NRC Staff Draft White Paper: Demonstrating the Acceptability of Probabilistic Risk Assessment Results Used to Support Advanced Non-Light Water Reactor Plant Licensing

A. INTRODUCTION

Purpose

This draft white paper provides current Nuclear Regulatory Commission (NRC) staff views and perspectives on demonstrating the acceptability of probabilistic risk assessment (PRA) results used to support advanced non-light water reactor (ANLWR) licensing. The NRC recognizes that some ANLWR applicants may utilize the guidance provided in NEI 18-04 (termed the "LMP guidance" in this draft white paper[1]), "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development" (Ref. 1) as endorsed in 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" (Ref. 2). The LMP guidance enhances the role that PRA results play in establishing the safety case and, therefore, imposes additional burden on demonstrating the acceptability of PRA results as discussed in this draft white paper.

The NRC staff is currently developing a draft regulatory guide for comment on the acceptability of PRA results used to support ANLWR licensing. Based on the staff's current understanding of ASME/ANS RA-S-1.4-2021 (termed the "ANLWR PRA standard" in this white paper[2]), "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants" (Ref. 3) and NEI 20-09, "Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard" (Ref. 4), the staff plans to endorse them in the draft regulatory guide. Depending on the content of the final published versions of the ANLWR PRA standard and NEI 20-09, the NRC staff may propose clarifications, qualifications, exceptions, or additions to these documents in any proposed endorsement. The NRC staff anticipates the issuance of this regulatory guide in calendar year 2021.

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[1] NEