements are similar to those included in § 50.65 ("maintenance rule"). While, for the maintenance rule, specific criteria must be developed to capture both safety related and non-safety-related but otherwise important SSCs, the proposed § 53.715 would cover safety-related SSCs and NSRSS consistent with other subparts in Framework A.

Proposed § 53.720 would provide the requirements for responding to a seismic event during the operating phase of the life cycle of a commercial nuclear plant and would be equivalent to the requirements in paragraph IV(a)(3) of appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to part 50.

The proposed part 53 would include a framework to address staffing, training, personnel qualifications, and human factors engineering in a manner that is risk informed, technology inclusive, performance based, and flexible in nature. The underlying approach was detailed in the white paper on "Risk-Informed and Performance-Based Human-System Considerations for Advanced Reactors" (ADAMS Accession No. ML21069A003). Key considerations within both frameworks include the recognition that staffing, operator qualifications, and human factors engineering are interconnected areas that must be approached in an integrated manner and, furthermore, that safety functions, including the means by which they are fulfilled, provide an effective method for informing technology-inclusive requirements. These requirements would be structured to be common to both Frameworks A and B, and proposed language that references requirements unique to each framework (e.g., for change control processes) would be used where appropriate.

The requirements associated with this approach would be in §§ 53.725 through 53.830. Section 53.725 discusses applicability and defines specific terms. Some definitions draw from those in § 55.4. Several new definitions would be introduced for use within the context of subpart F. These new definitions are the following: A*utomation*, *Auxiliary operator*, *Generally licensed reactor operator*, *Load following*, *Self-reliant mitigation facility*.

Sections 53.725 to 53.830 would be divided into four portions that would cover general operational requirements, operator and senior operator licensing requirements, generally licensed reactor operator requirements, and general training requirements for plant staff. The NRC intends to provide guidance addressing the review of operator staffing plans; the review of operator, senior operator, and generally licensed reactor operator examination programs; and the implementation of scalable human factors engineering reviews.

Certain routine communications are necessary to facilitate the application process for individual operator licenses. The NRC is proposing to adapt the requirements of § 55.5 in § 53.726 to accomplish this.

Specific information must be collected in order to facilitate the initial issuance of operator licenses, as well as to allow for license renewals and required updates thereafter. Such information collection activities must also be approved of by the OMB. The NRC is proposing to adapt the requirements of § 55.8, to include any needed updates in OMB approval information, in § 53.727 to accomplish this.

The information used within the regulatory processes of the NRC must be free from omissions and inaccuracies to facilitate effective regulation. Consistent with this, the NRC is proposing to adapt the requirements of § 55.9 in § 53.728 to require the completeness and accuracy of material information provided by individual applicants and license holders.

Section 53.730 would provide performance-based and technology-inclusive requirements for assessing the role of personnel in facility safety, applying human-system considerations within facility design, and incorporating operational approaches that are consistent with design-specific safety considerations. These provisions would apply to facilities licensed under both Framework A and Framework B (it should be noted that proposed § 53.4220 of Framework B states the rules in §§ 53.725 through 53.830 would apply under Framework B as well). Most of these requirements would be adapted from portions of §§ 50.34(f) and 50.54 and part 55, with considerable modification in order to reflect the introduction of new technologies, potential attributes of advanced nuclear reactors, and possible changes in the roles of personnel in preventing and mitigating events. The NRC is proposing that these technical requirements would, together, serve as a component of the required content of applications for OLs and COLs under part 53. Additionally, the NRC proposes that the specific technical requirements associated with human factors engineering, human-system interface design, concept of operations, functional requirements analysis, and function allocation would serve as a component of the required content of applications for both standard design certifications and standard design approvals as well.

Human factors engineering is essential to facilitate the role of personnel in facility safety in a manner that is both effective and reliable. The NRC proposes to adapt § 53.730(a) from the human factors engineering design requirements of § 50.34(f)(2)(iii). A key difference would be that the requirement would now be focused on settings where personnel fulfil their safety or emergency response roles, wherever that may occur. The NRC additionally proposes to include within the scope of this requirement activities for assuring the continued availability of plant equipment that is needed for safety and envisions that this may encompass relevant maintenance, inspections, and testing as well. The NRC intends that this requirement would be associated with staff guidance for conducting scalable reviews of human factors engineering that is planned to accompany part 53.

Human-system interfaces provide vital information to operators across a spectrum of operating conditions that can range from normal operations through severe accident conditions. The specific types of information that must be available to support operations staff during such conditions include, in part, those associated with safety function parameters, safety system status, possible core damage states, barrier integrity, and radioactive leakage. Due to the importance of such information, the NRC proposes under § 53.730 (b) to require such human-system interface design features for all facilities, irrespective of other flexibilities proposed under part 53. Therefore, the NRC proposes to adapt specific post-Three Mile Island requirements of § 50.34(f) in a technology-inclusive manner as detailed in the following:

- Paragraph (b)(1) would be adapted from § 50.34(f)(2)(iv).

- Paragraph (b)(2) would be adapted from § 50.34(f)(2)(v).
- Paragraph (b)(3) would be adapted from § 50.34(f)(2)(xi), 50.34(f)(2)(xii), and 50.34(f)(2)(xxi).
- Paragraph (b)(4) would be adapted from § 50.34(f)(2)(xvii), 50.34(f)(2)(xviii), 50.34(f)(2)(xix), and 50.34(f)(2)(xxiv).
- Paragraph (b)(5) would be adapted from § 50.34(f)(2)(xxvi).
- Paragraph (b)(6) would be adapted from § 50.34(f)(2)(xxvii).

In addition to the requirements of § 53.730(b)(1) through (6), a further set of human-system interface design requirements applicable only to those facilities that will be staffed by generally licensed reactor operators would be provided under § 53.730 (b)(7). This prescriptive set of design requirements for such facilities that demonstrate compliance with the criteria of § 53.800 would recognize that the application of human factors engineering under § 53.730(a) is anticipated to be significantly reduced at such facilities in the absence of an expected operator role for the fulfillment of safety functions. However, it should be noted that the capability for an immediately initiated, manual reactor shutdown would be conservatively mandated irrespective of any other design considerations.

The NRC proposes § 53.730I(c) to require the submittal of a concept of operations that is of sufficient scope and detail to appropriately inform the staff. The development of a concept of operations can facilitate a clear understanding on the part of the NRC staff for potential novel operating concepts. Additionally, such information is likely to reduce the degree of resources and interactions needed for the staff to obtain the understanding necessary to enable flexible requirements in areas such as staffing, operator qualifications, and human factors engineering.

The NRC proposes under § 53.730(d) to require the submittal of both a Functional Requirements Analysis and a Function Allocation. The identification of design-specific safety functions and how they are fulfilled serves as a primary means for achieving technology-inclusive requirements within areas such as staffing, operator qualifications, and human factors engineering. The Functional Requirements Analysis and Function Allocation processes (which are both human factors engineering methods derived from systems engineering principles), provide an effective means to identify both how safety functions will be satisfied and to characterize any associated operator role in doing so. Functional Requirements Analysis shows what features, systems, and human actions are relied upon to demonstrate safety (i.e., fulfill safety functions). A Function Allocation, in turn, then describes how safety functions are assigned to both personnel and automatic systems. However, an important adaptation of the Function Allocation for use under the proposed rule would be the further need to not only describe allocations of safety functions to human action and automation, but also to identify allocations made to active safety features, passive safety features, or inherent safety characteristics as well.

Operating experience provides an important source of information by which to inform various aspects of facility design and operations. Accordingly, the NRC proposes in § 53.730I(e) to adapt the requirements of § 50.34(f)(3)(i) for requiring an operating experience program.

Advanced reactors are expected to have technological considerations and concepts of operations that are more conducive to customizable licensed operator staffing requirements than the prescriptive requirements of § 50.54(m). Analyses and assessments that are based on human factors engineering principles provide a performance-based means of determining licensed operator and senior operator staffing needed to support safe operations. In contrast, for those facilities required to be staffed by generally licensed reactor operators, the NRC expects that the operator staffing plans will reflect a simpler approach of showing that a continuity of responsibility will be maintained for facility operations throughout the operating phase, with at least one generally licensed reactor operator providing continuous oversight and remaining immediately available when any units are fueled. Additionally, a revised approach to the traditional position of the shift technical advisor that focuses on the timely availability of engineering expertise as a means of addressing uncertainties and abnormal circumstances is more suitable within the context of advanced reactors and is intended to be applicable to all facilities, irrespective of other design and staffing considerations. Consistent with this approach, the NRC proposes under § 53.730(f) to require the submittal of a staffing plan that details operations staffing, how engineering expertise will be provided, and what staffing will be available to provide other needed support functions. The NRC intends that this requirement would be associated with staff guidance for reviewing operations staffing plans that is planned to accompany part 53 and that, following NRC approval, the staffing plan would become a condition of the facility license and subject to appropriate change control.

Operator training and qualification programs provide an essential component of supporting human performance in implementing tasks with safety implications. Such programs must include components that cover the stages of initial training, examination, and continuing training. Additionally, recognizing the potential for varying concepts of operations to affect traditional, prescriptive approaches to operator proficiency, the NRC proposes under part 53 to allow facilities to develop operator proficiency programs based on facility-specific considerations. Therefore, the NRC proposes in § 53.730(g)(1) to require approval of the programs that will be used for the initial training, initial examination, requalification training and examination, and proficiency of both licensed operators and senior operators. In a corresponding manner, the NRC proposes in § 53.730 (g)(2) to require approval of the programs that will be used for the generally licensed reactor operator equivalents of each of these programs for facilities with such staffing. The NRC intends that examination program requirements would be associated with staff guidance for the review of tailored examination processes that are planned to accompany part 53 and that, following NRC approval, both the training and examination programs would be subject to appropriate change controls and NRC oversight. Following the completion of an initial training program, continuing training programs provide an important means of sustaining the knowledge and abilities of individuals. The NRC is proposing to adapt the requirements of § 50.54(i through 1) in paragraph (g)(3) to require that operator continuing training programs be in effect to support operator performance. Under part 53, the NRC proposes to require these programs to be in effect concurrent with when the initial operator examinations first commence, in effect putting the programs in place only when they are needed; this represents a modification of the comparable requirement of § 50.54(I through 1), which links the commencement of these programs to a timeline driven by the licensing of the facility.

The authorization to manipulate controls of the facility that directly affect reactivity or power level is restricted to individuals who are either licensed operators, licensed senior operators, or generally licensed reactor operators. However, for practical purposes, situations in which an individual is participating in an approved training program or reestablishing proficiency may also require them to operate the controls of the facility under the cognizance of a licensed individual. The NRC is proposing to adapt the requirements of § 55.13 in § 53.735 to accomplish this, with a notable difference being the incorporation of the ability of generally licensed reactor operators to constitute one of the classes of individuals who are authorized to manipulate the controls of the facility.

Section 53.740 would provide general facility licensee (i.e., OL and COL holders under part 53) requirements. Portions of § 53.740 would be adapted from the conditions of facility licenses contained under § 50.54. In general, the conditions for operations staffing under part 53 would reflect considerations for potential technological differences and varying concepts of operation that are expected among advanced reactor applicants and licensees. Additionally, certain requirements would be specific to the operating phase while the conditionality of others would be such that they remain in effect following the permanent cessation of facility operations during the decommissioning phase.

All commercial nuclear plants licensed under part 53 would require some form of licensed operator staffing, whether it be by specifically or generally licensed operators. Consistent with this, the NRC is proposing under § 53.740(a) to require facility licenses to demonstrate compliance with the programmatic requirements for either specifically licensed operators and senior operators or for generally licensed reactor operators, as applicable to the facility.

The NRC recognizes that facility staffing at advanced reactors will need to account for a potentially wide range of concepts of operations; for this reason, flexible and performance-based approaches for establishing required facility staffing are appropriate. However, once the appropriate facility staffing has been determined and approved by the NRC, such staffing must be maintained to ensure that the appropriately qualified individuals will be available when needed to support the safe operation of the facility. Thus, the NRC is proposing under § 53.740(b) to require that the staffing described within the approved facility staffing plan be maintained as a condition of the facility license as opposed to prescriptive staffing requirements like those of § 50.54(k) and (m).

Due to their direct effects on reactivity or power level, the operation of facility controls must be restricted to only those individuals who possess appropriate levels of qualification and authorization. The NRC is proposing to adapt the requirements of § 50.54(i) in § 53.740I(c) to require that only specifically licensed operators and senior operators or, alternatively, generally licensed reactor operators, may operate facility controls, with allowance for specified exceptions for the purposes of operator training or proficiency.

Senior operators, by virtue of their license level, are qualified and authorized both to perform certain important administrative responsibilities and to direct the licensed activities of licensed operators. Therefore, facilities that are required to be staffed by specifically licensed operators must also include senior operators within their staffing. In contrast, facilities staffed with generally licensed reactor operators only have a single license level available and, therefore, there is no equivalent provision for such facilities. The NRC is proposing to adapt the requirements of § 50.54(l) in § 53.740(d) to require the licensing and designation of senior operators at facilities staffed by specifically licensed operators.

As detailed in the white paper on "Risk-Informed and Performance-Based Human-System Considerations for Advanced Reactors," load following where plant output automatically changes in response to externally originated instructions or signals is not permitted under the existing regulations of § 50.54. However, new technological considerations and concepts of operation may justify such an operational approach under appropriate circumstances. The NRC recognizes that, beyond electrical power generation, load following may also affect other applications of plant output, such as hydrogen production, desalination, or district heating. For load following to be permissible, measures must be in place to provide assurance that plant output considerations are not permitted to lead to challenges to safe reactor operations; these measures may consist of automated control systems, automatic protective features, or the continuous oversight and immediate intervention capability of an appropriately qualified and authorized individual. The NRC is proposing to adapt the requirements of § 50.54(j) in § 53.740(e) to require appropriate oversight of operations, other than those associated with the controls themselves, that may affect reactivity or power level. Section 53.740(f) would provide an allowance for load following, provided that appropriate measures are in place. In considering the acceptability of the measures associated with load following, the NRC expects that any automatic protection relied upon would be separate from that credited for reactor protection purposes and would employ setpoints that are sufficiently conservative as to prevent actuation of the credited protection systems.

Core alterations, such as those which occur during refueling operations, are associated with specific considerations that warrant limiting the oversight of such operations to only appropriately qualified and authorized individuals. Unlike other types of fuel handling operations, core alterations are expected to occur within the confines of a reactor vessel that is specifically designed to support and sustain nuclear criticality, thereby justifying the imposition of higher qualification levels within such contexts. The NRC is proposing to adapt the requirements of § 50.54(m)(2)(iv) in § 53.740(g) to require the supervision of core alterations by either a specifically licensed senior operator or by a generally licensed reactor operator, as applicable to the facility. Because certain advanced reactor designs may be capable of refueling while at power and, in any event, overall facility oversight would already be required by either a specifically licensed senior operator or by a generally licensed reactor operator, the NRC proposes to not impose this requirement during periods where core alterations occur while the plant is operating.

It is impossible to predict every possible scenario that a commercial nuclear plant might potentially encounter. Therefore, it is prudent to grant the authority for appropriately qualified individuals to depart from facility license conditions when emergency circumstances dictate that doing so is in the interest of public health and safety. The NRC is proposing to adapt the requirements of § 50.54(x) and (y) in § 53.740 (h) to permit specific individuals to authorize departures from facility license conditions when emergency conditions warrant doing so for the protection of the public health and safety. Recognizing that certain facilities licensed under part 53 may be staffed by generally licensed reactor operators in lieu of specifically licensed senior operators, the NRC proposes to extend this authority to generally licensed reactor operators.

Due to the unique authorities and responsibilities of both specifically and generally licensed operators, it is essential that any individual fulfilling such a role have demonstrated compliance with the regulatory requirements for operator licensing. The NRC is proposing to adapt the requirements of § 55.3 in § 53.745 to require that any person performing the function of an operator, senior operator, or generally licensed reactor operator must be authorized by a license issued by the Commission.

Licensees are required to comply with all regulations associated with their licenses. The NRC has authority under the AEA to take enforcement actions for violations of regulatory requirements on the part of licensees. The NRC is proposing to adapt the requirements of § 55.71 in § 53.750 to address such violations and associated enforcement actions.

The AEA provides for criminal penalties for violations of certain regulations that are issued by the NRC. Individuals who violate such regulations may be subject to criminal sanctions under the AEA because of their actions. The NRC is proposing to adapt the requirements of § 55.73 in § 53.755 to address such criminal penalties.

The NRC proposes to license individuals as operators under both specific and general licensing frameworks. Specific licenses would be utilized within the context of licensed operators (i.e., reactor operators) and senior operators (i.e., senior reactor operators), would be issued to a named person, and would be effective upon approval by the Commission of an application filed pursuant to the regulations in this part and issuance of licensing documents to the applicant. In contrast, generally licensed reactor operators would perform duties under the provisions of a general license that would be effective without the filing of an application with the Commission or the issuance of licensing documents to a particular person. The NRC proposes requirements for the specific licensing process for licensed operators and senior operators under §§ 53.760 through 53.795, with § 53.760 addressing applicability.

Medical fitness is an important component of the overall process of specifically licensing operators because it provides assurance that operators will be able to carry out important duties without being precluded from doing so by health-related issues, as well as providing assurance that such issues will not adversely affect the performance of assigned job duties or cause operational errors that endanger public health and safety. In addition to a requirement for medical fitness, a medical examination by a physician to confirm compliance with this requirement is necessary. The NRC is proposing to adapt the requirements of §§ 55.21, 55.23, 55.27, and 55.33 under § 53.765 to require medical fitness, examinations by physicians, and medical certification for specifically licensed operators and senior operators. In recognition of the fact that generally licensed reactor operators are not expected to have a role in the fulfillment of safety functions at the facilities at which they are licensed, the NRC also proposes to not extend a comparable medical requirement to generally licensed reactor operators.

Medical fitness also constitutes an integral part of the licensing decisions that the NRC makes for specifically licensed operators because of the associated potential for health-related effects on the abilities of individuals to carry out important duties. Therefore, the NRC is proposing to adapt the requirements of §§ 55.25 and 50.74(c) in § 53.770 to require that timely notifications be made to the NRC if a specifically licensed operator or senior operator develops a permanent physical or mental condition that adversely affects the performance of assigned operator job duties or could cause operational errors endangering public health and safety. Notwithstanding this requirement related to permanent medical conditions, the NRC continues to recognize that it is appropriate for facility licenses to impose administrative restrictions and conditions upon specifically licensed operators and senior operators in response to temporary medical conditions.

The process of specifically licensing individuals as licensed operators or senior operators requires the submittal of applications to the NRC for review. These applications must detail certain elements associated with licensing, including the demonstration of compliance with examination, experience, and medical requirements. The NRC is proposing to adapt the requirements of §§ 55.31 through 55.35 in § 53.775 to include requirements for the applications associated with the specific licensing of licensed operators and senior operators at commercial nuclear plants licensed under part 53. In contrast with the part 55 requirements, the NRC proposes to locate certain details associated with the preparation and submittal of these applications within guidance documentation in lieu of placement within this proposed rule itself.

The NRC proposes overall programmatic requirements for specifically licensed operator and senior operator training, examination, and proficiency in § 53.780. In general, the proposed requirements are adapted from those in part 55, with several additional flexibilities being incorporated to better account for potential variations in advanced reactor technologies and concepts of operations. The requirements proposed in § 53.780 cover, in part, the initial training, initial examination, requalification training, requalification examination, and proficiency of specifically licensed operators and senior operators.

The initial training process provides individuals with the knowledge and abilities needed to subsequently fulfil assigned duties as licensed operators or senior operators in a safe and reliable manner. The use of a systems approach to training (SAT) ensures that the training program is based upon job requirements in a manner that can be adapted to account for differences in plant technology, concepts of operations, and operator roles in the fulfillment of design-specific safety functions. The NRC is proposing under § 53.780(a) to require facility licensees to implement a SAT-based training program for the initial training of licensed operator and senior operator applicants that is adequate to ensure that applicants will be capable of performing the duties necessary both to protect the public health and safety and to maintain plant safety functions. The NRC further proposes that such programs be subject to NRC approval and subsequent change control processes of an appropriate nature.

Examinations provide a means of assessing that individuals have achieved a degree of knowledge and ability that will be sufficient to enable them to carry out assigned duties as licensed operators or senior operators in a manner that is safe and reliable. The NRC is proposing to adapt the requirements of §§ 55.40, 55.41, 55.43, and 55.45 in § 53.780(b) to require that facilities establish and implement an initial examination program. However, a key difference from the comparable requirements of part 55 would be that facilities have the flexibility to propose, subject to NRC approval, the examination methods and criteria to be used in assessing satisfactory applicant performance. The NRC intends that staff guidance would be available to facilitate the review of licensing examination programs that are proposed by facility licensees and that, following NRC approval, initial examination programs would be subject to an appropriate change control process. Furthermore, the NRC proposes that facility licensees under part 53 be provided the alternative of administering their own approved licensing examinations, with the NRC continuing to exercise appropriate oversight of the program, make licensing decisions based upon the examination results, and reserving the right to elect to administer the examinations in lieu of permitting the facility to do so. However, irrespective of the provided flexibilities in examination format and structure, the NRC expects that, at a minimum, topics from the following general categories of knowledge and abilities would be sampled in such examinations:

- Reactor Theory and Thermodynamic Principles
- Plant Systems and Components

- Reactivity Management and Manipulations
- Radiation Control and Safety
- Emergency, Abnormal, and Normal Operations
- Administrative Requirements and Conditions of the Facility License

Requalification training programs provide for the continuing training and examination of specifically licensed operators and senior operators to ensure that they maintain at acceptable levels the knowledge and abilities needed to support the safe and reliable performance of job duties following the completion of an initial training and examination program. The NRC is proposing to adapt the requirements of § 55.59 in § 53.780(c) to require that facilities implement both a SAT-based requalification training program and a biennial requalification examination program. However, a notable difference from the biennial requalification examinations required under part 55 would be that distinct annual operating test and biennial written examination components would not be mandated, with the facility licensee instead proposing the examination methods and criteria to be used in assessing satisfactory performance. The NRC intends that staff guidance would be available to facilitate the review of the requalification examination programs that are proposed by facility licensees and that, following NRC approval, requalification examination programs would be subject to an appropriate change control process.

For examinations to provide for valid assessments of the knowledge and abilities of individuals, the examinations must remain free from compromises that could affect their underlying integrity. The NRC is proposing to adapt the requirements of § 55.49 in § 53.780(d) to require that examinations and related activities remain free from any compromise that might affect the integrity of the examination process.

Simulators provide a valuable means of training and evaluating plant operators, and the NRC is specifically authorized under the Nuclear Waste Policy Act of 1982, as amended (NWPA), section 306 (42 U.S.C. 10226) to establish regulations for the use of simulators within such context. The NRC is proposing to adapt the requirements of § 55.46 in § 53.780(e) to address the use of simulation facilities for training, examinations, and applicant experience requirements. However, the proposed requirements of part 53 would not mandate that full scope, plant-referenced simulators be used and would allow for the potential use of alternative simulation facilities consisting of, for example, partial scope simulators or the plant itself, provided that all associated requirements can be demonstrated to be met using alternative approaches and methods. Additionally, in allowing for the possibility that an applicant or licensee might demonstrate compliance with training, examination, or experience requirements using the plant itself, it is not the intention of the NRC to allow for the initiation of transients on the actual plant. Consistent with this, aside from controlled reactivity manipulations that are conducted for the purposes of demonstrating compliance with experience requirements, actual plant components should not be operated for these purposes. Rather, the NRC perspective is that the use of the plant for training and examination purposes should be restricted to techniques such as walkthroughs, job performance measures, simulated tasks, use of augmented reality technology, and similar approaches that provide training and examination value while avoiding the operation of actual plant components.

There may be situations in which applicants for operator or senior operator licenses have previous training and experience that justifies waiving some, or all, of the examination requirements. The NRC is proposing to adapt the requirements of § 55.47

in § 53.780(f) to allow for consideration of requests for waivers of examinations requirements. In contrast with the part 55 requirements, the NRC proposes to locate certain details associated with such waiver requests within guidance documentation in lieu of placement within the rule itself.

For licensed operators and senior operators to perform their assigned duties safely and reliably, it is essential that they perform those duties frequently enough so as to maintain a sufficient degree of proficiency. The NRC is proposing to adapt the requirements of § 55.53(e) and (f) in § 53.780(g) to require that specifically licensed operators and senior operators maintain proficiency and, if proficiency is not maintained, regain proficiency prior to resuming licensed duties. However, in recognition of the fact that varying concepts of operations are possible for advanced reactor facilities, the NRC is proposing, in contrast with the requirements of part 55, to allow facility licensees to establish their own programs for operator proficiency, subject to NRC approval.

As the holders of specific licenses, licensed operators and senior operators must be subject to license conditions on an individual basis to ensure that the basis upon which the licenses were issued remains valid. The NRC is proposing to adapt the requirements of § 55.53 in § 53.785 to require appropriate conditions of licenses for specifically licensed operators and senior operators. However, in contrast with the requirements of § 55.53(e) and (f), the NRC is proposing to allow certain aspects of operator proficiency to be addressed by an NRC-approved facility proficiency program.

Licenses for specifically licensed operators and senior operators are issued by the NRC and must remain subject to modification or revocation. The NRC is proposing to adapt the requirements of §§ 55.51 and 55.61 in § 53.790 to address the issuance, modification, and revocation of licenses issued to specifically licensed operators and senior operators.

The licenses issued to specifically licensed operators and senior operators are valid for a period of six years, after which they expire, unless otherwise renewed. The NRC is proposing to adapt the requirements of §§ 55.55 and 55.57 in § 53.795 to address the expiration and renewal of licenses issued to specifically licensed operators and senior operators.

The white paper on "Risk-Informed and Performance-Based Human-System Considerations for Advanced Reactors" explored, in part, considerations that would justify the omission of the specifically licensed operators and senior operators. However, even for an inherently safe reactor with autonomous operation features, certain important administrative functions would still need to be accomplished by appropriately qualified and authorized individuals. Additionally, the staff further recognized that manual manipulations of facility reactivity controls must only be performed by individuals who have been appropriately licensed by the Commission. The NRC therefore proposes under § 53.800 to establish a new class of facility (defined as a self-reliant mitigation facility), according to the criteria contained in § 53.800(a)(1) and (a)(2) for Frameworks A and B, respectively. These facilities would utilize generally licensed reactor operators in lieu of specifically licensed operators and senior operators. The NRC proposes two parallel constructs within subpart F that would establish both of these operator qualification and licensing approaches. The generally licensed reactor operator framework offers enhanced flexibilities and targeted relaxations in a manner that is commensurate with the modified role of such operators to ensure the safe operation of the associated facilities.

A key determinant as to whether generally licensed operators can be utilized in facility staffing is the assessment of the operator’s role in maintaining and fulfilling safety functions at the facility, such as through the performance of credited actions for the mitigation of plant events. Specifically generally licensed operators would differ in that only the specifically licensed operators would be directly and independently evaluated by the NRC as part of their licensing process; this direct and independent evaluation remains appropriate when operators may reasonably be expected to exert a significant influence on public health and safety outcomes. The sets of criteria proposed in § 53.800 for use under Frameworks A and B would designate self-reliant mitigation facilities. However, they have each been adapted for use within their respective frameworks (including for those Framework B facilities employing an AERI approach). Each of these sets of criteria are derived from a common set of considerations:

- no human action needed to satisfy radiological consequence criteria;
- no human action needed to address LBEs;
- safety functions not allocated to human action;
- reliance upon either inherent or robust passive features; and
- adequate defense in depth achieved without reliance on human action.

Generally licensed reactor operators would perform duties under the provisions of a general license that would be effective without the filing of an application with the Commission or the issuance of licensing documents to a particular person. The NRC proposes requirements for the general licensing process for generally licensed reactor operators under §§ 53.805 through 53.820. The framework for generally licensed reactor operators would parallel that of senior operators in certain regards owing to their comparable administrative responsibilities. Nonetheless, the requirements for generally licensed reactor operators would be comparatively relaxed and incorporate greater flexibilities in a manner that is consistent with the generally licensed reactor operator's role in safety at self-reliant mitigation facilities.

In order to use generally licensed reactor operators in lieu of specifically licensed operators and senior operators, a self-reliant mitigation facility licensee would need to demonstrate compliance with the following requirements on an ongoing basis: maintaining generally licensed reactor operator qualifications for the performance of important functions and tasks; incorporating relevant programmatic controls into TS; administering the related programs for training, examination, and proficiency; and ensuring that the relevant provisions of parts 26 and 73 are met. Additionally, to provide for an accurate accounting of what individuals are licensed under the general license, facility licensees would be required to report the identities of all generally licensed operators to the NRC on an annual basis. The NRC therefore proposes under § 53.805 to establish requirements for facility licensees that address issues such as these.

The NRC proposes the general license for generally licensed reactor operators under § 53.810. Generally licensed reactor operators would be licensed as a class of individuals under the provision of § 53.810 (a) and would be subject to the conditions specified in § 53.810(b) through (g). Portions of these conditions are adapted from both those contained under § 55.53 and from those conditions currently included in the licenses issued to specifically licensed operators and senior operators. The NRC would retain the ability to suspend or prohibit individuals from operating under the general license should such action be warranted.

The NRC proposes overall programmatic requirements for generally licensed reactor operator training, examination, and proficiency under § 53.815. In general, these proposed requirements are adapted from those of part 55 and parallel those also proposed for specifically licensed senior operators in § 53.780. These requirements include increased flexibilities and several targeted relaxations that reflect the modified role of generally licensed reactor operators in facility safety. The requirements proposed under § 53.815 cover, in part, the initial training, initial examination, continuing training, requalification examination, and proficiency of generally licensed reactor operators. Section 53.805 would require the facility licensee to develop, implement, and maintain these programs. Section 53.810, in turn, would prescribe that the requirements of § 53.805 would need to be met as a requirement of the general license. The implication of this structure is that the facility licensee would need to implement these programs for training, examination, and proficiency, and generally licensed reactor operators would need to participate in these programs to demonstrate compliance with the requirements of their respective licenses.

The initial training process provides generally licensed reactor operators with the knowledge and abilities needed to fulfil assigned duties as generally licensed reactor operators. The use of a SAT serves to ensure that the training program is based upon job requirements in a manner that can be adapted to account for differences in plant technology and concepts of operations. The NRC is proposing under § 53.815(b) to require facility licensees to implement a SAT-based training program for the initial training of generally licensed reactor operators that is adequate to ensure that they have the necessary knowledge, skills, and abilities to perform their duties. The NRC further proposes that such programs would be subject to NRC approval, oversight, and appropriate change control processes. The training program must ensure that generally licensed reactor operators maintain the necessary knowledge, skills, and abilities.

Examinations provide a means of assessing that individuals have achieved a degree of knowledge and ability that will be sufficient to enable them to carry out assigned duties as generally licensed reactor operators in a manner that is both safe and reliable. The NRC proposes to adapt the requirements of §§ 55.40, 55.41, 55.43, and 55.45 in § 53.815(b) to require that facilities establish and implement an initial examination program. A key difference from the comparable requirements of part 55 would be that facilities would be afforded the flexibility to propose, subject to NRC approval, the examination methods and criteria to be used in assessing satisfactory individual performance. The NRC intends that staff guidance would be available to facilitate the review of initial examination programs that are proposed by facility licensees and that, following NRC approval, initial examination programs would be subject to an appropriate change control process. In contrast with both the requirements of part 55 and the proposed requirements of § 53.780, the NRC does not intend to administer or evaluate these initial examinations. Irrespective of the provided flexibilities in examination format and structure, the NRC expects that, at a minimum, topics from the following general categories of knowledge and abilities would be sampled in such examinations:

- Reactor Theory and Thermodynamic Principles
- Plant Systems and Components
- Reactivity Management and Manipulations
- Radiation Control and Safety
- Emergency, Abnormal, and Normal Operations
- Administrative Requirements and Conditions of the Facility License

Continuing training programs provide the ongoing training and examination of generally licensed reactor operators to ensure that they maintain at acceptable levels the knowledge and abilities needed to support the safe and reliable performance of job duties following the completion of an initial training and examination program. The NRC is proposing to adapt the requirements of § 55.59 in § 53.815(b) to require that facilities implement both a SAT-based continuing training program and a requalification examination program. However, a notable difference from the examinations required under part 55 would be that distinct annual operating test and biennial written examination components would not be mandated. The facility licensee would instead propose examination methods and criteria to be used in assessing satisfactory performance. Furthermore, unlike the comparable requirements of part 55 and those proposed for specifically licensed operators and senior operators, a biennial periodicity for requalification examinations would not be prescribed. However, adequate justification for the proposed periodicity of requalification examinations would be required. The NRC intends that staff guidance would be available to facilitate the review of the requalification examination programs that are proposed by facility licensees. Following NRC approval, requalification examination programs would be subject to an appropriate change control process.

For examinations to provide for valid assessments of the knowledge and abilities of individuals, the examinations must remain free from compromises that could affect their underlying integrity. The NRC is proposing to adapt the requirements of § 55.49 in § 53.815(d) to require that examinations and related activities remain free from any compromise that might affect the integrity of the examination process.

Simulators provide a valuable means of training and evaluating plant operators and the NRC is specifically authorized under the NWPA, section 306 (42 U.S.C. 10226) to establish regulations for the use of simulators within such context. The NRC is proposing to adapt the requirements of § 55.46 in § 53.815(e) to address the use of simulation facilities for training and examinations, and experience requirements. The use of full scope, plant-referenced simulators would not be mandated. The potential use of alternative simulation facilities consisting of, for example, partial scope simulators or the plant itself, would be allowed provided that all associated requirements could be demonstrated to be met using alternative approaches and methods. Additionally, in allowing for the possibility that an applicant or licensee might demonstrate compliance with training and examination requirements using the plant itself, it is not the intention of the NRC to allow for the initiation of transients on the actual plant. Consistent with this, aside from controlled reactivity manipulations that are conducted for the purposes of demonstrating compliance with experience requirements, actual plant components should not be operated for these purposes. Rather, the use of the plant for training and examination purposes should be restricted to techniques such as walkthroughs, job performance measures, simulated tasks, use of augmented reality technology, and similar approaches that provide training and examination value while avoiding the operation of actual plant components.

There may be situations in which generally licensed reactor operators have previous training and experience that justifies waiving some, or all, of the examination requirements. Therefore, the NRC is proposing under § 53.815(f) to allow facility licensees to waive examination requirements provided that such waivers are consistent with a program that has been approved by the NRC.

For generally licensed reactor operators to safely and reliably perform their assigned duties, it is essential that they perform those duties frequently enough so as to maintain a sufficient degree of proficiency. However, the NRC recognizes that facilities that utilize generally licensed reactor operators may have concepts of operation that warrant unique proficiency considerations. Therefore, the NRC is proposing in § 53.815(g) to require that facility licensees develop, implement, and maintain programs to maintain and reestablish, if needed (e.g., when an individual's extended absence from watch standing has rendered proficiency requirements unmet), the proficiency of generally licensed reactor operators.

The general license for an individual as a generally licensed reactor operator should remain in effect only while that individual remains employed in a position that may require them to manipulate the reactivity controls of the facility. The NRC proposes under § 53.820 to require the expiration of the general license for an individual when that individual's employment status becomes such that this is no longer the case. However, the NRC recognizes that for some types of self-reliant mitigation facilities, very long periods may elapse between circumstances that necessitate manual manipulation of reactivity controls. Therefore, the NRC's perspective is that, for the purposes of this proposed requirement, it is sufficient that an individual's current position could potentially require that individual to manipulate reactivity controls at some point within the course of their assigned job duties.

The NWPA, section 306 (42 U.S.C. 10226) authorizes and directs the NRC to, in part, issue regulations and guidance that address the training and qualifications of civilian nuclear power plant operators, supervisors, technicians, and other appropriate operating personnel. The NRC implements this in part 50 through the requirements of

§ 50.120. The NRC is proposing under § 53.830 to adapt the requirements of § 50.120 for use in part 53 to provide personnel training and qualification requirements that have greater flexibility and better reflect diverse concepts of operations.

The NRC recognizes that the categories of nuclear power plant personnel in § 50.120 may not be appropriate for diverse concepts of operations, staffing models, and non-traditional personnel roles and responsibilities. The NRC also recognizes that the timeframe prescribed in § 50.120 for the establishment of training programs may not be aligned with the schedules associated with the startup of certain types of commercial nuclear plant facilities. At the same time, however, the white paper on “Risk-Informed and Performance-Based Human-System Considerations for Advanced Reactors,” determined that the SAT-based training required under § 50.120 remains an appropriate means by which training programs should continue to be developed and implemented.

The NRC is proposing under § 53.830 to require SAT-based training programs with the timeframe for when such programs are required being based upon when the associated personnel are needed to support facility-specific needs. The training programs would cover the training and qualification of plant personnel in the general categories of supervisors, technicians, and other appropriate operating personnel. The training programs would not undergo approval by the Commission; however, they would be subject to periodic NRC inspection. The NRC intends to develop guidance to facilitate the inspection of these training programs but does not intend for such guidance to preclude the potential for the training programs to be maintained by a separate, NRC-approved accreditation process.

Proposed § 53.845 would require programs to be developed, implemented, and maintained to help ensure that design features and human actions have the necessary capabilities and reliabilities needed to demonstrate compliance with the safety criteria in subpart B throughout the operating life of each commercial nuclear plant. The proposed programmatic requirements in subpart F would also address areas such as radiation protection that are needed to control routine effluents during normal operations. The proposed §§ 53.850 through 53.910 would require programs to support specific activities needed to ensure the appropriate prevention or mitigation of unplanned events or to support normal operations in areas applicable to any reactor design. However, each holder of an OL or COL would be required to assess whether additional programs are needed for the specific reactor design and location of the commercial nuclear plant. Licensees would be able to combine, separate, and otherwise organize programs and related documents as appropriate for the technologies and organizations associated with the commercial nuclear plant.

Proposed § 53.850 would require a radiation protection program associated with the requirements in subparts B and C for public doses resulting from normal operations and the protection of plant workers. The proposed requirements related to doses from normal operations, including routine effluents, would be similar to those specified in § 50.36a, "Technical specifications on effluents from nuclear power reactors," and related requirements in standard TS for offsite dose calculation manuals. While the proposed section would include requirements that are technically and programmatically similar to part 50, the proposed § 53.850 would not include a requirement for effluent-related TS as is required in § 50.36a. A proposed requirement similar to that found in the administrative controls section of TS for operating reactors licensed under parts 50 and 52 would be included for programmatic controls of solid wastes to complement the design requirements in proposed § 53.425.

Proposed § 53.855 would require an emergency response plan that demonstrates compliance with the requirements in appendix E to part 50 and the planning standards in § 50.47(b). The regulations in § 50.47 prescribe how the NRC makes licensing decisions using findings of reasonable assurance that adequate protective measures can and will be taken to protect public health and safety in the event of a radiological emergency. The proposed § 53.855 also relates to an ongoing rulemaking activity that would provide alternatives to certain elements of the existing regulations. The draft final rule is described in SECY-22-0001, "Final Rule: Emergency Preparedness for Small Modular Reactors and Other New Technologies," dated January 3, 2022. If the NRC proceeds with revising its regulations as described in SECY-22-0001, the same flexibility in determining appropriate emergency preparedness measures, as directed by the Commission, would be added to part 53. In its "Policy Statement on the Regulation of Advanced Reactors," the Commission stated their expectation that, "…the safety features of advanced reactor designs will be complemented by the operational program for Emergency Planning (EP). This EP operational program, in turn, must be demonstrated by inspections, tests, analyses, and acceptance criteria to ensure effective implementation of established measures." Consistent with this policy statement, emergency plans are not used to demonstrate compliance with the safety criteria in Subpart B. In SECY-97-020, "Results of Evaluation of Emergency Planning for Evolutionary and Advanced Reactors," dated January 27, 1997, the staff determined that the rationale upon which EP for current reactor designs is based, that is, potential consequences from a spectrum of accidents, is appropriate for use as the basis for EP for evolutionary and passive advanced LWR designs and is consistent with the Commission's defense-in-depth safety philosophy. Also, in its Safety

Goal Policy Statement, 51 FR 30028, August 21, 1986, the Commission stated that: "A defense-in-depth approach has been mandated in order to prevent accidents from happening and to mitigate their consequences. Siting in less populated areas is emphasized. *Furthermore, emergency response capabilities are mandated to provide additional defense-in-depth protection to the surrounding populations*." (emphasis added). Consistent with this policy statement, proposed § 53.855 contributes to defense in depth for commercial nuclear plants. The analyses that would be required in the proposed subpart C for unlikely and very unlikely event sequences would be used to support the determinations described in SECY-22-0001.

Proposed § 53.860 would identify the applicable regulations for part 53 applicants under Framework A related to the programs for physical security, cybersecurity, FFD, AA, and information security. These programs are discussed in more detail in section II.C.2 of this document.

Proposed § 53.810(a) would establish the physical protection program and present a graded approach to physical protection requirements. If a licensee can demonstrate compliance with the proposed criterion in § 53.860(a)(2)(i) (i.e., that potential consequences resulting from a design-basis threat (DBT) initiated event assuming licensee mitigation and recovery actions, including any operator action, are unavailable or ineffective, would result in offsite doses below the values in § 53.210), then the requirement to protect against the DBT of radiological sabotage would not be applicable. This proposal would apply a new regulatory approach for certain commercial nuclear plants in which the DBT of radiological sabotage would not be applicable. Section 170D.a of the AEA permits the Commission to determine the licensed facilities that are part of a class of licensed facilities where NRC-conducted Force-On-Force

(FOF) exercises are appropriate to assess the ability of a private security force of a licensed facility to defend against any applicable DBT. It would not be appropriate to conduct NRC-conducted FOF exercises to evaluate performance at commercial nuclear plants where the DBT of radiological sabotage is not applicable and the facility poses a lower risk to public health and safety from potential radiation exposure. Additionally, these facilities would have tailored security requirements and oversight consistent with their relatively low risk.

The proposed criterion would align with the offsite dose values used in the safety criteria for unplanned events in proposed § 53.210. Where the criterion is met, the resulting physical protection requirements would be those for protection of SNM and Category 1 and Category 2 radioactive material, if applicable.

For those licensees not able to demonstrate compliance with the criterion, proposed § 53.860(a) would permit the licensee to choose one of two paths to provide physical protection: (1) the current set of requirements in § 73.55, which would include any changes resulting from the ongoing proposed rulemaking on Alternative Physical Security Requirements for Advanced Reactors[4] that provides pre-determined physical security alternatives or (2) the performance-based requirements in proposed § 73.100.

Proposed § 53.860(b) would require licensees to establish, implement, and maintain a FFD program in accordance with part 26. Section 53.860(c) would require licensees to establish, implement, and maintain an AA program in accordance with either § 73.56 or proposed § 73.120, as appropriate. Section 53.860(d) would require licensees to establish, implement, and maintain a cybersecurity program in accordance

[4] SECY-22-0072, "Proposed Rule: Alternative Physical Security Requirements for Advanced Reactors," dated August 2, 2022.

with either § 73.54 or proposed § 73.110. SectiI53.860(e) would require licensees to establish, implement, and maintain an information protection system that demonstrates compliance with the requirements of §§ 73.21, 73.22, and 73.23, as applicable.

Proposed § 53.865 would establish requirements for quality assurance and refer to proposed subpart K for the part 53 requirements for safety-related design features. The proposed requirement for a quality assurance program would be similar to regulations in parts 50 and 52. Proposed requirements related to evaluating and reporting changes to the quality assurance program would be included in proposed subpart I and would be equivalent to those found in § 50.54.

The proposed § 53.870 would require licensees to actively assess possible degradation of SSCs from the effects of aging, fatigue, and environmental conditions. The proposed inclusion of requirements related to designing and monitoring for possible degradation mechanisms reflects important lessons learned from the history of light-water reactors and the likely introduction of new design features and materials in future commercial nuclear plants. The allowable combinations of design features, operating experience, testing, and monitoring during operations would support performance-based approaches to the initial licensing of new technologies. The proposed performance-based approach to integrity assessment programs would also allow for the subsequent consideration of operating experience and appropriate corrective actions or allowable relaxations for ensuring that design features demonstrate compliance with the proposed functional design criteria of §§ 53.410 and 53.420. The proposed program would be based upon a comprehensive and integrated evaluation of the aging and other degradation mechanisms applicable to the design; identification of the affected SSCs; the allowances provided in the design of the SSCs for degradation;

and schedules and procedures for determining if and at what rate degradation is occurring and what is the cause. Risk insights could be used to prioritize the monitoring, evaluation, and management of degradation based upon the importance of the SSC to safety and the time frame for when the effects of degradation could be of concern.

Proposed § 53.875 would establish requirements for a fire protection program supporting operations as is provided for by § 50.48. The proposed fire protection program during operations would work in concert with specific fire protection requirements proposed in subpart C for design and analyses and in proposed subpart E for construction and manufacturing.

Proposed § 53.880 would establish requirements for an inservice inspection and inservice testing program, which are historically important activities conducted in accordance with ASME codes and regulations in § 50.55a. While the proposed part 53 would not incorporate specific consensus codes and standards into the regulations, it would include numerous mentions of using generally accepted codes and standards that have been endorsed by or otherwise found acceptable by the NRC. The proposed requirement for an inservice inspection and inservice testing program would reinforce the need to develop monitoring programs to be conducted during a plant's operations phase to complement the design process and address inherent uncertainties. The NRC encourages the continued use of consensus codes and standards supporting design, testing, and inspections to support integrated and performance-based approaches in demonstrating compliance with the proposed requirements in part 53.

Proposed § 53.890 would establish requirements for facility safety programs (FSPs). FSPs would complement proposed requirements in subpart C that call for using and periodically updating PRAs and other requirements within proposed Framework A related to configuration control and maintaining the capabilities and reliabilities of SSCs and programmatic controls consistent with underlying analyses. The proposed use of the PRA as a major part of the design and licensing of commercial nuclear plants under Framework A would also allow for its continued use during operations for evaluating changes, managing risks, and improving the relationship between the NRC's licensing and reactor oversight programs. The proposed requirements to periodically update the PRA and to address the possible differences between the assumptions in the analyses and the performance history of SSCs would be a significant change from the relatively static analyses and prescriptive compliance verifications that are used in many of the requirements in parts 50 and 52.

The FSP concept is being proposed, in part, to address the ability to and advantages of periodically assessing possible risk reduction measures considering technology changes, economic costs, operating experience, and new or revised hazard information. Various other sections within proposed Framework A would address the need to manage the risk profile of each commercial nuclear plant in a way that demonstrates compliance with the regulations and ensures consistency with the analyses performed in accordance with proposed subpart C. The proposed requirements for an FSP would supplement licensees' actions to maintain the SSCs, personnel, and programmatic controls consistent with the plant design and environs as understood at the time of initial licensing.

The FSP would contribute to the management of risks posed by commercial nuclear plants by providing periodic assessments in areas such as potential updated information on external hazards and having licensees consider when cost effective risk reduction measures would be appropriate. The FSP proposal for Framework A was adapted from similar programs in NRC regulations such as § 70.62, "Safety program and integrated safety analysis," and regulations issued by other Federal agencies such as the U.S. Department of Energy, U.S. Department of Transportation, and EPA. Similar to these existing examples, the proposed FSP requirements for Framework A are intended to provide a flexible, performance-based approach to address possible changes in various factors contributing to the risks posed by commercial nuclear plants. When fully considered as part of an overall regulatory regime, the FSP could enable an optimization of NRC oversight programs and more focused operating experience and hazard assessment programs and could contribute to addressing uncertainties associated with both design features and site characteristics during initial licensing. While the FSP may require additional effort from licensees, it would also provide more flexibility in addressing changes to a facility's risk profile than the current process. For example, an FSP could increase flexibility during initial licensing or NRC review of generic issues by providing assurance that new information and, when appropriate, possible risk reduction measures are being routinely assessed throughout the operating life of each commercial nuclear plant. The NRC has posed a question in section VII of this notice that asks about how the FSP could contribute to a more integrated regulatory approach and whether a similar requirement should be included in Framework B.

Proposed § 53.890 would provide criteria for considering risk-reduction measures when performing the proposed periodic assessments. The proposal would provide an screening criteria for considering risk reduction, below which, no cost-benefit-type analysis would be required by the licensee. The actual decision on whether to implement a change will include an assessment of costs and other factors. Guidance would be prepared to define appropriate factors and would consider existing guidance used by the

NRC in setting the dollars per person-rem factor, guidance covering regulatory analysis, and guidance used by applicants and licensees when evaluating severe accident mitigation alternatives as required by 10 CFR part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions.” The goal for establishing criteria for considering risk-reduction measures would be that they would be low enough to initiate the process when appropriate but not so low as to initiate unnecessary analyses. The proposed use of person-rem values as part of the criteria would support these types of cost-benefit assessments and would also introduce the consideration of broader societal impacts than is provided for by only the calculation of doses to hypothetical individuals as is done in the analyses required by proposed subpart C.

The remaining portions of proposed § 53.890 would provide the requirements to develop, implement, and maintain the FSP by developing an FSP plan. The FSP plan would be used to document the details of how assessments are performed; the licensee’s overall safety philosophy and safety culture as discussed in the Commission’s Safety Culture Policy Statement; the required participants and training; and the periodic reviews of the effectiveness of the FSP. The NRC would review the FSP plan as part of licensing reviews for OLs or COLs. Updates and revisions to the FSP plan would be required to be submitted at least every 24 months and would not be subject to NRC review and approval unless a proposed change to the FSP plan could not be implemented without an exemption from the requirements of § 53.890.

Proposed § 53.910 would establish requirements for developing, implementing, and maintaining procedures (e.g., operations and emergency operating procedures) and guidelines (e.g., accident management guidelines). The programmatic requirements for many of the procedures listed in this proposed section would be similar to the requirements found in the administrative controls section of TS for plants previously licensed under parts 50 and 52. The proposed inclusion, where appropriate, of accident management guidelines in these requirements is intended to ensure that an integrated set of procedures and guidelines would be established by licensees to ensure command and control across the spectrum of possible event sequences. The proposed required procedures would also include those needed to complement the design requirements in proposed § 53.440(m) related to criticality alarms and the equivalent of the procedures required in § 50.54(hh) to address notifications of potential aircraft threats.

Subpart G – Decommissioning Requirements

The proposed subparts G and Q in Frameworks A and B, respectively, would provide the regulatory requirements for the decommissioning phase of the life cycle of commercial nuclear plants. The only variations between proposed subpart G in Framework A and proposed subpart Q in Framework B are the references to various sections throughout part 53 (i.e., inter- and intra-subpart references in proposed subpart Q are made to the analogous sections in Framework B). The requirements being proposed in subparts G and Q for the decommissioning of a commercial nuclear plant are adopted from the current regulations in § 50.75, “Reporting and recordkeeping for decommissioning planning,” and § 50.82, “Termination of license.” Although the requirements from those sections of part 50 have been copied into proposed subparts G and Q with relatively few changes, the requirements are reorganized to fit within the part 53 structure. The few changes made were primarily to make the proposed requirements more technology inclusive by adding alternatives within sections, whereas some requirements in part 50 were developed specifically for LWRs.

An example of a regulatory requirement that would be made more technology inclusive within proposed subpart G is § 53.1020, “Cost estimates for decommissioning.” Section 50.75 provides minimum amounts of decommissioning funds required to demonstrate reasonable assurance of funds for decommissioning LWRs. Such generic amounts have not been developed for all of the reactor technologies that may be licensed under part 53. Therefore, the Commission proposes in § 53.1020 that site-specific cost estimates for decommissioning must be developed considering costs in such areas as engineering, labor, and waste disposal. The derivation of the generic cost estimates for LWRs in § 50.75 is provided in NUREG/CR-5884, “Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station,” and NUREG/CR-6187, “Revised Analyses of Decommissioning for the Reference Boiling Water Reactor Power Station.” Similar to part 50, a provision for an annual adjustment of decommissioning cost estimates would be provided by proposed § 53.1030. The equivalent sections in Framework B are proposed §§ 53.4620 and 53.4630.

Proposed part 53 would not address the final disposition of potential transportable reactors that could involve delivery of a fueled manufactured reactor module and subsequent removal of that module to a center for refurbishment or waste disposal. If needed, the NRC will address the requirements for transportable reactors removed from commercial nuclear plant sites through existing regulations or possibly a future rulemaking.

The NRC is currently pursuing another rulemaking, “Regulatory Improvements for Production and Utilization Facilities Transitioning to Decommissioning,” which was published as a proposed rule for public comment on March 3, 2022 (87 FR 12254). As these rulemakings progress, the NRC will consider revisions to part 53 to align the two rulemaking efforts. For example, the proposed §§ 53.1075 and 53.4675 in Frameworks A and B respectively could be expanded to include or reference requirements for decommissioning in areas such as emergency planning and security in addition to the proposed decommissioning fire protection plans that would provide an equivalent to § 50.48(f).

Subpart H – Licenses, Certifications, and Approvals

Proposed subpart H would provide requirements related to applications under Framework A of this part for NRC licenses, certifications, or approvals for commercial nuclear plants.

Proposed subpart H would address general application requirements applicable to all Framework A applications as well as requirements specific to Framework A applications for LWAs, ESPs, standard design approvals, standard design certifications, MLs, CPs, OLs, and COLs. Proposed subpart H would be equivalent to and include all the existing licensing, certification, and approval processes currently covered under parts 50 and 52, with the exception of the process for early review of site suitability issues. Interactions with external stakeholders during the development of the proposed rule did not identify significant interest in or need for including the process for early review of site suitability issues in part 53. Consequently, much of the proposed subpart H regulatory text is identical to the corresponding language in parts 50 and 52, as described in the preamble for subpart H, with minor changes to account for cross references in Framework A, to make language technology neutral, or to reflect the unique analytical approach in Framework A. In these instances, this preamble discussion will describe the language as “equivalent” to the existing corresponding requirement in part 50 or part 52 and will describe any deviations, where applicable

Because Framework A carries over the majority of the licensing options from parts 50 and 52, there are several sections in proposed subpart H that are similar to existing regulations in parts 50 and 52. Some of this text relates to proposed requirements that are also being addressed in the proposed rulemaking on “Alignment of Licensing Processes and Lessons Learned from New Reactor Licensing” (Docket ID NRC-2009-0196) for parts 50 and 52, hereafter referred to as the “parts 50/52 rulemaking.” To minimize confusion and duplicative efforts between this rulemaking and the parts 50/52 rulemaking, the NRC will reconcile similar requirements between the parts 50/52 rulemaking and the part 53 rulemaking once the parts 50/52 rulemaking is issued as a final rule. Therefore, proposed subpart H largely reflects the current version of parts 50 and 52. However, as described in section VII of this *Federal Register* notice, the NRC staff is seeking public comment on the extent to which the contemplated changes to regulations in the parts 50/52 rulemaking effort should, or should not, be carried over to part 53.

Proposed § 53.1100 would address filing of applications for licenses, certifications, or approvals under oath or affirmation and is equivalent to § 50.30. The proposed § 53.1100 does not include the current requirement in § 50.30(a)(2) that the applicant maintain the capability to generate additional copies of the general information and the Safety Analysis Report, or part thereof or amendment thereto, for subsequent distribution, because it is unnecessary in the age of electronic submissions. In addition, the existing requirement on applications for OLs in § 50.30(d) is included in proposed

§ 53.1124(g)(2), “Relationship between sections,” covering OLs, rather than in proposed § 53.1100.

Proposed § 53.1101 would lay out activities requiring an NRC license and is equivalent to § 50.10(d). Proposed § 53.1103 would address combining applications and is equivalent to §§ 50.31 and 50.52 and 52.8. Proposed § 53.1103(b) would continue the Commission’s practice of combining multiple authorizations for a facility under parts 30, 40, 50, and 70 into one license based on Commission's authority under Section 161h. of the AEA to combine NRC licenses. Proposed § 53.1106 would address elimination of repetition and is equivalent to § 50.32.

Proposed § 53.1109 would provide general information requirements for the Framework A content of applications to the NRC and is equivalent to § 50.33, with the exception of § 50.33(f) on financial qualifications, which is covered in proposed subpart J, and § 50.33(h) on earliest and latest dates for completion of construction, which is covered in § 53.1306 of this subpart. Each application would need to include information to address the items in proposed § 53.1109 as cited in the appropriate section of this subpart for the application type. Proposed § 53.1109 (g) and (i) would be updated accordingly following the Commission’s decision regarding the final rule on “Emergency Preparedness for Small Modular Reactors and Other New Technologies” (Docket ID NRC-2015- 0225). One change from current requirements can be found in proposed § 53.1109(i), which is not limited to electricity generation, as it is currently in part 50. Some prospective NRC applicants are considering development of nuclear plants for other commercial ventures, such as process heat generation or hydrogen production.

Proposed § 53.1112 would address environmental conditions and is equivalent to § 50.36(b). Proposed § 53.1115 would address requirements for agreements limiting access to classified information and is equivalent to § 50.37. Proposed § 53.1118 would address ineligibility of certain applicants and is equivalent to § 50.38. Proposed § 53.1120 would address exceptions and exemptions from licensing requirements for Department of Defense and Department of Energy facilities and is equivalent to § 50.11. Proposed § 53.1121 would address public inspection of applications and is equivalent to § 50.39.

Proposed § 53.1124 would address the relationship between the various licenses, certifications, and approvals provided in this subpart, and the requirements are equivalent to a number of similar provisions in part 52 including §§ 50.10, 52.13, 52.43, 52.73, 52.133, and 52.153. New provisions are provided in § 53.1124(c) and (d) which would allow an application for either a standard design approval or a standard design certification under Framework A of this part to reference applicable licensing basis information that supported issuance of an OL or COL under Framework A that is essentially the same as the information supporting a standard design for which approval or certification is being requested. These provisions are being proposed to offer additional flexibility beyond what is currently allowed under parts 50 or 52 for a vendor who may wish to license a first-of-a-kind reactor for operation prior to seeking generic approval or certification of the reactor design.

Proposed § 53.1124(e) is equivalent to § 52.153, with additions for loading fresh fuel into the manufactured reactor module at the manufacturing facility. The addition of this option in part 53 is intended to allow for future commercial reactor deployment models, such as microreactors. As discussed further in the preamble for subpart E, the NRC has previously considered fuel load to be the point at which facility operation begins, which requires a COL or OL. Therefore, if a ML holder loads fuel prior to shipping, it must either obtain a COL or take sufficient measures to demonstrate that the module cannot achieve criticality and therefore is not a utilization facility while those measures remain in place. To accommodate this deployment model, part 53 allows COL applications to reference a ML, but not a CP. The NRC has not included provisions that would allow a CP applicant to reference a ML even though part 52 currently allows this. Because MLs require ITAAC and a significant amount of design finalization, the NRC staff does not anticipate an applicant for a CP to benefit from a referencing a ML. In addition, under a CP, this deployment model would have raised questions about a fueled manufactured reactor module being delivered to a site that did not have a license approving operation of the reactor.

Proposed § 53.1130 would address LWAs and is equivalent to § 50.10. However, in proposed part 53, the definition of construction from § 50.10 would be included in § 53.024, "Definitions specific to Framework A," rather than in this section on requesting LWAs. Also, the part 53 equivalent of § 50.10(d) is § 53.1130.

Proposed §§ 53.1140 through 53.1188 would govern the content of ESP applications. Proposed § 53.1140 is equivalent to § 52.12. Proposed § 53.1143 would address filing of applications and is equivalent to § 52.15. Proposed § 53.1144 would address general information requirements for the content of applications and is equivalent to § 52.16.

Proposed § 53.1146 would specify requirements for the technical contents of applications and is equivalent to § 52.17. Note that the proposed requirements in § 53.1146(b)(2) may be affected by issuance of the final rulemaking on "Emergency Preparedness for Small Modular Reactors and Other New Technologies" (Docket ID NRC-2015- 0225).

Proposed § 53.1149 would address standards for review of ESP applications and administrative review of applications, including hearings, and is equivalent to §§ 52.18 and 52.21. Proposed § 53.1155 would address referral to the ACRS and is equivalent to § 52.23. Proposed § 53.1158 would address issuance of ESPs and is equivalent to § 52.24. Proposed § 53.1161 would address the extent of activities permitted and is equivalent to § 52.25. Proposed § 53.1164 would address the duration of an ESP and is equivalent to § 52.26. Proposed § 53.1167 would address provisions for requesting a LWA after issuance of an ESP and is equivalent to § 52.27. Proposed § 53.1170 would address transfers of ESPs and is equivalent to § 52.28. Proposed § 53.1173 would address applications for ESP renewals and is equivalent to § 52.29. Proposed § 53.1176 would address criteria for renewal of an ESP and is equivalent to § 52.31. Proposed § 53.1179 would address the duration of an ESP renewal and is equivalent to § 52.33. Proposed § 53.1182 would address the use of a site for purposes other than those described in the permit and is equivalent to § 52.35. Proposed § 53.1188 would address finality of ESP determinations and is equivalent to § 52.39.

Proposed §§ 53.1200 through 53.1221 would govern the contents of standard design approval applications. Proposed § 53.1200 is equivalent to § 52.131. Proposed § 53.1203 would address filing of applications and is equivalent to § 52.135. Proposed § 53.1206 would address general information requirements for the content of applications and is equivalent to § 52.136.

Proposed § 53.1209 would address requirements for the technical content of applications and is largely equivalent to § 52.137. In proposed § 53.1209(a), the NRC proposes new text that uses § 52.137 as a starting point and expands the discussion of “major portion” standard design approvals. Additional discussion regarding standard design approvals for a major portion of a standard design can be found in the NRC's "A Regulatory Review Roadmap for Non-Light Water Reactors," (ADAMS Accession No. ML17312B567), which considers the Nuclear Innovation Alliance report "Clarifying 'Major Portions' of a Reactor Design in Support of a Standard Design Approval." Proposed § 53.1209(b) outlines the required content of the Final Safety Analysis Report (FSAR). Proposed requirements in § 53.1209(b)(2) for portions of the application addressing design information state that the application must include design information equivalent to that required for a standard design certification. This reference to the pertinent design certification requirements (specifically, those in proposed § 53.1239(a)(2) through (27)) is an efficiency that would save needing to repeat many of the same requirements for the content of a standard design approval application.

Proposed § 53.1210 would address requirements for the content of a standard design approval application other than the FSAR. Proposed § 53.1210(a) would require the inclusion of a description of availability controls that are not included in the FSAR.

Proposed § 53.1212 would address standards for review of applications and is equivalent to § 52.139. Proposed § 53.1215 would address referral to the ACRS and is equivalent to § 52.141. Proposed § 53.1218 would address staff approval of designs and duration of design approvals and is equivalent to §§ 52.143 and 52.147. Proposed § 53.1221 would address finality of standard design approvals and information requests and is equivalent to § 52.145.

There is no equivalent to proposed § 53.1221(d) in part 52 for standard design approvals. This provision would state that the Commission will require, before granting a CP, COL, OL, or ML which references a standard design approval that engineering documents, such as analyses, drawings, procurement specifications, or construction and installation specifications, be completed and available for audit if the more detailed information is necessary for the Commission to verify the information in the application and make its safety determination., including the determination that the application is consistent with the design approval information. A similar provision is included in part 52 in relation to a standard design certification; and the NRC would require that design and analysis information needed for the Commission to make its safety determination be complete and available for any application the NRC is reviewing. Making this explicit in the case of an application referencing a standard design approval – just as it is explicit for an application referencing a standard design certification – provides increased clarity to future standard design approval applicants.

Proposed §§ 53.1230 through 53.1263 would address standard design certifications. Proposed § 53.1230 would address general provisions for standard design certifications and is equivalent to § 52.41. Proposed § 53.1233 would address filing of applications and is equivalent to § 52.45. Proposed § 53.1236 would address general information requirements for the content of applications and is equivalent to § 52.46. Proposed § 53.1239 would address requirements for the technical content of applications and is equivalent to § 52.47(a). The requirements in proposed § 53.1239 have been modified from the analogous requirements in § 52.47(a) to align with the technical requirements in proposed Framework A of this part.

Proposed § 53.1241 would address requirements for the content of a standard design certification application other than the FSAR and is equivalent to § 52.47(b) and (d).

Proposed § 53.1242 would address review of applications and is equivalent to §§ 52.48 and 52.51. Proposed § 53.1242(c) would include a provision that would allow a design certification applicant to reference applicable licensing basis information for an OL or COL issued under Framework A. As explained previously, this provision is being proposed to offer additional flexibility beyond what is currently allowed under parts 50 or 52 for a vendor who may wish to license a first-of-a-kind reactor for operation prior to seeking certification of the generic reactor design. This proposal would provide finality provisions similar to those for a design certification applicant referencing a standard design approval.

Proposed § 53.1245 would address referral to the ACRS and is equivalent to § 52.53.

Proposed § 53.1248 would address issuance of standard design certification and is equivalent to § 52.54. Proposed § 53.1251 would address duration of certification and is equivalent to § 52.55. Proposed § 53.1254 would address application for renewal and is equivalent to § 52.57. Proposed § 53.1257 would address criteria for renewal and is equivalent to § 52.59. Proposed § 53.1260 would address duration of renewal and is equivalent to § 52.61. Proposed § 53.1263 would address finality of standard design certifications and is equivalent to § 52.63.

Proposed §§ 53.1270 through 53.1291 would address MLs covering manufacturing activities at one or more licensee facilities. Proposed § 53.1270 would address the scope of these sections and is equivalent to § 52.151.

Proposed § 53.1273 would address filing of applications. Proposed § 53.1273(a) is equivalent to § 52.155. Proposed § 53.1273(b) would address applicants that anticipate factory loading of fuel into a manufactured reactor module and would require that such applicants also possess, apply for, or reference licenses and certifications required by parts 70 and 71. As discussed in the preamble for subpart E, if a ML holder loads fuel prior to shipping, it must either treat the module as a utilization facility and obtain a COL or take sufficient measures to demonstrate that the module cannot achieve criticality and therefore is not a utilization facility while those measures remain in place.

Proposed § 53.1276 would address general information requirements for the content of ML applications and is equivalent to § 52.156, with one exception. Proposed § 53.1276 would require each application for a ML to also include the information required by § 53.1109(e). This information includes the type of license applied for, the use to which the facility will be put, the period of time for which the license is sought, and a list of other licenses, except operator's licenses, issued or applied for in connection with the proposed facility. The reason for this addition is to address applicants that anticipate factory loading of fuel into a manufactured reactor module that would possess or be applying for licenses and certifications required by parts 70 and 71. It is possible that such applicants would be applying for both the ML and other licenses simultaneously or in a single application. The NRC could combine both the ML and other licenses, as is currently the practice for the granting of COLs under part 52 that include license authority under parts 30, 40, and 70.

Proposed § 53.1279 would address requirements for the technical content of applications for MLs to be included in the FSAR, and is equivalent to § 52.157, except modified with proposed additions to address applicants that anticipate factory loading of fuel into a manufactured reactor module. In addition, the requirements in proposed § 53.1279(a) and (b) have been modified from the analogous requirements in § 52.157 to align with the technical requirements in proposed Framework A of this part. Proposed § 53.1279(a)(2) outlines the required content of the application addressing design information and states that the application must include design information equivalent to that required for a standard design certification. This reference to the pertinent design certification requirements is an efficiency that would save needing to repeat the same requirements for the content of a ML application.

Proposed § 53.1279(c) would provide application requirements related to the deployment of the completed manufactured reactor or manufactured reactor module. Proposed § 53.1279(c)(1) would require inclusion of information related to the procedures governing the preparation of the manufactured reactor or manufactured reactor module for shipping to the site where it is to be operated, the conduct of shipping, and the verification of the condition of the shipped items upon receipt at the site. Proposed § 53.1279(c)(2) would require that the application include information on the interaction of the design, manufacture, and installation of a manufactured reactor or manufactured reactor module within the applicant’s organization and the manner by which the applicant will ensure close integration between the designer, contractors, and any facility in which the manufactured reactor or manufactured reactor module is to be installed. Finally, proposed § 53.1279(c)(3) would require that the application include a description of the measures used for the control of interfaces between the holder of the ML and the holder of the COL for the commercial nuclear plant at which the manufactured reactor or manufactured reactor module is to be installed. This information is necessary for the NRC to determine that the applicant would have appropriate controls in place to ensure coordination between parties involved in the design, manufacture, and eventual operation of any reactor or reactor module manufactured under a ML.

Proposed § 53.1279(d) would focus on application requirements related to an application for a ML that includes the installation of fuel at the factory. These requirements would address the fueling operations and the required protections to prevent criticality and otherwise ensure the safety of workers and the public during the manufacture, storage, and transport of each manufactured reactor module. Proposed § 53.1279(d)(1) would require a description of the safety program and integrated safety analysis required by subpart H of part 70. This information would be in the form of a reference to the applicable part 70 application and license, if already issued, or provided in the Safety Analysis Report supporting the ML if the same application is used for the ML and part 70 license. Proposed § 53.1279(d)(1)(i) would require that the application address the measures taken for fuel loading, in-factory inspections, and non-nuclear testing, including protections to be taken to prevent criticality. It is important to emphasize that only non-nuclear testing is allowed under a ML in part 53. In accordance with the AEA, additional part 53 licenses would be required to perform any testing on manufacture reactor modules that cannot be performed with sufficient measures to demonstrate that the module cannot achieve criticality and therefore is not a utilization facility while those measures remain in place. Proposed § 53.1279(d)(1)(ii) would require that an application explain the functional design criteria and design features included in the manufactured reactor module, or explain the physical or programmatic controls implemented during manufacturing, storage, or transport to prevent criticality. This information is necessary for the NRC to verify that controls would be in place to prevent criticality under all circumstances covered by the ML and transport of the manufactured reactor module to the site where it is to be installed and operated.

Proposed § 53.1279(d)(2) would require a description of the procedures governing the transfer of authorities and responsibilities for the manufactured reactor module from the holder of the ML to the holder of the COL for the installation site.

Proposed § 53.1279(d)(3) would require a description of the controls needed to demonstrate compliance with the requirements of proposed § 53.620 to address the receipt, storage, and loading of SNM into a manufactured reactor module, including a FFD program, a radiation protection program, an information security program, a physical security program, a fire protection program, an emergency plan, and a plant staff training program.

Proposed § 53.1282 would provide requirements for other application content for MLs and is equivalent to § 52.158 with the exception of added requirements to address applicants that anticipate factory loading of fuel into a manufactured reactor module. Proposed § 53.1282(a) would provide requirements to include in the ML application the ITAAC that the licensee who will hold the COL authorizing operation of the manufactured reactor or manufactured reactor module must perform and the relationship between ITAAC when a standard design certification is referenced in the ML application. Proposed § 53.1282(a)(2) would address the referenced standard design certification ITAAC that would be satisfied at the manufacturing facility. Proposed § 53.1282(a)(3) would state that the COL application may include a notification that required referenced standard design certification ITAAC have been satisfied at the manufacturing facility. Proposed § 53.1282(b) would require a ML application to include an environmental report and, consistent with existing requirements, proposed § 53.1282(b)(2) would note that if the ML application references a standard design certification, the environmental report need not contain a discussion of severe accident mitigation design alternatives for the manufactured reactor or manufactured reactor module as used in a commercial nuclear plant. However, this proposed paragraph would include an additional requirement for an application for a ML that references a standard design certification but includes the installation of fuel at the factory. Such an application would be required to discuss severe accident mitigation design alternatives for the reactor module while at the factory and severe accident mitigation alternatives for the factory itself. The evaluation and discussion of severe accident design alternatives for loading fuel in a factory should reflect the characteristics of the manufactured reactor module and fuel at the time of assembly in the manufacturing facility.

Proposed § 53.1285 would provide standards for review of applications and administrative review of applications for MLs, including hearings, and is equivalent to §§ 52.159 and 52.163.

Proposed § 53.1286 would address referral of applications to the ACRS and is equivalent to § 52.165. Proposed § 53.1287 would address issuance of a ML and is equivalent to § 52.167.

Proposed § 53.1288 would address finality of MLs and is equivalent to § 52.171. Proposed § 53.1291 would address the duration of MLs and is equivalent to § 52.173. Proposed § 53.1293 would address the transfer of MLs and is equivalent to § 52.175. Proposed § 53.1295 would address the renewal of MLs and is equivalent to §§ 52.177, 52.179 and 52.181, with a minor exception. Proposed § 53.1295(a)(3) would state that a ML for which a timely application for renewal has been filed remains in effect until the Commission has made a final determination on the renewal application, provided, however, that the holder of a ML may not begin manufacture of a manufactured reactor or manufactured reactor module less than six months before the expiration of the license. The proposed 6-month time frame for this provision is changed from the 3-year period in the equivalent provision in part 52 because future reactor applicants may present smaller, simpler designs, to include microreactor designs, in ML applications than those that were envisioned when the existing requirements were written. A 6-month time frame for this provision would provide greater flexibility for ML holders without raising significant regulatory concerns related to manufactured reactors being produced when the ML expires.

Proposed §§ 53.1300 through 53.1348 would address licensing requirements for CPs. Proposed § 53.1300 would set out general requirements for CPs and is equivalent to § 50.23. Proposed § 53.1306 would address the general information requirements for the content of applications for CPs and is equivalent to § 50.33, with the exception of § 50.33(h) on earliest and latest dates for completion of construction, the equivalent of which is found in proposed § 53.1309(b).

Proposed § 53.1309 would address requirements for the technical content of applications for CPs and is equivalent to §§ 50.34(a) and (e) and 100.21(f). This section would reference the requirements for the content of an ESP application to address application requirements related to siting and would reference the requirements for the content of a design certification application to address application requirements related to design of the commercial nuclear plant. Proposed § 53.1309(a)(2)(ii) would address the treatment of preliminary design information and notes that information provided in the application may include some aspects of the design that are not fully developed. This provision would require that the completed design, including any changes during construction, be described in the FSAR in an application for an OL. This would include the requirement for a description of the PRA required by § 53.450(a) and its results. PRAs developed for commercial nuclear plants prior to construction would be based on the design and other information available at the time of the CP application. PRAs performed in early design stages or prior to construction may be inherently less detailed and may include projected information that will be subsequently verified or revised when the plant is built. The FSAR will describe the final, updated design information and analysis results.

Proposed § 53.1312 would address other application content for CPs. Proposed § 53.1312(a)(1)(i) is equivalent to § 50.30(f). Proposed § 53.1312(a)(1)(ii) is equivalent to § 50.10(d)(2). Proposed § 53.1312(b)(1) is equivalent to § 52.79(b), (c), and (d) but is adapted for a CP application. Section 53.1312(b)(2) is equivalent to portions of §§ 52.63(b)(1), 52.79(b)(2), 52.80, and 52.93, but is adapted for a CP application. Guidance for equivalent requirements in parts 50 and 52 is also addressed in RG 1.206, Rev 1, section C.1.7.

Proposed § 53.1315 would address standards for review of applications and administrative review of applications, including hearings, and is equivalent to § 50.58, and to §§ 52.81 and 52.85, but is adapted for a CP application.

Proposed § 53.1318 would address finality of NRC approvals, licenses, and certifications referenced in a CP application and is equivalent to § 52.83 but is adapted for a CP application.

Proposed § 53.1324 would address referral to the ACRS and is equivalent to § 50.58 and to § 52.87 but is adapted for a CP application.

Proposed § 53.1327 would address authorization to conduct LWA activities and is equivalent to § 50.10(g). Proposed § 53.1330 would address exemptions, departures, and variances for CP applicants. Proposed § 53.1327(a) is equivalent to § 50.11 and to § 52.93 but is adapted for a CP application. Proposed § 53.1327(b) is equivalent to §§ 52.39(d) and 52.93 but is adapted for a CP application.

Proposed § 53.1333 would address issuance of CPs. Proposed § 53.1333(a) is equivalent to § 50.35(a). Proposed § 53.1333(b) is equivalent to §§ 50.35(b) and 50.36(b), and to § 52.97(c) but is adapted for a CP application. Proposed § 53.1336 would address finality of CPs and is equivalent to § 50.35(b) and (c). Proposed § 53.1342 would address the duration of CPs. Proposed § 53.1342(a) is equivalent to § 50.55(a). Proposed § 53.1342(b) is equivalent to § 50.55(b). Proposed § 53.1345 would address the transfer of CPs and is equivalent to § 50.80. Proposed § 53.1348 would address the termination of CPs and is equivalent to §§ 52.3(b)(8) and 52.110(a)(1) but is adapted for a CP application.

Proposed §§ 53.1360 through 53.1405 address requirements for OLs.

Proposed § 53.1366 would address requirements for the general content of applications for OLs. It would refer to general content requirements in proposed § 53.1109 and would require supplemental information. Proposed § 53.1366 (a) is equivalent to § 50.33(f). Proposed § 53.1366 (b) is equivalent to § 50.75.

Proposed § 53.1369 would provide requirements for the technical content of applications for OLs to be included in the FSAR and is equivalent to § 50.34(b), but has been modified to align with the technical requirements in proposed Framework A of this part. It would require that the FSAR include and, as needed, update information provided in the Preliminary Safety Analysis Report (PSAR) that was submitted and reviewed to support the associated CP application.

Similar to the proposed requirements for the content of CP applications, proposed § 53.1369(a) would reference the requirements for the content of an ESP application to address application requirements related to siting. Section 53.1369(b) would reference the requirements for the content of a design certification application to address some of the application requirements related to design of the commercial nuclear plant.

Proposed § 53.1369(c) is equivalent to § 50.34(b)(7). Proposed § 53.1369(d) would require a description of the Integrity Assessment Program that would be required by proposed § 53.870. Proposed § 53.1369(e) is equivalent to § 50.34(e). Proposed § 53.1369(g) would provide requirements for OL application content to support proposed § 53.730 related to the role of personnel in the operation of the commercial nuclear plant and is adapted from requirements in part 55. Likewise, proposed § 53.1369(h) would provide requirements for OL application content related to training programs to support proposed §§ 53.730(g) and 53.830, and includes requirements equivalent to § 50.34(b)(8) and requirements in part 55. Proposed § 53.1369(i) would provide requirements for OL application content related to emergency plans to support proposed § 53.855 and is equivalent to § 50.34(b)(6)(v). The NRC plans to update these requirements following publication of the final rulemaking on “Emergency Preparedness Requirements for Small Modular Reactors and Other New Technologies” (Docket ID NRC-2015-0225) to be consistent with the final requirements in that rulemaking.

Proposed § 53.1369(j) would provide requirements for OL application content related to the applicant's organizational structure and is equivalent to § 50.34(b)(6)(i). Proposed § 53.1369(k) would provide requirements for OL application content related to the applicant’s proposed maintenance program to support proposed § 53.715 and is equivalent to § 50.34(b)(6)(iv). Proposed § 53.1369(l) would provide requirements for OL application content related to the applicant's quality assurance program to support proposed § 53.865 and is equivalent to § 50.34(b)(6)(ii). Proposed § 53.1369(m) would provide requirements for OL application content related to the applicant’s proposed radiation protection program to support proposed § 53.850 and is equivalent to § 52.79(a)(39).

Proposed § 53.1369(n) through (p) would provide requirements for OL application content related to the applicant's proposed physical security program to support proposed § 53.860(a) and are equivalent to § 50.34(c) and (d). Proposed § 53.1369(q) would provide requirements for OL application content related to the applicant's proposed cybersecurity plan to support proposed § 53.860(d) and is equivalent to §§ 52.79(a)(36)(iv) and 73.54. Proposed § 53.1369(r) would provide requirements for OL application content related to the implementation of proposed security, safeguards and cybersecurity plans to support proposed § 53.860 and is equivalent to § 52.79(a)(35)(ii) and 52.79(a)(36)(iv).

Proposed § 53.1369(s) would provide requirements for OL application content related to the applicant's proposed fire protection program to support proposed § 53.875 and is equivalent to § 52.79(a)(40). Proposed § 53.1369(t) would provide requirements for OL application content related to the applicant's proposed inservice inspection and inservice testing program to support proposed § 53.880 and is equivalent to § 52.79(a)(11). Proposed § 53.1369(v) would provide requirements for OL application content related to the applicant's FSP to support proposed § 53.890. Proposed § 53.1369(w) would provide requirements for OL application content related to the applicant's general employee training program to support proposed § 53.830 and is equivalent to § 52.79(a)(33). Proposed § 53.1369(x) would provide requirements for OL application content related to the applicant's FFD program to support part 26 and is equivalent to § 52.79(a)(44). Proposed § 53.1369(y) would provide requirements for OL applicant's programs to demonstrate that any safety questions identified at the CP stage have been resolved and is equivalent to § 50.34(b)(5). Proposed § 53.1369(z) would provide requirements for OL applicants to describe how the performance of each safety design feature has been demonstrated capable of fulfilling functional design criteria considering interdependent effects through either analysis, appropriate test programs, prototype testing, operating experience, or a combination thereof to support proposed § 53.440(a) I is largely equivalent to §§ 50.34(b)(5) and 50.43(e). Proposed § 53.1369(aa) would provide requirements for OL application content related to the applicant's proposed TS to support proposed § 53.710(a) and is equivalent to § 50.34(b)(6)(vi).

Proposed § 53.1372 would address requirements for the content of OL applications other than the FSAR. Proposed § 53.1372(a) would require submission of an environmental report and is equivalent to § 51.53(b). Proposed § 53.1372(b) is new and would require the inclusion of a description of availability controls that are not included in the FSAR to support proposed § 53.710(b).

Proposed § 53.1375 would address standards for review of OL applications and the administrative review of applications, including hearings, and is equivalent to §§ 52.81 and 52.85, with one exception. The NRC has removed part 54 from the list of standards in the proposed § 53.1375(a). Proposed part 53 does not include requirements related to renewal of an OL, although a placeholder for possible future requirements has been included as proposed § 53.1595. The NRC will decide after the part 53 final rule is published whether this future section will be retained in part 53 to address license renewal or whether the agency will take another approach to address license renewal for part 53 licensees, such as amending part 54 to address part 53 licensees.

Proposed § 53.1381 would address referral to the ACRS and is equivalent to §§ 50.58 and 52.87. Proposed § 53.1384 would address exemptions, departures, and variances for OL applicants. Section 53.1381(a) is equivalent to § 52.93 but is adapted for OLs. Proposed § 53.1381 (b) is equivalent to §§ 52.39(d) and 52.93 but is adapted for OLs.

Proposed § 53.1387 would address issuance of OLs. Proposed § 53.1387(a)(1)(i) is equivalent to §§ 50.50 and 50.57(a)(1). Proposed § 53.1387(a)(1)(ii) is equivalent to § 50.50. Proposed § 53.1387(a)(1)(iii) is equivalent to § 50.57(a)(2). Section 53.1387 (a)(1)(iv) is equivalent to § 50.57(a)(3). Proposed § 53.1387(a)(1)(v) is equivalent to § 50.57(a)(4). Proposed § 53.1387(a)(1)(vi) is equivalent to § 50.57(a)(6). Proposed § 53.1387(a)(1)(vii) is equivalent to § 50.57(a)(5). Proposed § 53.1387(a)(1)(viii) is equivalent to § 52.97(a)(1)(vi). Proposed § 53.1387(b) is equivalent to § 52.103(g). Proposed § 53.1387(c) is equivalent to § 50.57(b). Proposed § 53.1387(d) is equivalent to §§ 50.36(b) and 50.50.

Proposed § 53.1390 would address finality of OLs and is equivalent to § 50.109. Proposed § 53.1396 would address duration of an OL and is equivalent to § 50.51. Proposed § 53.1399 would address transfer of an OL and is equivalent to § 50.80. Proposed § 53.1402 would address applications for renewal of an OL and refers to proposed § 53.1595. Proposed part 53 does not include requirements related to renewal of a COL, although a placeholder for possible future requirements has been included as proposed § 53.1595. Proposed § 53.1405 would address continuation of an OL and is equivalent to § 52.109 but is adapted to address an OL.

Proposed §§ 53.1410 through 53.1461 would address requirements for COLs. Proposed § 53.1410 is equivalent to § 52.71. Proposed § 53.1413 would address general information requirements for the content of applications for COLs and is equivalent to § 52.77, which references § 50.33. Most of the provisions from § 50.33 are restated in proposed § 53.1109. Requirements in § 50.33 related to financial qualifications and construction timelines are addressed in other sections of Framework A.

Proposed § 53.1416 would address the technical content to be included in applications for COLs in FSAR and is equivalent to § 52.79 except as modified to reflect the technical requirements in Framework A of this part and with one addition. Proposed § 53.1416 includes the statement that the Commission will require, before issuance of a COL, that engineering documents, such as analyses, drawings, procurement specifications, or construction and installation specifications, be completed and available for audit if the more detailed information is necessary for the Commission to verify the information in the application and make its safety determination. This statement is equivalent to design certification application requirements in § 52.47 and is included in proposed § 53.1416 for clarity.

Similar to the proposed requirements for the content of OL applications, proposed § 53.1416(a)(1) would reference the requirements for the content of an ESP application to address application requirements related to siting. Section 53.1369(b) would reference the requirements for the content of a design certification application to address some of the application requirements related to design of the commercial nuclear plant.

Proposed § 53.1419 would address other application content for COLs and is equivalent to § 52.80. Proposed § 53.1419(b) is new and would require the inclusion of a description of availability controls that are not required to be included in the FSAR.

Proposed § 53.1422 would address standards for review of applications and the administrative review of applications, including hearings, and is equivalent to §§ 52.81 and 52.85. The NRC has removed part 54 from the list of standards in proposed § 53.1422(a). Proposed part 53 does not include requirements related to renewal of an OL, in relation to proposed §§ 53.1375 and 53.1595.

Proposed § 53.1425 would address the finality of NRC approvals referenced in a COL application and is equivalent to § 52.83. Proposed § 53.1431 would address the referral of COL applications to the ACRS for review and is equivalent to § 52.87. Proposed § 53.1434 would address the authorization to conduct LWA activities and is equivalent to § 52.91. Proposed § 53.1437 would address exemptions, departures, and variances and is equivalent to § 52.93. Proposed § 53.1440 would address issuance of COLs and is equivalent to § 52.97. Proposed § 53.1443 would address finality of COLs and is equivalent to § 52.98.

Proposed § 53.1449 would address inspection during construction and is equivalent to § 52.99. Proposed § 53.1452 would address operation under a COL and is equivalent to § 52.103. Proposed § 53.1455 would address duration of COL and is equivalent to § 52.104. Proposed § 53.1456 would address the transfer of a COL and is equivalent to § 52.105. Proposed § 53.1458 would address application for renewal and is equivalent to § 52.107. Proposed § 53.1461 would address continuation of COL and is equivalent to § 52.109.

Proposed § 53.1470 would address standardization of nuclear power plant designs and licenses to construct and operate commercial power reactors of identical design at multiple sites and is equivalent to appendix N of parts 50 and 52. The Commission's regulations in 10 CFR part 2, “Agency Rules of Practice and Procedure,” specifically provide for the holding of hearings on particular issues separately from other issues involved in hearings in licensing proceedings, and for the consolidation of adjudicatory proceedings and of the presentations of parties in adjudicatory proceedings such as licensing proceedings (§§ 2.316 and 2.317). This section would set out the particular requirements and provisions applicable to situations in which applications for CPs and subsequent OLs, or COLs, under this part are filed by one or more applicants for licenses to construct and operate nuclear power reactors of identical design ("common design") to be located at multiple sites.

Subpart I – Maintaining and Revising Licensing Basis Information

Part 53 would establish requirements for the maintenance of licensing basis information in subpart I for Framework A and in subpart S for Framework B. The two subparts would be in most respects similar, except as described in the following paragraphs, and included separately within the frameworks to support clarity and ease of use due to the differences in the internal references between Framework A and Framework B.

Sections 53.1500 and 53.6000 would describe the purpose of the subparts in terms of the common definition of licensing basis information in subpart A. Subparts I and S would be closely tied to the requirements in subparts H and R, which would provide the requirements for contents of applications for the various types of licenses issued under Framework A or Framework B of this part. Subparts I and S would generally be organized into sections dealing with: (1) licensing basis information that licensees are not authorized to change without NRC approval (e.g., licenses, regulations), and (2) licensing basis documents that licensees may change provided specified criteria are satisfied (e.g., FSAR, program descriptions). The subparts would also capture certain general conditions on licenses and changes to the licenses related to the transfer and termination of licenses.

Sections 53.1502 and 53.6002 within Frameworks A and B, respectively, would define specific terms and conditions of licenses. These terms and conditions would be equivalent to the regulations in: (1) § 50.54(h) stating that each license is subject to the provisions of the Act and requirements issued by the Commission; (2) § 50.54(aa) stating that each license is subject to the specified sections of the Federal Water Pollution Control Act; and (3) § 50.54(dd) stating that a holder of an OL or COL may take reasonable actions that depart from the license in a national security emergency.

Sections 53.1505(a) and 53.6005(b) in Frameworks A and B, respectively, would serve as an introduction to and overview of the sections that follow on changes to licensing basis information requiring prior NRC approval, namely the elements of licensing basis information defined by licenses, orders, and regulations. The related sections within these subparts would primarily deal with the process of how a licensee requests and the NRC issues an amendment to a license or issues an order that modifies a license. Another important element of licensing basis information that a part 53 licensee would not be able to change or deviate from without NRC approval would be the NRC regulations themselves. Sections 53.1505(b) and 53.6005(b) would refer to the common § 53.080 in subpart A that would provide the criteria for a licensee or other party to satisfy when requesting an exemption from NRC regulations.

Sections 53.1510 and 53.6010 would be equivalent to § 50.90 and would require that a licensee must submit an application to request an amendment to a license. The required assessments that would be included within an application to amend a license under Framework A would need to address the safety criteria and analysis requirements of subparts B and C. As with parts 50 and 52, licensees in both frameworks would be required to include in their applications to amend a license an analysis of whether the amendment involves no significant hazards consideration using the standards in §§ 53.1520 and 53.6020, which would be equivalent to the standards in § 50.92.

Sections 53.1515 and 53.6015 would establish requirements for public notices and state consultations associated with the NRC's processing of a license amendment request. These sections would be equivalent to § 50.91 for the NRC's processes related to applications to amend an OL or COL issued under part 53. Likewise, §§ 53.1520 and 53.6020 in Frameworks A and B respectively would be equivalent to § 50.92 on the issuance of an amendment. Section 50.91(b) includes a statement that "(The Commission will make available to the licensee the name of the appropriate State official designated to receive such amendments.)" While the Commission fully intends to continue following this practice, this statement in § 50.91(b) does not amount to a legal requirement and therefore the Commission has not included it in proposed part 53. The Commission would revise §§ 53.1515(b)(3) and 53.6015(b)(3) compared to § 50.91(b)(3) for clarity; these revisions are not intended to revise the substance of the provisions in part 53 compared to part 50.

Sections 53.1520 and 53.6020 would be based on § 50.92. Both sections would continue to use the criteria in § 50.92 for determining that a proposed amendment involves no significant hazards consideration. Although more specific terms such as event sequence are used throughout Framework A, § 53.1520 would use the term "accident" to maintain consistency between the two frameworks in part 53 and with the long history of making no significant hazards consideration determinations under part 50.

Sections 53.1525 and 53.6025 would provide requirements for holders of an OL or COL requesting to revise information from a design certification rule that was referenced in the initial license application and included in or incorporated by reference into the facility Final Safety Analysis Report (FSAR). In keeping with the current requirements in part 52, the portion of the part 53 facility licensing basis information obtained from the certified design would be divided into two categories. The most significant design information, including ITAAC, would be certified by rule and designated as “certification information.” The remaining information, which makes up the majority of the design information approved as part of the design certification, would not be certified by rule and is not considered “certification information.” Part 52 refers to these categories of information as Tier 1 and Tier 2 information, respectively, and refers to a change made to that information on a plant-specific basis as a departure. Under part 52, a departure from Tier 1 information requires an exemption and, for information incorporated into the license, a license amendment.

Both Framework A and Framework B would dispense with the Tier 1 and Tier 2 terminology. Rather, §§ 53.1525 and 53.6025 would use the term “certification information” in place of Tier 1, and a plant-specific departure from the certification information would require both a request for an exemption from the associated design certification rule and, for information such as ITAAC incorporated into the license, a license amendment. However, as would be provided in §§ 53.1525(c) and 53.6025(c), a plant-specific departure from the information approved by the NRC as part of the design certification rule but which is not certification information would be assessed using the process and criteria defined in §§ 53.1550 and 53.6050 for changes to a FSAR. An applicant or licensee would need to identify such a change as a departure from the referenced standard design in the updated FSAR. The process for making a generic change to a certified design would be described in the associated sections in subparts H and R of Framework A and Framework B, respectively.

Sections 53.1535 and 53.6035 would establish requirements for license amendments during construction. The sections would provide the equivalent options and requirements for the holders of a CP as those in § 50.35(b). The regulations would allow but do not require the holder of a CP or LWA to request an amendment under §§ 53.1510 and 53.6010 if the licensee desires to obtain NRC approval of a specific design feature or specification. The requirements for obtaining an amendment to a COL to address changes during construction would also be provided in §§ 53.1535 and 53.6035. The proposed process would differ from the current requirements in part 52 by adopting a requirement similar to that included in SECY-22-0052, "Proposed Rule: Alignment of Licensing Processes and Lessons Learned from New Reactor Licensing (Regulation Identifier Number [RIN] 3150 AI66)." The proposed regulation would allow the holder of a COL to proceed at its own risk in making a change during the construction process and would require that licensee to submit a license amendment request no later than 45 days from the date the licensee begins to implement the change or departure requiring NRC approval.

Sections 53.1540 and 53.6040 in Frameworks A and B, respectively, would serve as an introduction to the sections that follow on changes to licensing basis information that are primarily under the control of a licensee but for which evaluations are made to determine if a submittal to the NRC requesting approval would be required. The sections would also include definitions that would be applicable when using the processes in §§ 53.1545 through 53.1565 and §§ 53.6045 through 53.6065 of Frameworks A and B, respectively. The definitions would be largely equivalent to those in § 50.59(a) but include some revision to reflect the structure and terminology in other subparts in part 53. For example, the definition of "change" in § 53.1540(b)(1) would address a "safety function" rather than a "design function," because the former is a defined term in Framework A. In contrast, the corresponding text in § 53.6040(b)(1) would use the definition in § 50.59(a)(1) without modification. Similarly, in § 53.1540(b)(2), the phrase "design basis" from § 50.59(a)(2) would be replaced with "functional design criteria for SR SSCs."

Sections 53.1545 and 53.6045 would provide the proposed requirements for updating of FSARs. The proposed requirements in Framework B under § 53.6045 would be equivalent to the current requirements in § 50.71. While the process-related requirements proposed for Framework A under § 53.1545 would be largely the same as those in Framework B and § 50.71, the specifics of information to be updated would differ due to the role of PRA in satisfying the requirements in subparts B and C. The use of the risk-informed approach in subpart C would result in some but not all PRA information being in the FSAR or another licensing basis document and therefore a separate PRA update requirement similar to § 50.71(h) may not be needed in the proposed subpart I.

Proposed § 53.1239(a)(18) in subpart H and the related references to this proposed requirement for the holders of OLs and COLs would require a description of the PRA required by § 53.450(a), and its results to be included in FSARs. However, guidance documents are planned to clarify the division of PRA-related information that would need to be in the FSAR, in other possible licensing basis documents, and controlled as plant records subject to inspections and audits. At a minimum, the information from the PRA that would be needed to show compliance with subpart C would be included in the FSAR (e.g., PRA summary and analytical results for LBEs). The submittal of voluminous PRA information was initially required under part 52, but that proved to be impractical and was revised in the 2007 revision of part 52. Guidance is being developed to ensure sufficient information is submitted to the NRC to support the licensing process and the NRC's regulatory findings while avoiding submittal of detailed information that is better maintained and controlled by the applicant..

The NRC has posed a question in section VII of this notice that asks about the appropriate level of detail for PRA-related information in an FSAR and whether other licensing basis documents might be more appropriate to both provide information to the NRC and ensure the PRA is maintained and updated as proposed in subpart C. For example, a possible approach for Framework A could be to include a summary of the PRA results in the FSAR and control that information under this section but to create a separate document related to the broader PRA analyses and related processes as a program document under § 53.1545. The program document would provide more detail than the summaries in the FSAR but still be a much-condensed source of information in comparison to the documentation of the PRA. This alternate approach would similarly consider the role of the PRA in the licensing process and in maintaining margins to the safety and evaluation criteria in subparts B and C but may allow a more appropriate evaluation process to address the particulars and complexities of the PRA-related documents.

Section 53.1545(a)(3) and (4) would be based on the inclusion of at least a summary of PRA results and the related margins to safety criteria in the FSAR and would require updates to that information. The routine reporting of these margins would also inform application of the criteria for allowing changes without an amendment in the following section (§ 53.1550) in subpart I.

Sections 53.1550 and 53.6050 would establish requirements for evaluating changes to a facility as described in its FSAR. These proposed sections would provide the equivalent of the requirements in § 50.59 for evaluating changes to an Updated Final Safety Analysis Report (UFSAR) and determining if a license amendment is required to implement a change to a facility or procedures. The differences between the frameworks related to how safety analyses are performed and used to derive or confirm design requirements and related programmatic controls would result in differences in the evaluation criteria to determine if an amendment is required for a change to a facility or plant procedures. Because Framework B would be similar to parts 50 and 52 in its structure and terminology, § 53.6050 would propose to use the same evaluation criteria as provided in § 50.59. The evaluation criteria proposed in § 53.1550 would reflect the role of the PRA in the safety analyses under Framework A and would include several measures related to the changes in plant risk resulting from a change in the plant design or plant procedures. Examples would include criteria that rely on the identification of risk-significant event sequences in accordance with the analysis requirements of § 53.450; exceeding the LBE evaluation criteria as defined in § 53.450; the consideration of potential reductions in margin between the estimated plant risks and the cumulative risk measures in the safety criteria in § 53.220; changes to the safety classification of SSCs, and consideration of reductions in defense in depth.

Both §§ 53.1550 and 53.6050 would include a criterion related to a departure in a method of evaluation used in the safety analyses. Specifically, § 53.6050 would use the same wording as all eight criteria in § 50.59, including criterion (viii) on departing from methods of evaluation, and therefore licensees could consider technically relevant information in existing guidance documents to support such evaluations for § 53.6050; however, at this time, the NRC is not issuing or endorsing guidance for use in applying §§ 53.1550 and 53.6050. Moreover, much of the existing guidance would not be relevant for consideration in § 53.1550 of Framework A.

Section 53.1550 in Framework A would include certain concepts taken from existing guidance for § 50.59 in the proposed criteria related to DBAs. Specifically, criterion (iv) would be equivalent to a change in a method of evaluation under § 50.59 for changes made to a method of evaluation of DBAs under § 53.450(f), and criterion (viii) would be similar to the § 50.59 criterion (v) on assessing if a change creates a possibility for an accident of a different type than previously analyzed in the FSAR. Guidance documents will be prepared to address the content of applications for PRA-related information under proposed Framework A, and this guidance will also influence how potential changes in the evaluation of LI other than DBAs analyzed under § 53.450(e) are evaluated and reported under the proposed criterion (iv).

Criterion (xi) under the proposed § 53.1550 would require evaluating plant changes to ensure they would not prevent satisfying the design requirements in § 53.440(j) related to the impact of a large commercial aircraft would be included within the proposed § 53.1550 in Framework A. The inclusion of a proposed requirement under § 53.1550 related to design features for protecting against aircraft impact would reflect the proposed design requirement in subpart C and related proposed requirements in subpart H to address the proposed design requirement in FSARs submitted under Framework A. Framework B would include an assessment that would be equivalent to § 50.150 under proposed § 53.6054.

Section 53.6052 would provide requirements for maintaining the risk evaluation required by § 53.4730(a)(34). This requirement would be equivalent to existing requirements under § 50.71(h) with changes that reflect the different ways in which risk is assessed in Framework B (i.e., through a PRA or an AERI). Guidance for maintenance of risk evaluations that are based on a PRA is provided in RG 1.200 (for LWRs) and RG 1.247 (for NLWRs), which endorse industry consensus PRA standards. Additional detail on addressing maintenance of a risk evaluation that is based on the AERI approach will be provided in the guidance that is being developed.

Proposed § 53.6054 would help ensure that pertinent elements of § 50.150 would be appropriately captured in Framework B. While the technical requirements in § 50.150(a) and (b) would be included subpart R, the change control provisions in § 50.150(c), with minor differences for cross-references, would be included in subpart S consistent with the purpose of Subpart I. Since Framework A would include design and analyses requirements related to aircraft impact assessments in proposed subpart C and the content of FSARs in proposed subpart H, criterion (viii) was added to the proposed § 53.1550 to require licensees to evaluate plant changes to ensure the protections against aircraft impacts would be maintained.

Sections 53.1560 through 53.1565 in subpart I and §§ 53.6060 through 53.6065 in subpart S would define the processes for a licensee to evaluate changes to the program documents included in the licensing basis information submitted to the NRC and to modify such programs without NRC prior approval.

Sections 53.1560 and 53.6060 would include the proposed requirements for updating program documents included in licensing basis information and would provide the equivalent of UFSAR updates for key program documents. The proposed requirements in these sections would provide a uniform approach for updating program documents, which correspond to the programs required under subparts F and P.

The proposed §§ 53.1565 and 53.6065, in Framework A and B, respectively, would provide a process for licensees to make changes to program documents included in licensing basis information without obtaining prior NRC approval. The proposed requirements would include several generic criteria that, if not satisfied, would prompt the need for NRC approval of a change to a program document. These generic criteria would include whether a change would comply with TS and NRC regulations. Another proposed criterion for evaluating changes to program documents would be conforming with program-specific requirements, including NRC-approved program documents with more specific criteria for a particular program, regulations, administrative controls sections of TS, and NRC-approved program documents.

Proposed §§ 53.1565(d) and 53.6065(d) would include specific criteria for evaluating changes to several program documents that have well established processes and guidance for licensees under parts 50 and 52. The program documents specifically addressed in the proposed sections in both Framework A and B would include those with well-defined guidance under parts 50 and 52, including quality assurance programs that would be equivalent to § 50.54(a), an emergency preparedness program that would be equivalent to § 50.54(q), and the security program documents which would be equivalent to § 50.54(p).

The proposed §§ 53.1570 and 53.6070 would establish requirements for the transfer of commercial nuclear plant licenses by providing the equivalent requirements of § 50.80 for the possible transfer of an ESP, CP, OL, or COL. Likewise, the proposed §§ 53.1575 and 53.6075 would establish requirements for the termination of an OL or

COL under Framework A or B by providing the equivalent requirements of § 50.82. Other proposed requirements related to decommissioning and license termination would be included in subparts G and Q.

Sections 53.1580 and 53.6080 would establish requirements for information requests the NRC could send to the various types of licensees under proposed Frameworks A and B and would provide requirements that would be equivalent to requirements in § 50.54(f). The proposed §§ 53.1585 and 53.6085 would provide the requirements that would be equivalent to requirements in § 50.100 to address revocation, suspension, modification of licenses, and approvals for cause within either Framework A or B. Sections 53.1590 and 53.6085 would propose to address backfitting requirements by providing requirements that would be equivalent to those in § 50.109 within Framework A and B.

Proposed §§ 53.1595 and 53.6095 would address license renewals under Frameworks A and B with simple statements that licenses may be renewed. These sections are likely to be expanded through future rulemakings to more fully describe or reference the processes related to requesting and processing applications to renew ESPs, CPs, OLs, and COLs issued under part 53 (if finalized).

Subpart J – Other Administrative Requirements

Part 53 would address various reporting and administrative requirements in subpart J for Framework A and in subpart T for Framework B. The two subparts would be essentially the same and would be included separately within the frameworks to support clarity and ease of use due to the differences in the internal references and terminology between Framework A and Framework B.

Sections 53.1600 and 53.6300 would explain the organization of the various sections within the subparts related to providing unfettered access to NRC inspectors; maintaining certain records and reporting specified events or conditions; demonstrating compliance with financial qualification requirements and providing specified financial reports; and maintaining financial protections to address potential accidents.

Sections 53.1610 and 53.6310 in Frameworks A and B, respectively, would establish requirements for unfettered access for inspections. These requirements would be equivalent to § 50.70 with only minor changes proposed to provide additional flexibilities and address possible differences related to reactors licensed under part 53 and the possibility that some commercial nuclear plants may not be assigned resident inspectors.

Sections 53.1620 and 53.6320 would provide for maintenance of records and the making of various reports to the NRC. These requirements would be largely equivalent to § 50.71. This section is not intended to reflect all provisions in § 50.71; several important requirements in § 50.71 would be captured in other sections of part 53. For example, § 53.1545 within subpart I and § 53.6045 within subpart S would provide requirements that would be equivalent to § 50.71(e), updating FSARs, and §§ 53.1680 and 53.6380, “Annual financial reports,” would provide the equivalent of § 50.71(b), which covers financial reports. A reporting requirement related to completion of power ascension testing would be added to §§ 53.1620 and 53.6320 to support the assessment of annual fees under 10 CFR part 171, “Annual Fees for Reactor Licenses and Materials Licenses, Including Holders of Certificates of Compliance, Registrations, and Quality Assurance Program Approvals and Government Agencies Licensed by the NRC,” which normally commence upon completion of those testing activities.

Sections 53.1630 and 53.6330 would establish requirements for immediate notification requirements for operating commercial nuclear plants. These requirements would be equivalent to § 50.72 with minor changes proposed to make the reporting criteria technology inclusive. The slight differences between the sections in Frameworks A and B would reflect the differences in terminology and approaches to topics such as safety functions. Whereas Framework A would refer to the derivation of safety functions in accordance with § 53.230, Framework B would refer to a standard set of safety functions used in defining safety-related SSCs and organizing principal design criteria. A separate rulemaking activity, "Reporting Requirements for Nonemergency Events at Nuclear Power Plants," has been initiated to consider possible changes to the requirements in § 50.72. At a future date, the NRC may consider reconciling future changes to § 50.72 with the requirements proposed in part 53, which have been taken or derived from the current reporting requirements.

Sections 53.1640 and 53.6340 would address the licensee event report system. These requirements would be equivalent to § 50.73 with minor changes proposed to make the requirements inclusive of various reactor technologies and to reflect appropriate internal references to other sections in Framework A and Framework B and framework-specific terminology.

Sections 53.1645 and 53.6345 would require periodic reporting of the quantity of radionuclides released to unrestricted areas in liquid and gaseous effluents. These reporting requirements in Framework A and Framework B would be largely equivalent to the reporting requirements in § 50.36a, "Technical specifications on effluents from nuclear power reactors." The only difference would be that a § 50.36a requirement to specifically address conditions where the dose to the maximally exposed individual could be significantly above design objectives would refer to a design objective of 10 mrem/year instead of referring to the design objectives in appendix I to part 50.

Sections 53.1650 and 53.6350 would include a reporting requirement to support safeguards agreements between the United States and IAEA and would be equivalent to § 50.78.

Sections 53.1660—53.1700 in Framework A and §§ 53.6360—53.6400 in Framework B, respectively, would address financial requirements and would be largely similar to existing regulations in parts 50 and 52. Sections 53.1670 and 53.6370 would be entitled “Financial qualifications” and would require applicants other than electric utilities to possess or have reasonable assurance of obtaining funds for the activities for which the license is being sought. The NRC is seeking feedback on these sections and their ramifications for merchant plants in section VII of this document. Although not addressed in these sections related to the commercial nuclear plants, potential applicants for a ML planning for the factory loading of fuel should note that § 70.23, “Requirements for the approval of applications,” includes a provision addressing financial qualifications of applicants where the nature of the proposed activities is such as to require consideration of financial matters by the Commission. The remaining financial reports in both Frameworks A and B would be equivalent to § 50.71(b) for annual financial reports, § 50.76 for a change of status, and § 50.81 for creditor regulations.

Sections 53.1710 through 53.1730 in Framework A and §§ 53.6410 through 53.6430 in Framework B, respectively, would address financial protection requirements. Sections 53.1720 and 53.6420 would require insurance to stabilize and decontaminate a plant following an accident. These requirements would be taken from § 50.54(w) with the only notable change being the addition of a provision allowing plant-specific estimates of costs to stabilize and decontaminate a plant as an alternative to the $1.06 billion minimum coverage in § 50.54(w). Sections 53.1730 and 53.6430 would refer to the requirements in 10 CFR part 140, “Financial Protection Requirements and Indemnity,” related to financial protection requirements and indemnity agreements that are often discussed in the context of the related legislation, namely the Price-Anderson Act.

Subpart K – Quality Assurance Criteria for Commercial Nuclear Plants

The proposed subparts K and U would provide a consolidated set of quality assurance requirements for applicants and licensees implementing either framework in proposed part 53. The two subparts would essentially be the same with some differences resulting from framework-specific approaches and terminology related to safety classification schemes and supporting safety analyses. Both proposed subparts K and U would be equivalent to appendix B to part 50, with the only differences being those needed to reflect part 53 terminology and safety classifications. For example, the term “commercial nuclear plant” is used throughout the proposed part 53 as a way to distinguish it from parts 50 and 52, which use terms such as “nuclear power plant,” and that difference would be reflected in the proposed subparts K and U. An example relative to only Framework A is that subpart K would not use the term “design bases,” as defined in part 50 and used in appendix B to that part. Instead, in subpart K of part 53, the staff is proposing to use the term “functional design criteria,” defined in part 53, subpart A, in place of the term “design bases.” Most of the proposed sections within subparts K and U would align directly with the associated criteria in appendix B to part 50. Proposed changes in terminology or context in subparts K and U from appendix B to part 50 are highlighted in the following discussions of specific sections.

The requirements in § 53.1800 in Framework A would be equivalent to the introduction to appendix B in part 50 and the proposed § 53.6600 in Framework B, except for the proposed use of “licensing-basis events, including DBAs, as described in § 53.240,” in lieu of “postulated accidents.” The reason for this language in proposed subpart K is to align it with the event classification terminology used in subpart C of Framework A. In defining the scope of the quality assurance requirements, both proposed subparts K and U would specify that they would apply to safety-related SSCs as defined within the respective frameworks. This change is proposed to clarify the scope of the requirements considering the differences in the safety analyses and related terminology between part 50, Framework A, and Framework B. Although there might be some differences between the SSCs that would be classified as safety-related in the three frameworks, the quality assurance requirements on those SSCs designated as safety related would be the same in part 50, Framework A, and Framework B.

Unlike criterion II, “Quality assurance program,” in appendix B to part 50 and the proposed equivalent in § 53.6610 in subpart U, the phrase “importance to safety” is not used in the proposed § 53.1810 in subpart K or elsewhere in Framework A. However, the effect of this quality assurance program requirement would be the same in both frameworks because SSCs traditionally characterized as important to safety would be captured within Framework A by the performance-based controls and special treatments developed and implemented for NSRSS SSCs within the design and analyses requirements in the proposed subpart C and the operational requirements proposed for subpart F. This proposed construct within Framework A would provide a clear distinction between safety-related SSCs and NSRSS SSCs and their relationship to the quality assurance requirements in subpart K.

Section 53.1815 in the proposed subpart K would address design control and would be equivalent to criterion III, “Design control,” in appendix B to part 50 and § 53.6615 in the proposed subpart U. However, this is an instance where subpart K would use the term “functional design criteria” instead of “design basis” to describe how design requirements for SSCs are translated into specifications, drawings, procedures, and instructions. This proposed change in terminology reflects that “design basis” is not used in Framework A in large part to avoid confusion relative to current requirements in part 50. Nonetheless, "functional design criteria” in this section would serve the same purpose as “design basis” in subpart U and appendix B to part 50 because it will ensure that the needed information for specific SSCs is translated to other engineering documents and that the appropriate quality assurance measures are applied.

V. Framework B

Subpart N – Siting Requirements

Subpart N would provide the siting requirements for Framework B. The scope of subpart N would be outlined in § 53.3505.

Section 53.3510 would provide definitions applicable to subpart N. This section would include definitions from § 100.3 and appendix S to part 50. Other definitions from the current siting requirements in § 100.3 would be included as definitions common to the entirety of part 53 in § 50.020 (e.g., “Exclusion Area,” “Low Population Zone”). Section 53.3510 would also include two terms that are not currently defined in the existing regulatory framework: “Ground Motion Response Spectra” and “Probabilistic Seismic Hazard Analysis.” These terms would be defined in a manner that is largely equivalent to their use in Framework A (subpart D) and would support the use of the alternative, risk-informed, performance-based, graded approach to seismic design provided in § 53.4733.

The proposed siting requirements are based largely on the existing requirements for siting in part 100. Section 53.3515 would provide siting requirements that would be equivalent to the existing requirements in § 100.20 with proposed changes to support the use of these requirements in Framework B. Section 53.3520 would provide siting requirements that would be equivalent to the existing requirements in § 100.21 with proposed changes to support the use of these requirements in Framework B.

Section 53.3525 would provide siting requirements that would be equivalent to the existing requirements in § 100.23, with certain changes to support the alternative use of multiple DBGMs in lieu of the single Safe Shutdown Earthquake Ground Motion when developing the Ground Motion Response Spectra for seismic design purposes.

Section 53.3525 would rely on the development and use of site-specific Ground Motion Response Spectra, as defined in § 53.3510. Ground Motion Response Spectra are based on geologic, seismic, and geotechnical investigations of the site in question. Applicants using appendix S to part 50 would use the Ground Motion Response Spectra to derive the Safe Shutdown Earthquake Ground Motion for use in the design of SSCs important to safety, consistent with the approach taken in the existing regulatory frameworks in part 50 and part 52. Under the alternatives in § 53.4733, applicants would use the Ground Motion Response Spectra to derive the DBGMs for use in the design of SSCs important to safety.

The phrase “site subsurface material properties” would be used in § 53.3525 instead of the phrase “site foundation material” that is currently in § 100.23. This difference reflects that the material properties of the subsurface layers above and below the foundation level must be accounted for when assessing soil-structure interactions.

Subpart O – Construction and Manufacturing Requirements

Proposed subpart O would provide the requirements for construction and manufacturing activities for a commercial nuclear plant for applicants and licensees under Framework B of this part. See the discussion for Framework A, subpart E, in section IV for a detailed description of the regulatory requirements for construction and manufacturing under Frameworks A and B of part 53.

Subpart P – Requirements for Operation

The proposed subpart P would provide the requirements for the operations phase of a commercial nuclear plant in Framework B. Section 53.4200 would provide a general overview of the objectives in subpart P and would draw a connection between plant safety, personnel, and programmatic controls, including those associated with maintenance effectiveness. This section would be equivalent to § 53.700 in Framework A with the exception of language specific to the safety functions in § 53.230 and other high-level performance requirements that are specific to Framework A. Framework B is more closely aligned with existing regulatory requirements (e.g., part 52) in that overarching safety goals and design requirements would be met by adherence to design criteria and other specific programmatic and technical requirements. To accommodate this difference in structure compared to Framework A, the language in § 53.4200 would not include references to specific, overarching safety functions. A definition of safety functions is included in § 53.020, which describes how safety functions are addressed for in each framework.

Section 53.4210 would address maintenance, repair, and inspection programs. The proposed requirements in this section were derived from the requirements in § 50.65 with changes proposed to adapt these requirements for Framework B. These modifications include changes to the language in § 53.4210(a)(3) to reflect that certain commercial nuclear power plants may not have refueling schedules or cycles structured similarly to those in the current LWR fleet that were used to develop the existing requirements in § 50.65(a)(3). The modified language in § 53.4210(a)(3) would still limit the maximum time between program evaluations to 24 months.

The scope of SSCs that would be covered by the maintenance, repair, and inspection programs described in § 53.4210 would be modified to ensure technology inclusiveness when compared to § 50.65(b). These modifications would be reflected in § 53.4210(b)(1), which would use the definition of safety-related SSCs from § 53.028 to assist in defining the scope of SSCs that would need to be considered, as opposed to the prescriptive list of safety-related SSCs outlined in § 50.65(b)(1). One additional modification made in § 53.4210(a)(2), as compared to § 50.65(a)(2), would be the requirement that documentation be maintained to demonstrate that an SSC monitoring is not required for certain SSCs when the licensee is able to demonstrate that they are being appropriately maintained. This modification would ensure that there is clarity and traceability in establishing that a particular SSC demonstrates compliance with the requirements in § 53.4210(a)(2).

The proposed requirements for TS in § 53.4213 would be largely equivalent to the requirements in § 50.36 for plants licensed under the existing requirements in part 50 or part 52. Unlike proposed subpart F, which did not carry over into Framework A certain specific requirements from § 50.36 such as the safety limits, limiting safety system settings, and limiting control settings, as explained in the discussion of subpart F, Framework B is carrying over most of the requirements from § 50.36 because Framework B aligns more closely with the traditional licensing approach in part 50 and part 52. Modifications to the requirements taken from § 50.36 would support technology inclusiveness. Modifications to the language taken from existing requirements are also proposed to limit the applicability of these requirements to licenses issued under AEA section 103.

The requirements for establishing LCOs were the only modifications made in comparison to the analogous requirements in § 50.36. Specifically, of the four criteria covered under proposed § 53.4213(b)(2)(ii), criteria 1, 3, and 4 were modified when compared to § 50.36.

Proposed modifications to criterion 1 in § 53.4213(b)(2)(ii)(A) would acknowledge that the reactor coolant pressure boundary for a LWR provides a fission product retention barrier for the release of radionuclides. However, in some non-LWRs, the reactor coolant pressure boundary may not serve this function. The language proposed in § 53.4213(b)(2)(ii)(A) would add flexibility for designs in which fission product retention would be provided by functional containment rather than a reactor coolant pressure boundary. Modifications to criterion 3 in § 53.4213(b)(2)(ii)(C) would acknowledge that there are aspects of the functional containment that are not part of the “primary success path” but are still expected to be available. Modifications to criterion 4 in § 53.4213(b)(2)(ii)(D) would provide flexibility in that a risk evaluation could be used to inform the establishment of an LCO. This is broader than the current language in

§ 50.36(c)(2)(ii)(D) which only focuses on establishment of an LCO based on operating experience or a PRA. Risk evaluations in Framework B refer to either a PRA or an AERI, the latter of which could be used to determine whether an LCO is warranted for applicants satisfying the criteria in § 53.4730(a)(34)(ii). All other provisions from § 50.36 were carried over to the proposed requirements in § 53.4213 with no material changes.

The proposed requirements in § 53.4215 are derived from § 50.54(ff) with no material changes made to the underlying requirements relative to shutting down and restarting a commercial nuclear plant following vibratory ground motion that exceeds the operating-basis earthquake (OBE) ground motion or causes significant damage to the plant. Proposed requirements in § 53.4733 would provide, and appendix S to part 50 would be amended to also provide, related requirements for the OBE ground motion for Framework B licensees and applicants.

The requirements for staffing, training, personnel qualifications, and human factors engineering in Framework B would be collocated with those in Framework A. Section 53.4220 would reflect this collocation and would note that the requirements for these areas in proposed §§ 53.725 through 53.830 would be applicable to Framework B.

Section 53.4300 would address programs and would provide equivalent requirements to those proposed in § 53.845, including the flexibilities to combine, separate, and otherwise organize programs and related documents as appropriate for the technologies and organizations associated with the commercial nuclear plant.

Section 53.4310 would address the programmatic requirements for radiation protection. The programmatic requirements for radiation protection in Framework B would be equivalent to those under Framework A in § 53.850. The proposed rule language in § 53.4310 would only vary from § 53.850 due to different internal references for the two Frameworks. These references would be reflected in § 53.4310 as conforming changes, when compared to § 53.850, such that § 53.4310 would reference, for Framework B, the equivalent sections, paragraphs, and subparagraphs of those in Framework A.

Section 53.4320 would address emergency preparedness program requirements. These requirements would be collocated with those under Framework A in proposed § 53.855. A reference to the requirements in § 53.855 would be provided.

Section 53.4330 would address requirements for security programs. The programmatic requirements for security in Framework B would be equivalent to those under Framework A in proposed § 53.860. The proposed rule language in § 53.4330 would only vary from § 53.860 due to different internal references for the two Frameworks, such that § 53.4330 would reference, for Framework B, the equivalent sections, paragraphs, and subparagraphs as those referenced by § 53.860 in Framework A.

Section 53.4340 would address requirements for quality assurance programs. The programmatic requirements for quality assurance in Framework B would be largely equivalent to those under Framework A in proposed § 53.865. The proposed rule language in § 53.4340 would only vary from § 53.865 due to the difference in internal references for the two Frameworks (i.e., proposed subpart K for Framework A and proposed subpart U for Framework B).

Section 53.4350 would provide requirements for fire protection under Framework B. Section 53.4350(a) would describe the requirements for the fire protection plan. It would be equivalent to § 53.875(a) in Framework A and to § 50.48(a). Section 53.4350(a)(3) would describe the fire protection plan requirements for design certifications and standard designs. Section 53.4350(b) would establish the general requirements for the fire protection program and would be equivalent to section II.A of appendix R to part 50. Section 53.4350(b)(3) would be equivalent to § 53.250 of Framework A. Section 53.4350(b)(4) would establish that there are performance requirements for the fire protection program and that a fire hazards analysis must be part of the fire protection program.

Section 53.4350(c) would establish general performance requirements for the fire protection program established in § 53.4350(b). It would be equivalent to § 53.440(e) in Framework A and General Design Criterion 3 of appendix A to part 50. The GDC establish minimum requirements for the principal design criteria for water-cooled commercial nuclear plants. However, as indicated in appendix A to part 50, the GDC are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

Section 53.4350(d) would establish the general requirements for performing the fire hazards analysis. It would be equivalent to § 53.450(g)(1) of Framework A and section II.B of appendix R to part 50.

Section 53.4360 would provide requirements for in-service inspection (ISI) and in-service testing (IST) for applicants and licensees under Framework B. The proposed requirements would be bifurcated for technology inclusiveness. Boiling or pressurized water-cooled commercial nuclear plant licensees would be required to demonstrate compliance with the existing requirements under § 50.55a. The applicability of § 50.55a is limited to LWRs under Framework B to ensure that Framework B provides an equivalent level of safety as an LWR licensed under part 50 or part 52. Non-LWR licensees would be required to demonstrate compliance with the same requirements for ISI and IST in Framework A in § 53.880.

Section 53.4380 would provide the requirements for environmental qualification (EQ) of electric equipment important to safety for commercial nuclear plants licensed under Framework B. This section would provide requirements that would be equivalent to those currently in § 50.49, with minor exceptions. Notably, in § 53.4380(e)(3), combustion would be added as a new chemical effect to consider as part of the EQ program due to the wide spectrum of fluids and materials that may be employed in certain non-LWR designs. Sodium combustion and the resulting combustion products (e.g., NaOH) provides one example of a chemical effect specific to a particular type of reactor design. Additional examples include combustion products from sodium-concrete reactions that lead to hydrogen release, burning, and non-sodium fires. Other modifications to the proposed requirements, as compared to the existing requirements in § 50.49, would conform to the requirements to Framework B.

Section 53.4390 would provide requirements for the development, implementation, and maintenance of procedures and guidelines. This section would provide the equivalent of the requirements in § 53.910 with conforming changes proposed to accommodate the use of these proposed requirements in Framework B. Section 53.4390(b)(5) would provide the equivalent requirements to § 50.54(hh) regarding potential aircraft threats that are applicable to licensees under the existing regulatory frameworks in part 50 and part 52.

Section 53.4400 would address the requirements for integrity assessment programs. The language in proposed § 53.4400 was developed to mirror the requirements in proposed § 53.870 to ensure that licensees under Framework B adequately address the effects of aging, cycling and transient loads, and other degradation mechanisms on certain SSCs. The primary difference between the integrity assessment program requirements in Framework B, as compared to Framework A, would be the scope of SSCs that would be included in the program. This difference would result from the fundamental variation in how SSCs are classified and categorized between the two frameworks. The scope of SSCs within the integrity assessment program under proposed Framework B was developed to be equivalent to the scope of SSCs for which maintenance effectiveness would be assessed under § 53.4210(b).

Section 53.4410 would establish regulatory requirements for primary containment leakage rate testing programs for water-cooled commercial nuclear plants. This section would provide the equivalent of § 50.54(o) with conforming changes proposed to support the use of these provisions in Framework B. This requirement is being proposed to accommodate the potential for new water-cooled commercial nuclear plants licensed under Framework B. Inclusion of this proposed requirement would provide assurance that leak-tight containments used in water-cooled commercial nuclear plant designs are designed, operated, and maintained consistent with the requirements under the existing regulatory framework to ensure that LWR designs licensed under Framework B will provide an equivalent level of protection of public health and safety as LWR designs licensed under parts 50 or part 52.

Section 53.4420 would establish the regulatory requirements for commercial nuclear plant applicants and licensees to mitigate beyond-design-basis events. The proposed requirements would be largely equivalent to the existing, analogous requirements under § 50.155, with certain modifications to ensure technology-neutrality. Examples of modifications proposed include the replacement of light-water-reactor-specific damage states (e.g., loss of normal access to the normal heat sink for passive reactor designs) with a more generic focus on damage states that would immediately challenge the safety functions of the commercial nuclear plant.

Subpart Q – Decommissioning

Proposed subpart Q would provide the requirements for decommissioning of a commercial nuclear plant for applicants and licensees implementing Framework B of this part (subparts N through U). The requirements for decommissioning in this subpart would be equivalent to those provided in proposed subpart G, except for minor reference changes. Specifically, the only variations between proposed subpart G in Framework A and proposed subpart Q in Framework B would be the references to various sections throughout part 53 (i.e., inter- and intra-subpart references in proposed subpart Q are made to the analogous sections in Framework B). See the discussion for Framework A, subpart G, in section IV for a more detailed description of the regulatory requirements for decommissioning.

Subpart R – Licenses, Certifications, and Approvals

Subpart R would provide requirements for applications under Framework B for NRC licenses, certifications, or approvals for commercial nuclear plants. The proposed requirements in subpart R would govern general application requirements applicable to all Framework B applications as well as specific application requirements for Framework B applicants for LWAs, ESPs, standard design approvals, standard design certifications, MLs, CPs, OLs, and COLs. Accordingly, the proposed requirements in subpart R would cover all of the licensing, certification, and approval processes currently covered under parts 50 and 52, with the exception of the process for early review of site suitability issues. As with the proposed rules in Framework A, interactions with external stakeholders during the development of the proposed rule did not identify significant interest in or need for including the process for early review of site suitability issues in part 53. Consequently, much of the proposed subpart R regulatory text is identical to the corresponding language in parts 50 and 52, as described in the following paragraphs, with minor changes to account for cross references in Framework B, to make language technology inclusive, or to reflect the unique analytical approach in Framework B. In these instances, this preamble discussion will describe the language as "equivalent" to the existing corresponding requirement in part 50 or part 52 and will describe any deviations, where applicable.

The proposed structure and requirements in subpart R closely align with the proposed structure and requirements in subpart H. All of the proposed sections that would provide administrative and process (i.e., non-technical) requirements are identical. Examples of these requirements include issue finality, referrals to the ACRS, and the duration of a given license. The explanation of those sections in the portion of the Preamble discussing subpart H of Framework A also apply to the corresponding section in subpart R and are not repeated here. For example, the explanation of § 53.1143, "Filing of applications," for ESPs in Framework A covers § 53.4753, "Filing of applications," for ESPs in Framework B. Therefore, the subsequent discussion regarding the proposed requirements in subpart R focuses only on those sections, paragraphs, and subparagraphs that differ between subpart H and subpart R.

Section 53.4730 would provide general technical requirements. Subsequent technical contents of application sections for the individual application types would specify which general technical requirements in § 53.4730 apply. For example, § 53.5016 would denote which requirements in § 53.4730 are applicable to a COL application. Consolidation of the technical requirements into one section would minimize the length of the rule and limit the potential that future modifications to these requirements would be implemented inconsistently across the application types.

The majority of the requirements proposed in § 53.4730 would be derived from comparable requirements in § 52.79, which governs contents of applications for COL applications in Part 52. Detailed discussion of requirements that would differ from those in existing part 50 or 52 requirements is provided in the following paragraphs. Additional information regarding the source reference for these requirements is also discussed, as applicable.

Section 53.4730(a)(1) would contain site safety analysis requirements, derived from § 52.79(a)(1), which provides content of application requirements for a Part 52 COL, and would provide similar requirements for other Framework B license application types as described in § 53.4730(a). Modifications have been made to accommodate different licensing applications apart from a COL, such as an OL, CP, Design Certification Rule (DCR), Standard Design Approval (SDA), or a ML, and to provide for a sufficiently technology-inclusive set of requirements related to radiological releases in § 53.4730(a)(1). To maintain technology inclusivity, § 53.4730(a)(1)(vi) includes the language “fuel or core damage or potential for large radiological releases from sources other than the reactor system” to address all radiological accident sources, including those that are not inside the traditional reactor coolant boundary but directly support reactor operation, in order to account for novel design features. Examples could include fuel/coolant cleanup systems in molten fuel designs or online continuous fueling systems.

Due to the broad range of technologies currently being considered, and in order to provide rule text that can accommodate further unanticipated technology types, the requirement in § 53.4730(a)(1) would be written such that multiple approaches are viable for demonstrating compliance with the regulation. The requirement in § 53.4730(a)(1)(vi) to analyze a “postulated fission product release, using the expected demonstrable barrier leak rate(s) and any fission product cleanup systems intended to mitigate the consequences of the accidents” does not require consideration of a postulated fission product release that would result in potential hazards exceeded by those from any accident considered credible, as required in the corresponding provision in § 52.79(a)(1)(vi). Instead, two options would be provided in § 53.4730(a)(1)(vi) to provide flexibility: use of a mechanistic source term derived from physically representative models of the facility response, or use of a bounding assessment assuming severe plant conditions, such as those evaluated when considering severe accident vulnerabilities as would be required in § 53.4730(a)(5)(iv). In both cases, the assessment must be based on conditions more severe than those analyzed for DBAs in proposed § 53.4730(a)(5)(ii), and must serve to effectively demonstrate defense in depth, consistent with Commission policy.

Similar to the existing requirements in § 52.79(a)(1)(vi), the fission product release assumed for this evaluation in § 53.4730(a)(1)(vi) must be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of credible accidental events. Realistic best estimate assessments (that include considerations for uncertainty) of the scenario, whether directly analyzed as part of a physical sequence used in a mechanistic source term or created to provide a demonstrably bounding event, would be appropriate for use in performing the assessments under this section.

Section 53.4730(a)(1)(vi) would also explicitly require that the application contain information related to the barriers credited in the assessment. This information is intended to in part address how the design provides defense in depth through multiple barriers. Due to the variety of potential radionuclide release mitigation options, this section would not presume a fuel-coolant boundary and integral leak-tight containment barrier arrangement like the more requirements in Parts 50 and 52 that were developed for LWRs. Accordingly, the purpose of proposed § 53.4730(a)(1)(vi) is to ensure that doses to the public remain below acceptable levels in the event of a major accident. Acceptance criteria in § 53.4730(a)(1)(vi)(A) and (B) for these analyses would be the same as the comparable requirements in each application type in parts 50 and 52, with the addition of a supplemental requirement in § 53.4730(a)(1)(vi)(C) that the design satisfy acceptable dose consequence criteria. This supplemental requirement was added for applications where the applicant elects to use a more restrictive set of dose consequence criteria (e.g., 1 rem TEDE over 96 hours) to demonstrate compliance with other alternative requirements (e.g., emergency preparedness for small modular reactors [SMRs] and other nuclear technologies) that may be applicable to a particular applicant.

Section 53.4730(a)(2) would provide requirements related to the facility description to be provided in the FSAR and is derived from § 52.79(a)(2), which applies to a Part 52 COL, and would provide similar requirements for other Framework B license application types as described in § 53.4730(a). This section would require that information on SSCs and facility design features be discussed insofar as they are pertinent to the safety of the facility, and examples of SSCs expected to fall within that category are included in the regulation. These examples are not exclusive or limiting, and what SSCs fall within this category will be dependent on the design. Safety impacts of SSCs would need to consider interfaces with other aspects of the facility. Details of the design or function of many secondary system components may not be pertinent to the safety of a given facility, but the characteristics and boundary conditions associated with the secondary system interface may be pertinent to the response of the facility and therefore should be provided.

Section 4730(a)(3) would provide requirements related to the kinds and quantities of radioactive materials to be discussed in the FSAR and is derived from § 52.79(a)(3) for a COL and similar requirements for other license application types. Section 53.4730(a)(3) would include additional language beyond that currently in § 52.79(a)(3) to clarify the role of a combination of programmatic controls and design features function to satisfy ALARA principles. The proposed rules relative to ALARA principles are equivalent to those proposed in § 53.260(b) under Framework A.

Section 53.4730(a)(4) would require applicants to provide principal design criteria (PDC). Use of design criteria is foundational in providing regulatory evaluation standards for a deterministic approach. Section 52.79(a)(4) provides a comparable requirement in the existing regulations. But § 53.4730(a)(4) would more clearly delineate the requirements for LWR PDC, which are required to be based on the GDC in appendix A to part 50, and non-LWR PDC, which are required to provide PDC that establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs important to safety, consistent with existing requirements in Appendix A to part 50 for the role of the PDC. Non-light-water reactor applicants should use the GDC or other generally accepted consensus codes and standards (such as RG 1.232) as guidance to inform the development of the provided PDC.

Section 53.4730(a)(5) would contain requirements for analysis and evaluation for initiating events and is derived from § 50.34(a)(4) (similar requirements exist for part 52 applications). The requirements would provide an additional level of detail with respect to the categorization of events and the associated acceptance criteria and analysis requirements. These requirements would be consistent in concept with existing regulations and international standards for these classes of events. The phrase “to include the cumulative risk from all radionuclide sources on site licensed under Framework B of part 53” would be included in § 53.4730(a)(5)(i)(B) to make it clear that the analysis must consider all radionuclide sources licensed as part of the commercial nuclear reactor license, but not others (such as spent fuel from a licensed ISFSI). This clarification would consistently regulate radionuclide sources while providing a technology-inclusive pathway to do so, as some reactor designs may not confine all the radionuclide sources within a single component (the fuel plus any coolant activity, in the case of traditional LWRs).

Section 53.4730(a)(5)(ii) would set forth requirements for DBAs. It would require applicants to define acceptance criteria for DBAs, similar to the existing part 50 and 52 requirements for LWR fuel acceptance criteria. SSCs used to mitigate the effects of DBAs would be required to be safety-related, consistent with existing regulations and practices for this class of events. The requirements in § 53.4730(a)(5)(ii)(D) would provide an avenue for an applicant to provide bounding analyses (potentially involving conservative (unrealistic) assumptions to adequately bound events) for some or all of the analytical requirements for this part. This is largely consistent with existing practice—a single analysis to cover a category of event (e.g., overcooling) is often provided as part of a safety analysis. Section 53.4730(a)(5)(ii)(D) would go a step further and also allow for bounding analyses to be provided to cover larger portions of the DBA analytical space, provided the analysis envelopes the full range of conditions it is stated to bound.

Section 53.4730(a)(5)(ii)(E) and (F) would include requirements largely equivalent to § 50.46, which addresses emergency core cooling systems, for all reactor designs. Applicants would be required to identify surrogate safety acceptance criteria, akin to peak cladding temperature for LWRs, and track and report errors in the analysis for these acceptance criteria. These acceptance criteria would offer the flexibility to use either (1) the traditional specified acceptable fuel design limits (e.g., departure from nucleate boiling to protect fuel cladding), (2) the specified acceptable SARRDL concept, discussed in RG 1.232, or (3) other acceptance criteria of the applicant's choosing that serve a similar role in their analysis. For LWRs, NRC anticipates the § 50.46 criteria will be the ones chosen.

Section 53.4730(a)(5)(iii) would provide requirements for normal operation and AOOs. For AOOs in particular, applicants would be required to provide analyses that demonstrate the consequences of AOOs comply with part 20 acceptance criteria, consistent with the existing regulatory framework. Sections 53.4730(a)(5)(iii)(B) and (C) would require that applicants analytically demonstrate that AOOs can be precluded from escalating to more severe events and do not impair the capability of safety-related SSCs to perform safety functions to mitigate design-basis events, consistent with current requirements for this class of events in parts 50 and 52.

Section 53.4730(a)(5)(iv) would establish requirements for a subset of events with consequences potentially greater than DBAs, referred to as “additional licensing-basis events” (ALBEs). The deterministic process for analyzing design-basis events in Part 50 that Framework B is based on dates back more than 50 years in the nuclear industry and is well-established as a means to provide a reasonably comprehensive assessment of challenges to a design. However, decades of operating experience and research throughout the world have shown there are limitations associated with such an approach. In particular, the approach may not adequately address challenges that occur due to overreliance on a single analysis, design function or feature within the analysis, or considerations resulting from common cause failure from a single initiating event.

Consistent with the reasoning outlined in SECY-83-293, “Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," and the June 21, 1988 rulemaking 10 CFR 50.63, “Loss of all alternating current power,” (52 FR 23203), the NRC has deemed it appropriate to provide requirements for reducing risk associated with credible events that have the potential to lead to plant conditions that exceed design parameters and can lead to unacceptable consequences. Section 53.4730(a)(5)(iv)(A) would require that applicants perform assessments for events that are unlikely but credible that could lead to situations not considered for DBAs. Although not subject to the full panoply of requirements that apply to the analysis of DBAs and the inclusion of design features and other measures to prevent or mitigate DBAs, ALBEs would be subject to the requirements that § 53.4730(a)(5)(iv)(A) would establish.

Section 53.4730(a)(5)(iv)(B) would require mitigation of recognized initiators or complex accident sequences that may have substantial uncertainty associated and are of similar consequence to a recognized initiator. Given the reduced role of PRA in

Framework B, which is a central tenet of this framework, it is difficult to directly establish a threshold for the area of concern. The provisions in § 53.4730(a)(5)(iv)(B) would include requirements for events of comparable frequency to design-basis events (considering uncertainty) that are not captured within the scope of the requirements associated with those design-basis events and with the potential for unacceptable consequences. For existing light-water reactors, operating experience has shown that ATWS and SBO events (as examples) were credible and had the potential to result in core damage, challenges to containment, and the release of radioactivity to the public and environment. The conceptual phenomena associated with these events could also apply to other reactor technology types.

Under the proposed approach in § 53.4730(a)(5)(iv), a designer or applicant would have numerous alternatives beyond adding systems to address such events: credit an already existing (non-safety-related) system or procedural action with mitigating the event in question and either augment the quality or availability of the system or identify how or why the alternative would be expected to be available to mitigate the event; provide additional analysis or testing to demonstrate the frequency of the event is not similar to design-basis events, considering the efficacy of systems that already exist and are credited to mitigate similar design-basis events; or identify how defense-in-depth measures (e.g., additional barriers, programmatic controls) for the design prevent consequences from reaching unacceptable levels by providing sufficient safety margin. Structures, systems, and components identified to perform this mitigation could, but not need be, safety-related, in accordance with § 53.4730(a)(5)(iv)(B)(*1*).

Section 53.4730(a)(5)(iv)(A) would identify events for which § 53.4730(a)(5)(iv)(C) would establish performance, reliability, and availability targets for safety functions. This requirement would be akin to the regulatory treatment of non-safety systems (RTNSS) under part 52.

Section 53.4730(a)(5)(v) would require a description and analysis of design features for the prevention and mitigation of severe accidents. This requirement is intended to provide a technology-inclusive requirement filling the same role as § 52.79(a)(38). This paragraph would require applicants to provide information regarding safety features and barriers used in the analysis and identify severe accident vulnerabilities that could result in fission product releases. Notably, the proposed requirements would not contain dose or analytical acceptance criteria—only analysis and evaluation requirements, consistent with Part 52.

Section 53.4730(a)(5)(vi) would provide a new technical requirement for applicants to consider chemical hazards of licensed material. The broad spectrum of reactor technologies that could be licensed under Part 53 includes those using coolants and other materials that pose unique chemical hazards in addition to radionuclide source terms. The language in this section would be based on the requirement proposed in § 53.440(k) with conforming changes made to provide the same technical requirement in Framework B.

Section 53.4730(a)(6) would provide requirements for fire protection and would be equivalent to the existing requirements under § 52.79(a)(6) and 52.79(a)(40), with conforming changes to reference equivalent requirements for operation under subpart P.

Section 53.4730(a)(7) would provide requirements for combustible gas control and would be equivalent to the existing requirements under § 52.79(a)(8). The requirements from § 50.44 that would be referenced in this paragraph are considered sufficiently technology-inclusive such that § 50.44 could be met by any future design without the need for an exemption.

Section 53.4730(a)(8) would provide application requirements for the EQ of electric equipment important to safety and would be equivalent to the existing requirements under §§ 52.79(a)(10) and 50.49, by extension. Modifications from the source requirements in parts 50 and 52 have been proposed to reflect equivalent program requirements for EQ in subpart P (§ 53.4380). The requirements in § 53.4730(a)(8) would be reorganized when compared to § 52.79(a)(10). This reorganization reflects that the level of detail provided for EQ of electric equipment important to safety varies depending on the application type.

Section 53.4730(a)(9) would provide requirements for the role of personnel and would be equivalent to the existing requirements under § 52.79(a)(14) and § 52.79(a)(34). Modifications from the source requirements in part 52 have been proposed to reflect references to the relevant requirements for operation (e.g., operator licensing) that have been proposed in subpart F.

Section 53.4730(a)(10) would provide application requirements for maintenance effectiveness programs and would be equivalent to the existing requirements under §§ 52.79(a)(15) and 50.65, by extension. Proposed modifications reflect conforming changes made to cross-reference the equivalent requirements for operation under subpart P.

Section 53.4730(a)(11) would require applicants to identify ALARA design objectives for limiting radiation dose to members of the public. It would also require that the Safety Analysis Report include information to demonstrate that the design is adequate to demonstrate compliance with the ALARA design objectives during normal reactor operations, included expected operational occurrences (not accidents). The Safety Analysis Report would estimate the quantities of radionuclides to be released during normal reactor operation and the dose to the maximally exposed member of the public from the planned licensed operation.

The requirement in § 53.4730(a)(11) would be similar to the existing requirements under § 50.34a, which require that ALARA design objectives be developed for normal effluents and that applicants describe the means for keeping doses ALARA for part 50 and part 52 reactor licensees. However, instead of referencing appendix I to part 50 for the design objectives (as is done in § 50.34a), a criterion of 10 mrem/year total effective dose equivalent (TEDE) for planned licensed operation is proposed. Reasons for the change include the following. First, appendix I to part 50, is based on International Commission on Radiological Protection (ICRP) Publication 2 dose methodology, which is inconsistent with most of the other regulatory dose criteria used by the NRC, such as the dose criteria in part 20, which use TEDE criteria (ICRP 26 and 30). The use of TEDE criteria would allow the same dose methodology to be used for part 20 and the ALARA design objectives in Framework B. In addition, TEDE criteria include organ dose weighting factors, which allows for simplified dose criteria, instead of including separate criteria for organ doses. Second, the proposed part 53 TEDE criteria for the ALARA design objectives would also include consideration of direct radiation (non-effluent) exposure. This is appropriate because while the quantities to radioactive material released from the facility are expected to be low and the total dose to the public is expected to be low, many new reactors may have a smaller site footprint and members of the public could also be located nearer to radiation sources. For current licensees, direct radiation is normally included in the offsite dose calculation manual and included in annual radioactive effluent reports. Last, the 10 mrem/year TEDE dose criterion is consistent with § 20.1101(d), which is used for non-power reactor air emissions.

As stated in the proposed requirements under § 53.4730(a)(11) and consistent with § 50.34a and appendix I to part 50, the 10 mrem/year ALARA design objective for the planned licensed operation is not a dose limit and different criteria for design objectives may be allowed with justification, based on a specific applicant or licensees circumstances. While the part 53 ALARA design requirements do not include a cost-benefit population dose criterion, as is used in section II.D of appendix I to part 50, a cost-benefit analysis may be used to justify an alternative design objective value. NUREG-1530, Revision 1, “Reassessment of NRC’s Dollar Per Person-Rem Conversion Factor Policy, Final Report,” and RG 1.110, “Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors,” provide guidance that may be useful in evaluating the cost-benefit of adding addition features to reduce the radiation exposure to members of the public.

While the design objective is not a limit for operation, licensees would generally maintain the release of radioactive material effluents and the dose to members of the public in unrestricted areas at small percentages of the dose limits specified in § 20.1301 and within design objectives for the licensed operations. At the same time, the licensee would be permitted flexibility of operation. Section 53.4730(a)(11) would nonetheless require a licensee using this flexibility to exert its best efforts to keep levels of radioactive material in effluents and doses to members of the public as low as is reasonably achievable.

Section 53.4730(a)(12) would provide three requirements for post-accident radiation monitoring and protection that would be equivalent to the existing requirements under § 50.34(f)(2)(vii), 50.34(f)(2)(viii), and 50.34(f)(2)(xv). Consistent with the existing requirements under § 50.34(f), these would only be required if they are technically relevant to an applicant's proposed design.

Section 53.4730(a)(14) would provide requirements for the seismic design of certain SSCs for applicants under Framework B that are equivalent to the existing requirements under § 52.79(a)(19) and appendix S to part 50, by extension, with modifications to conform the existing language to Framework B. These conforming changes include language acknowledging that the GDC in appendix A to part 50 provide guidance for establishing the PDC for other designs even though the GDC may not be directly applicable to all designs under Framework B. This is specifically reflected in the requirements, which would require that applicants must demonstrate compliance with the PDC corresponding to GDC 2. This paragraph would also provide flexibility for applicants, with an option to use alternative, risk-informed and performance-based seismic design requirements under subpart N if sufficient risk insights are available.

Section 53.4730(a)(15) would provide application requirements for emergency plans and would be equivalent to the existing requirements under § 52.79(a)(21) with conforming changes made to reference specific program requirements under subpart P.

Section 53.4730(a)(16) would provide requirements for State, participating Tribal, and local government cooperation in emergency planning and would be equivalent to the existing requirements under § 52.79(a)(22) with modifications related to participating Tribal organizations. These also align with the Commission's "Tribal Policy Statement" (82 FR 2402) and the 2019 issuance of NUREG-0654/FEMA-REP-1, Revision 2, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” which encourages the involvement of Tribal Governments in NRC activities.

Section 53.4730(a)(17) would provide requirements for safety feature testing, analyses, operating experience, and prototypes and would be equivalent to the existing requirements under §§ 52.79(a)(24) and 50.43(e) with conforming changes made to reference requirements equivalent to § 50.43(e) that are proposed in § 53.090(c).

Section 53.4730(a)(18) would provide application requirements for quality assurance. Section 53.4730(a)(18)(i) would be equivalent to the existing requirements under § 50.34(f)(3)(iii), with modifications made for conforming changes. Section 53.4730(a)(18)(ii) would be equivalent to the existing requirements under § 52.79(a)(25), with modifications made for conforming changes.

Section 53.4730(a)(19) would provide requirements for organizational structures and would be equivalent to the existing requirements under § 52.79(a)(26).

Section 53.4730(a)(20) would provide requirements for managerial and administrative controls and would be equivalent to the requirements under § 52.79(a)(27), with conforming changes, and § 50.34(f)(3)(vii), with modifications made to limit the applicability of this requirement based on its relevance to a given applicant consistent with the existing applicability of requirements under § 50.34(f). Specifically, the phrase “as applicable” would serve the same purpose as the phrase “technically relevant” in § 50.34(f) and applicants would only have to address this requirement if it were applicable. Additional changes would include changing “nuclear steam supply vendor” to “commercial nuclear reactor vendor” to ensure that the requirement is technology-inclusive.

Section 53.4730(a)(21) would provide requirements for preoperational testing and initial startup and would be equivalent to the existing requirements under § 52.79(a)(28).

Section 53.4730(a)(22) would provide requirements for normal operations and maintenance and would be equivalent to the existing requirements under § 52.79(a)(29) with conforming changes made to reference specific program requirements under subpart P.

Section 53.4730(a)(23) would provide application requirements for TS and would be equivalent to the existing requirements under § 52.79(a)(30) with conforming changes made to reference specific requirements under subpart P and modifications to differentiate the requirements applicable to different application types. Specifically, these modifications would acknowledge the difference in level of detail for TS required for each application type and would be consistent with the existing requirements under parts 50 and 52.

Section 53.4730(a)(24) would provide application requirements for FFD programs and would be equivalent to the existing requirements under § 52.79(a)(44) and part 26, by extension. Changes that are proposed for part 26 as part of this rulemaking would also be applicable to Framework B.

Section 53.4730(a)(25) would provide requirements for assessing the risks associated with multi-unit sites (construction-related impacts on operating reactors) and would be equivalent to the existing requirements under § 52.79(a)(31).

Section 53.4730(a)(26) would provide requirements for the technical qualifications of an applicant and would be equivalent to the existing requirements under § 52.79(a)(32).

Section 53.4730(a)(27) would provide application requirements for training programs and would be equivalent to the existing requirements under § 52.79(a)(33) with conforming changes made to reference specific program requirements under subpart F (subpart P points to subpart F for these requirements).

Section 53.4730(a)(28) would provide application requirements for physical security programs and would be equivalent to the existing requirements under § 52.79(a)(35) with conforming changes made to reference specific program requirements under subpart P.

Section 53.4730(a)(29) would provide application requirements for safeguards, security, and related training and qualifications and would be equivalent to the existing requirements under § 52.79(a)(36) with a modification that would include an additional reference to § 73.22.

Section 53.4730(a)(30) would provide requirements for assessing operating experience and would be equivalent to the existing requirements under § 52.79(a)(37) with modifications that acknowledge that newer designs may have limited operating experience.

Section 53.4730(a)(31) would provide application requirements for radiation protection programs by requiring a description of the radiation protection program required by § 53.4310, which in turn requires compliance with the requirements of part 20. This section would be equivalent to the existing requirements under § 52.79(a)(39) with conforming changes made to reference specific program requirements under subpart P.

Section 53.4730(a)(32) would provide requirements for preventing criticality accidents and would be equivalent to the existing requirements under § 52.79(a)(43) with modifications that reference the proposed requirements in § 53.440(m) in lieu of the existing requirements under § 50.68.

Section 53.4730(a)(33) would provide requirements for minimization of contamination to facilitate decommissioning and would be equivalent to the existing requirements under §§ 52.79(a)(45) and 20.1406, by extension.

Section 53.4730(a)(34) would require a description of a risk evaluation and its results in the Safety Analysis Report. Framework B would maintain the traditional role of specific design rules (including use of the single failure criterion as a tool in the reactor safety review process and deterministic approaches to define LBEs and performance requirements for SSCs) and the establishment of principal design criteria to ensure that safety criteria are met. Therefore, the risk evaluation would be used to help confirm that a commercial nuclear plant can be constructed and operated without undue risk to the public health and safety. The risk evaluation would be based on a PRA or an AERI provided that specified entry conditions are met.

The proposed use of risk evaluation in Framework B would implement existing Commission policy. In its policy statement on the regulation of advanced reactors (73 FR 60612; October 14, 2008), the Commission stated its expectation that advanced reactor designs will comply with the Commission's safety goal policy statement (51 FR 28044; August 4, 1986, as corrected and republished at 51 FR 30028; August 21, 1986), the Commission's policy statement on the use of PRA in regulatory activities (60 FR 42622; August 16, 1995), and the Commission's policy statement on severe accidents regarding future designs and existing plants (50 FR 32138; August 8, 1985). The proposed use of PRA as one approach to develop a risk evaluation would be equivalent to existing requirements in part 52 and to proposed requirements for part 50 CPs and OLs as discussed in SECY-22-0052, “Proposed Rule: Alignment of Licensing Processes and Lessons Learned from New Reactor Licensing (RIN 3150 AI66).” The proposed use of AERI in lieu of a PRA, provided that specified entry conditions are met, would be consistent with the Commission’s policy statement on the use of PRA in regulatory activities. There, the Commission noted that “… not all of the Commission's regulatory activities lend themselves to a risk analysis approach that utilizes fault tree methods. In general, a fault tree method is best suited for power reactor events that typically involve complex systems. Given the dissimilarities in the nature and consequences of the use of nuclear materials in reactors, industrial situations, waste disposal facilities, and medical applications, the Commission recognizes that a single approach for incorporating risk analyses into the regulatory process is not appropriate.” With respect to using risk evaluation in a confirmatory role during initial licensing processes, an AERI approach would be expected to provide results that are equivalent to the results provided by a PRA if the AERI entry conditions are satisfied. Specifically, AERI would yield the following insights: (1) a demonstrably conservative risk estimate for comparison to the Commission’s QHOs; (2) a search for severe accident vulnerabilities; (3) a qualitative identification of risk insights; and (4) an assessment of defense-in-depth adequacy.

The proposed AERI entry conditions in § 53.4730(a)(34) would allow applicants for approvals, certifications, licenses, and permits under Framework B to develop an AERI in lieu of a PRA when the consequences of a postulated bounding event are below specified criteria. If the postulated bounding event demonstrates compliance with the proposed dose and distance criteria, then the proposed commercial nuclear plant would reasonably be expected to comply with the Commission’s safety goals without the need to estimate the likelihood of individual event sequences. The criteria also contain a proposed qualification to show that the dose and distance criteria would be met without reliance on active safety features, certain passive safety features, and human actions. This would limit use of the proposed AERI approach to commercial nuclear plants whose designs are relatively straightforward and do not involve overly complex systems and interactions and, accordingly, would not warrant development of a PRA to provide quantitative risk insights. The proposed AERI entry conditions would not be safety or siting criteria. Rather, they would be used to determine which applicants could develop an AERI in lieu of a PRA to demonstrate compliance with the proposed risk evaluation requirement in § 53.4730(a)(34), when the requirements to address the mitigation of beyond-design-basis events in § 53.4420 must be met, and when the requirements to address combustible gas control in § 53.4730(a)(7) must be met. In addition, the proposed AERI entry conditions would be used in combination with other conditions to determine when a commercial nuclear plant is a self-reliant mitigation facility, as provided in § 53.800(a)(2).

Section § 53.4730(a)(35) would provide requirements for applicants to assess the potential effects of aircraft impacts. These proposed requirements would be equivalent to the aircraft impact requirements in the existing regulatory frameworks in § 50.150. The proposed requirements in this section have been modified from § 50.150 to ensure that they are technology-inclusive. These modifications would allow the underlying technical requirements of the existing rules under § 50.150 to be applied consistently to any reactor technology without material changes.

Section 53.4730(a)(36) would include requirements for retaining radionuclides in a containment. The proposed rules would be split into two sets of requirements. Water-cooled reactors would be required to have a leak-tight containment structure, subject to the same requirements in place for a part 50 or 52 application. This is consistent with the existing requirements and Commission policy and is proposed to ensure a similar level of protection of public health and safety as parts 50 and 52. For non-LWRs, the proposed functional containment requirements would be provided in lieu of existing containment-related regulatory requirements, though non-LWR applicants could choose to provide a containment structure consistent with the LWR containment requirements as their functional containment. These requirements are intended to be high-level, technology-inclusive requirements for non-LWRs, consistent with the Commission policy stated in SECY-18-0096, "Functional Containment Performance Criteria for Non-Light-Water Reactors," and the associated SRM. These requirements would establish what constitutes a functional containment and makes functional containment SSC qualification commensurate with the purpose of the component (safety-related if used to mitigate against DBAs).

Section 53.4730(a)(37) would provide requirements that would be applicable only to water-cooled reactor designs. These requirements would provide an equivalent level of safety to the requirements set forth in parts 50 and 52 using a deterministic approach while also maintaining technology-inclusive requirements overall. The rules in this section would apply certain technology-specific requirements from parts 50 and 52 to applicants for water-cooled reactor designs under Framework B. The applicability of the individual paragraphs in this section would be refined further in some instances to maintain alignment with the existing requirements under parts 50 and 52 (e.g., station blackout requirements under § 50.63 are only applicable to light-water cooled commercial nuclear plants and not all water-cooled designs).

Section 53.4731 would describe risk-informed classification of SSCs and would provide alternatives equivalent to those in § 50.69. No material changes have been proposed when comparing the alternatives in § 53.4731 to those currently in § 50.69. The proposed alternatives in § 53.4731 would not be available for applicants and licensees that elect to perform an AERI approach in lieu of a PRA to demonstrate compliance with the requirements for a risk evaluation in accordance with § 53.4730(a)(34). This limitation is proposed since an AERI may not provide the quantitative risk information that has historically been required to implement the risk-informed SSC classification scheme permitted by § 50.69.

Section 53.4733 would provide alternative rules for the seismic design of certain SSCs. These rules would provide an alternative to appendix S to part 50. Section 53.4733 could be used by applicants and licensees that have risk insights sufficient to grade the assumptions and inputs necessary for the seismic analyses and qualification of SSCs important to safety. This would be accomplished through the use of DBGMs in lieu of the single safe shutdown earthquake ground motion; either would be determined in accordance with the requirements of subpart N.

Each DBGM set would have horizontal and vertical components that could be applied to the dynamic analysis and qualification of a particular SSC commensurate with the risk -significance of the SSC (e.g., less severe design assumptions for a low risk-significant SSC with a correspondingly low SDC). Guidance would be developed to provide additional clarity on the implementation of this graded approach to seismic design.

The proposed alternatives in § 53.4733 would largely be equivalent to those proposed in Framework A (§ 53.480), with some exceptions. The primary difference between § 53.480 and § 53.4733 is the SSCs to which the rules would apply. The scope of SSCs to which the alternatives in § 53.4733 would apply are those that are important to safety – in contrast, these rules would apply to safety-related and NSRSS SSCs in Framework A. The existing regulations in part 50 and part 52 use this concept for determining the scope of SSCs that must be designed to withstand the effects of earthquakes. Framework B has largely adopted the same approach as that used in the existing regulations.

The alternatives in § 53.4733 would also focus on the use of principal design criteria (PDC) instead of functional design criteria, which are used in Framework A. This difference would be reflected in §§ 53.4733(c) and 53.4733(d) (as compared to §§ 53.480(c) and 53.480(d)). The use of PDC instead of functional design criteria would be a fundamental difference between the frameworks where the former is consistent with the existing regulatory philosophy in part 50 and part 52 that aligns closely with the approach taken in Framework B. Subpart U would provide quality assurance requirements for certain design activities performed to demonstrate compliance with the alternatives under § 53.4733.

Section 53.4756 would provide the technical content of application requirements for ESPs under Framework B. This section would provide the equivalent of § 52.17 with modifications made for technology-inclusiveness and conforming changes necessary for Framework B. Most conforming changes would reference paragraphs in § 53.4730 that would be equivalent to the existing paragraph requirements and references in § 52.17.

Section 53.4809 would provide the technical content of application requirements for standard design approvals under Framework B. This section would provide the equivalent of § 52.137 with modifications made for technology-inclusiveness and conforming changes necessary for Framework B. Most conforming changes would reference paragraphs in § 53.4730 that would be equivalent to the existing paragraph requirements and references in § 52.137.

Section 53.4839 would provide the technical content of application requirements for design certifications under Framework B. This section would provide the equivalent of § 52.47(a) with modifications made for technology-inclusiveness and conforming changes necessary for Framework B. Most conforming changes would reference paragraphs in § 53.4730 that would be equivalent to the existing paragraph requirements and references in § 52.47(a).

Section 53.4841 would provide requirements for other design certification application content (ITAAC, an environmental report, and safeguards information protection) and provides the equivalent of § 53.1241 with some modifications that align this section closer to the content of application requirements for a design certification under part 52 (i.e., § 52.47). These modifications would reflect that the application requirements under Framework B are generally more aligned with those currently used under the existing regulatory framework. Specific modifications to this section, as compared to § 53.1241, would reflect the differences in the SSC classification schemes used in each framework and also reflect the use of similar, but different requirements in some portions of each framework (e.g., § 53.440(a) versus the equivalent provisions in § 53.090(c)).

Section 53.4879 would provide the technical content of application requirements for MLs under Framework B. This section would provide the equivalent of § 52.157 with modifications made for technology-inclusiveness and conforming changes necessary for

Framework B. Most conforming changes would reference paragraphs in § 53.4730 that would be equivalent to the existing paragraph requirements and references in § 52.157.

Section 53.4882 would provide requirements for other ML application content (ITAAC, an environmental report, and safeguards information protection) and provides the equivalent of § 53.1282 with some modifications that align this section closer to the content of application requirements for a ML under part 52 (i.e., § 52.158). These modifications reflect that the application requirements under Framework B would generally be more aligned with those currently used under the existing regulatory framework. Specific modifications would reflect the use of similar, but different requirements in some portions of each framework (e.g., § 53.440(a) versus the equivalent provisions in § 53.090(c)).

Section 53.4909 would provide the technical content of application requirements for CPs under Framework B. This section would provide the equivalent of § 50.34(a) with modifications made for technology-inclusiveness and conforming changes necessary for Framework B. Most conforming changes would reference paragraphs in § 53.4730 that would be equivalent to the existing paragraph requirements and references in § 50.34(a).

Section 53.4912 would establish requirements for the contents of applications for CPs, other application content and would provide the equivalent of § 53.1312. Section 53.4915 would govern standards for review of applications and administrative review of applications; hearings and provides the equivalent of § 53.1315. Section 53.4918 would address finality of referenced NRC approvals, licenses, and certifications in CP applications and would provide the equivalent of § 53.1318.

Section 53.4969 would provide the technical content of application requirements for OLs under Framework B. This section would provide the equivalent of § 50.34(b) with modifications made for technology-inclusiveness and conforming changes necessary for Framework B. Most conforming changes would reference paragraphs in § 53.4730 that would be equivalent to the existing paragraph requirements and references in § 50.34(b).

Section 53.4972 would provide requirements for other OL application content (environmental report and the mitigation of beyond-design basis events) and provides the equivalent of § 53.1372 with some modifications that would align this section closer to the content of application requirements for an OL under part 50 (i.e., § 50.34(b)). These modifications would reflect that the application requirements under Framework B are generally more aligned with those currently used under the existing regulatory framework. Specific modifications to this section, as compared to § 53.1372, would reflect fundamental differences between the frameworks when considering SSC classification schemes and related requirements for operation (e.g., availability controls). This section would also include a requirement for OL applicants to address requirements for mitigation of beyond-design-basis events through a reference to subpart P (§ 53.4220) if these applicants do not satisfy the AERI entry criteria. This proposed requirement would be aligned with requirements in the existing regulatory framework that require OL applicants to demonstrate compliance with the equivalent requirements in § 50.155.

Section 53.5016 would provide the technical content of application requirements for COLs under Framework B. This section would provide the equivalent of § 52.79 with modifications made for technology-inclusiveness and conforming changes necessary for

Framework B. Most conforming changes would reference paragraphs in § 53.4730 that would be equivalent to the existing paragraph requirements and references in § 52.79.

Section 53.5019 would provide requirements for other COL application content (ITAAC and an environmental report) and provides the equivalent of § 53.1419 with some modifications that align this section closer to the content of application requirements for a COL under part 52 (i.e., § 52.80). These modifications reflect that the application requirements under Framework B would generally be more aligned with those currently used under the existing regulatory framework. Specific modifications to this section, as compared to § 53.1419, would reflect fundamental differences between the frameworks when considering SSC classification schemes and related requirements for operation (e.g., availability controls). This section would also include a requirement for COL applicants to address requirements for mitigation of beyond-design-basis events through a reference to subpart P (§ 53.4220) if these applicants do not satisfy the AERI entry criteria. This proposed requirement would be aligned with requirements in the existing regulatory framework that would require COL applicants to demonstrate compliance with the equivalent requirements in § 50.155.

Subpart S – Maintaining and Revising Licensing Basis Information

Proposed subpart S would address the maintenance of licensing basis information for Framework B. Refer to the discussion in Framework A, subpart I, in section IV for a more detailed description of the regulatory requirements for maintaining and revising licensing basis information for licensees under Frameworks A and B.

Subpart T – Reporting and Other Administrative Requirements

Proposed subpart T would address reporting and other administrative requirements for Framework B. Refer to the discussion in Framework A, subpart J, in section IV for a more detailed description of the regulatory requirements for reporting and other administrative requirements under Frameworks A and B.

Subpart U – Quality Assurance Criteria for Commercial Nuclear Plants

Proposed subpart U would address quality assurance requirements for Framework B. Refer to the discussion in Framework A, subpart K, in section IV for a more detailed description of the regulatory requirements for quality assurance under Frameworks A and B.

VI. Changes to Other Parts of 10 CFR

10 CFR Part 26

Part 26 placeholder.

10 CFR Part 73

Part 73 placeholder.

VII. Specific Requests for Comments

The NRC is seeking advice and recommendations from the public on the proposed rule. We are particularly interested in comments and supporting rationale from the public on the following:

Part 26 – Fitness for Duty Program

Part 26 placeholder.

Part 53 – Overall Organization

Part 53 is structured as two separate and generally independent frameworks with the subparts in each framework providing technical, licensing, and administrative requirements for the various stages of the life cycle of a commercial nuclear plant. The organization of part 53 in this manner puts the complete set of requirements in one place for each framework but results in the duplication of certain requirements in the two largely independent frameworks.

The NRC is seeking comment on the proposed organization of the requirements in part 53 and possible improvements to how specific requirements (e.g., examples of which specific sections) could be consolidated or otherwise reorganized to make the rule clearer and more concise.

Part 53, Subpart B – Quantitative Health Objectives

The NRC is proposing to use the QHOs from the Commission's policy statement on Safety Goals for the Operation of Nuclear Power Plants (51 FR 28044; August 4, 1986, as corrected and republished at 51 FR 30028; August 21, 1986) as one of several performance standards in Framework A. Specifically, the QHOs would be used as a cumulative risk measure and provide one element of the safety criteria for LBEs other than DBAs in proposed § 53.220. The use of the QHOs as a performance standard in an integrated risk-informed decisionmaking process is similar to that used in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, dated January 2018 (ADAMS Accession No. ML17317A256).

The NRC is seeking comment on the use of the QHOs in Framework A as one of several performance standards; if appropriate, what other performance standard could be used instead of, or in addition to, the QHOs as part of an integrated decisionmaking process to inform the safety criteria in § 53.220 and, in turn, to address the cumulative risks posed by proposed commercial nuclear plants? Please provide your considerations and rationale for your recommendation.

Part 53, Subparts B and R – As Low as Reasonably Achievable

The NRC is proposing in both Frameworks A and B that a combination of design features and programmatic controls be used to maintain doses to the public from normal plant operation and to plant workers ALARA.

The NRC is seeking comment on whether there is a means other than including requirements in part 53 to maintain ALARA as a guiding principle for radiation protection during normal plant operation. If so, please explain how an alternative would be considered in the licensing, certification, or approval of reactor designs or specific commercial nuclear plants licensed under part 53 and what, if anything, would be appropriate to address resultant inconsistencies from the proposed alternative with requirements in parts 50 and 52.

Part 53, Subpart B – Defense in Depth

Proposed § 53.250 would establish requirements based on the longstanding NRC philosophy of providing defense in depth to address uncertainties about the design, operation, and performance of commercial nuclear plants during LBEs.

The NRC is seeking comment on the inclusion of the proposed requirements to assess and provide defense in depth. The NRC is also seeking comment on whether to include specific provisions in § 53.250 and subpart B to more explicitly address the possible role of inherent characteristics of some SSCs in preventing or mitigating unplanned events, or what alternatives to these provisions should be considered. Please provide your considerations and rationale for your recommendation.

Part 53, Subparts C and D and N and R – Earthquake Engineering

Proposed § 53.480 would establish requirements related to seismic design considerations in Framework A. Proposed § 53.4733 would provide alternative requirements for seismic design in Framework B that are similar to § 53.480. These proposed sections are intended to provide a clear connection between siting activities and seismic design activities and to support various approaches to presenting seismic hazards and addressing those hazards in designs. Both Frameworks A and B are proposing to allow approaches like those currently in parts 50 and 100 or approaches that might be endorsed by the NRC in the future that incorporate more risk insights from PRAs. An example might be standards such as the ASCE/ SEI 43-19, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities,” if these methods are endorsed or otherwise found acceptable in the future by the NRC.

The NRC is seeking comment on whether the proposed requirements for earthquake engineering provide appropriate flexibility in addressing seismic risks while also ensuring that the regulations continue to adequately address seismic hazards.

Part 53, Subparts E and O – Construction and Manufacturing

1. There are numerous references in both proposed Frameworks A and B to other NRC regulations. Examples of such references include those in proposed §§ 53.610 and 53.4110 to NRC regulations related to radiation protection (part 20), FFD (part 26), physical security (part 73), and material control and accounting (10 CFR part 74, “Material Control and Accounting of Special Nuclear Material”).

The NRC is seeking comment on whether such references to other regulations in various sections in the proposed part 53 provide benefits to applicants and licensees or could be removed to reduce the length of part 53.

2. Proposed §§ 53.610(b)(1)(iii) and 53.4110(b)(1)(iii) would require procedures that describe how construction will be controlled so as not to impact other features important to the design (e.g., dewatering, slope stability, backfill, compaction, and seepage).

The NRC is seeking comment on whether such specific requirements are useful or whether these requirements could be met through other requirements proposed in part 53 (e.g., quality assurance requirements in subparts K and U).

Part 53, Subparts E and H, and O and R – Manufacturing Licenses

1. Significant changes related to MLs are included in the proposed regulations in part 53. The proposed requirements are equivalent in Frameworks A and B with the proposed requirements governing manufacturing set forth in subparts E and O, and the proposed requirements governing the licensing processes in subparts H and R. Some of the proposed changes are intended to state requirements to govern a factory-style model that has been suggested for some microreactor designs.

The NRC is seeking comment on whether the proposed regulations are necessary and sufficient to govern various scenarios for the possible manufacturing and deployment of manufactured reactors.

2. The proposed regulations in subparts H and R allow holders of or applicants for a COL to reference a ML but do not include such a provision for the holder of a CP. This proposed change from the current relationships between subparts in part 52 was made to simplify the provisions in the proposed part 53 for licensing and deploying manufactured reactors.

The NRC seeks comment on whether the possible references to a ML by holders of a CP is a relationship that should be included in part 53 and, if so, how it would be used.

3. Proposed §§ 53.1295 and 53.4895 state that the holder of a ML could not begin manufacture of a manufactured reactor or manufactured reactor module less than 6 months before the expiration of the license. This limitation is similar to the current restriction in § 52.177, which states that the manufacture of a reactor cannot begin less than 3 years before the expiration of the license. The restriction was revised from 3 years to 6 months in the proposed part 53 in recognition of the likely use of MLs to support a factory-type model for microreactors.

The NRC seeks comment on how MLs might be used and on whether it is necessary or appropriate to revise the restrictions on when manufacturing activities could begin in relation to license expiration.

4. Proposed §§ 53.1288 and 53.4888 provide the finality provisions for MLs and include, as does existing § 52.171, limitations for the NRC to impose new requirements on either the design or the requirements for the manufacture of a manufactured reactor.

No MLs have been issued under part 52 and there is no practical experience with the proposed finality sections. While the implications of the finality provisions related to the design of a manufactured reactor can reasonably be inferred from experience with design certifications and COLs, the expectations are less clear regarding finality for "requirements for the manufacture of the manufactured reactor or manufactured reactor module."

The NRC is seeking comment on the proposed finality provisions for MLs and specifically if and how finality for manufacturing processes might be requested and used.

5. A significant change proposed for MLs in part 53 compared to the existing regulations in part 52 relate to § 53.620(d) in subpart E, § 53.4120(d) in subpart O, and the associated licensing provisions in subparts H and R. Specifically, these provisions would establish requirements for the loading of fuel into a manufactured reactor module for subsequent transport and use at a commercial nuclear plant with a COL. Because the Commission has treated fuel load into a reactor as operation, this could be accomplished under a COL or OL issued under parts 52 or 50, respectively. The proposed sections, however, include a possible Commission finding that, upon modification of a manufactured reactor to include certain design features to preclude criticality, the reactor would no longer be a utilization facility as defined under the AEA. Such a finding would allow fuel load under part 70 and the ML.

The NRC is seeking comment on whether the proposed provisions to authorize the loading of fuel into a manufactured reactor module should be included in part 53 and, if so, whether the provisions included in the proposed rule are reasonable in terms of both being workable and limiting risks to public health and safety.

Part 53, Subpart F – Staffing and Generally Licensed Reactor Operators

1. *Categories of Individuals Who May Manipulate Facility Controls*: The NRC is proposing requirements that allow the manipulation of the controls of certain facilities by generally licensed reactor operators in lieu of specifically licensed reactor operators and senior reactor operators. Reactor operators and senior reactor operators are the only categories of individuals currently allowed to be licensed to manipulate the controls of utilization facilities under part 55.

The NRC is interested in public perspectives on this proposed addition of the generally licensed reactor operator category, particularly in light of new reactor technologies and concepts of operations.

2. *Criteria for Generally Licensed Reactor Operator Staffing*: The NRC is proposing criteria under which facilities would be staffed by generally licensed reactor operators in lieu of specifically licensed reactor operators and senior reactor operators. These criteria, which are proposed for both Framework A and Framework B, establish a new class of self-reliant mitigation facilities for which distinct generally licensed reactor operator licensing and staffing requirements would apply.

The NRC is soliciting public feedback regarding whether these proposed criteria are appropriate and what, if any, alternative criteria should be considered. Please provide your considerations and rationale for your answer.

3. *Medical Requirements for Generally Licensed Reactor Operators*: Based on the criteria that a self-reliant mitigation facility must meet, the NRC is proposing to not subject generally licensed reactor operators to requirements for medical fitness and medical examination. This is in contrast with the proposed requirements associated with

specifically licensed reactor operators and senior reactor operators, as well as the existing requirements for reactor operators and senior reactor operators under part 55.

The NRC is soliciting public feedback regarding whether generally licensed reactor operators should be subject to medical fitness and/or medical examination requirements like reactor operators and senior reactor operators. Please provide your considerations and rationale for your answer.

4. *Onshift Engineering Expertise*: The NRC is proposing to require that timely engineering expertise be accounted for within facility staffing plans. This proposed requirement would be in lieu of the traditional position of the Shift Technical Advisor. The NRC is further proposing that individuals providing such engineering expertise would need to, in part, possess either a qualifying 4-year degree or licensure as a Professional Engineer.

The NRC is interested in feedback from the public regarding the appropriateness of this requirement, including any alternatives that should be considered. Please provide your considerations and rationale for your answer.

5. *Training Program Accreditation*: The NRC is proposing requirements for commission-approved training programs. The NRC does not intend, however, for these requirements to preclude the possibility of programmatic accreditation as an acceptable means of demonstrating compliance with the requirements.

The NRC is soliciting public feedback regarding the role of training program accreditation at part 53 facilities.

6. *Use of Simulation Facilities as Human Factors Engineering Testbeds*: The NRC is proposing to establish regulations pertaining to the use of simulation facilities within the context of the licensing programs both for specifically licensed reactor

operators and senior reactor operators as well as for generally licensed reactor operators. However, these regulations, as currently proposed, do not address the use of simulation facilities within the context of serving as testbeds for human factors engineering-related analyses and assessments. Rather, the NRC staff currently envisions that the use of simulation facilities as human factors engineering testbeds is more appropriately addressed via guidance documents.

The NRC is soliciting public feedback regarding whether simulation facility requirements should also address the use of simulation facilities as human factors engineering testbeds. Please provide your considerations and rationale for your answer.

Part 53, Subpart F – Facility Safety Programs

The NRC is proposing to include in Framework A a requirement for holders of OLs or COLs to develop, implement, and maintain a FSP. The FSP concept is being proposed, in part, to address the advantages of periodically assessing possible risk reduction measures given the proposed role of PRA in the licensing process under Framework A and the resultant need to routinely update the PRA.

The NRC is soliciting public feedback on whether the FSP concept could contribute to improving the NRC's overall regulatory program that includes licensing, various programmatic requirements, NRC inspections, NRC hazard assessment and generic safety programs, NRC backfit assessments, and NRC annual fees? If so, should the NRC develop and include similar provisions in Framework B?

Part 53, Subpart F – Integrity Assessment Program Requirements

Decades of operating experience with light-water-reactors suggests that phenomena such as environmentally assisted fatigue and chemical interactions could impact certain SSCs early in the life of a commercial nuclear plant. Under the existing regulatory framework, these phenomena are generally not addressed until significant safety issues arise (e.g., see numerous generic letters, bulletins, orders, and development and implementation of vessel integrity and materials reliability programs) or a licensee voluntarily pursues renewal of an OL under part 54. The NRC is proposing to include a new set of programmatic requirements for an Integrity Assessment Program that would ensure these phenomena are addressed early in the life of a commercial nuclear plant licensed under part 53. The requirements would be provided in §§ 53.870 and 53.4400 under Framework A and Framework B, respectively.

The NRC is seeking comment on whether the proposed requirements under the Integrity Assessment Program appropriately complement design requirements to address concerns regarding aging, cyclic or transient load limits, and degradation mechanisms related to chemical interactions, operating temperatures, effects of irradiation, and other environmental factors. In addition, the NRC is interested in views on whether, and if so how, degradation mechanisms are or could be addressed in other programs.

Part 53, Subparts G and Q – Decommissioning

On March 3, 2022, the NRC published the proposed rule entitled "Regulatory Improvements for Production and Utilization Facilities Transitioning to Decommissioning" (87 FR 12254). This rulemaking would amend the NRC's current regulations to provide an appropriate regulatory framework for nuclear power reactors transitioning from operations to decommissioning. The rulemaking would address lessons learned from licensees that have completed or are currently in the decommissioning process.

What aspects of this proposed rule, if any, should be incorporated in a part 53 final rule and why?

Part 53, Subparts H and I – Probabilistic Risk Assessment Information

Proposed § 53.1239(a)(18) in subpart H and the related references to this proposed requirement for the holders of OLs and COLs would require a description of the PRA required by § 53.450(a), and its results to be included in FSARs. However, guidance documents may further clarify the division of PRA-related information needed to be in the FSAR, in other possible licensing basis documents, and controlled as plant records subject to inspections and audits.

The NRC is seeking comment on the appropriate placement of PRA-related information between various licensing basis documents and plant records. In addition to the placement of PRA-related information, the NRC is seeking comment on the appropriate control of that information and on the routine submittal of updates to the NRC.

Part 53, Subparts H and I – Changes to Manufacturing Licenses

Proposed §§ 53.1530 and 53.6030 would not allow the holder of a ML or the holder of a COL using a manufactured reactor to make changes to the design of the manufactured reactor or manufactured reactor module without requesting a license amendment from the NRC. The proposed requirements do not include a specific mention of the manufacturing processes for which the NRC could possibly provide finality under proposed §§ 53.1288 and 53.4888.

The NRC is seeking comment on the appropriate change control provisions for MLs, including whether criteria could be developed to determine when a license amendment request would not be required and whether those criteria should address changes in manufacturing processes as well as changes in the design. Please provide your considerations and rationale for your recommendation.

Part 53, Subpart P – Specific Requirements for Technical Specifications

The NRC is proposing to include requirements for TS in Framework B that are largely equivalent to those under § 50.36, with some minor exceptions. One exception relates to the third criterion used for determining whether a Limiting Condition for Operation (LCO) needs to be established for a particular item (e.g., equipment). Specifically, the NRC proposes to add the phrase “or acts as a precursor to identify an issue that would affect” to the third criterion under § 53.4213(b)(2)(ii)(C). This criterion is used to establish LCOs for structures, systems, or components that are used to mitigate accidents and transients such that their failure could ultimately impact the integrity of a fission product barrier. The additional phrase would account for the potential use of functional containment SSCs that may not be part of the primary success path or actuate to mitigate a DBA for non-LWRs that may be licensed under Framework B. This phrase does not currently appear in the existing requirements under § 50.36(c)(2)(ii)(C) but is necessary to account for the variety of functional containment concepts that may exist in a commercial nuclear plant that would be licensed under Framework B.

The NRC is seeking comment on whether the proposed language in § 53.4213(b)(2)(ii)(C) effectively addresses the need for an LCO if an item is credited as part of a functional containment. Please provide your considerations and rationale for your recommendation.

Part 53, Subpart R – Alternative Evaluation for Risk Insights

1. The NRC is proposing to include a requirement in Framework B for identified applications for licenses, certifications, or approvals to include a description of risk evaluations based on either a PRA or an AERI. The AERI approach would use the analyses of postulated bounding events as surrogates for the more detailed PRA analyses and related comparisons of the estimated risks to the NRC's safety goals. The use of the AERI approach would be limited to those applicants showing that the offsite consequences from bounding events were less than criteria in proposed § 53.4730(a)(34)(ii).

The NRC is seeking comment on whether the NRC should retain this AERI approach under Framework B. If so, what changes, if any, would be recommended to the proposed criteria and approach in proposed Framework B? Please provide the considerations and rationale for your answer.

Could the AERI criteria as written or potentially as revised and the related analyses of bounding events be used to support other regulatory decisions in Framework B (e.g., physical security, cybersecurity, AA, FFD and emergency preparedness)? If so, which design areas and programs could logically use the AERI criteria and related analyses and how could requirements in those areas be scaled or graded based on the proposed § 53.4730(a)(34)(ii) or a similar concept?

2. Proposed § 53.4730(a)(34)(ii) would establish the AERI criteria and would support decisions on allowing alternatives to PRAs and on allowing use of generally licensed reactor operators (GLROs) under proposed § 53.800. The proposed criteria involve demonstrating the hypothetical lifetime dose from a bounding event is less than stated values without reliance on protective actions, active safety features, passive safety features that require actuation, or operator actions. Applicants would need to show that passive safety functions would be expected to survive accident conditions and that the passive SSCs cannot be made unavailable or otherwise defeated by credible human errors of commission and omission.

The NRC is seeking comment on the criteria and how they are used in both justifying an alternative to PRAs and in allowing the use of GLROs, as well as possible alternatives to the proposed criteria. Please provide your considerations and rationale for your recommendation.

Part 53, Subpart T – Reporting

Proposed § 53.6340(a)(2)(iv) would require that licensees report any event or condition that results in manual or automatic actuation of a safety-related system. A similar proposed requirement in Framework A is provided by § 53.1640(a)(2)(iv) and includes inadvertent operation of any SSC classified as safety related for an identified safety function under § 53.460 or the unplanned sole reliance on a safety-related system for those systems that are in constant operation.

The NRC is seeking comment on these proposed reporting requirements and how best to address possible actuation or use of safety systems for reactor designs that differ significantly from the LWRs, for which the reporting requirements were developed.

Financial Qualifications

Utility new reactor applicants are exempt under § 50.33(f) from financial qualification reviews because they are generically presumed to be financially qualified for construction and operations. In contrast, merchant power plant new reactor applicants are required under § 50.33(f)(2) to submit information that demonstrates they possess or have reasonable assurance of obtaining the funds necessary to cover estimated construction and operating costs for the period of the license. A "merchant power plant new reactor applicant" is a non -rate-regulated entity (e.g., a nonutility) that engages in the business of production, manufacturing, generating, buying, aggregating, marketing, or brokering electricity for sale at wholesale or for retail sale to the public. Over the past decade, the agency has heard some concerns about the challenges that merchant power plant applicants face in demonstrating compliance with the current financial qualification requirements. Please provide a detailed explanation of challenges, if any, posed by the standard.

Should parts 50 and 52 have the same financial qualification requirements as part 53? Why or why not?

Are there categories of merchant new reactor applicants for which a part 70 "appears to be financially qualified" standard would be more appropriate?[5] If so, please explain what types of applicants should be able to use the part 70 financial qualification standard and what distinguishes these applicants from ones that should not be able to use this standard.

[5] Section 70.23(a)(5).

If a part 70 financial qualification standard were to apply to a category of merchant new reactor applicants, should it also apply to pre-construction license transfer applications for these reactors? Why or why not?

Is there another standard the agency should consider for financial qualification of merchant new reactor applicants? Commenters are encouraged to provide specific suggestions and the basis for those suggestions.

Part 73, Section 73.100 – Physical Security

Part 73 placeholder.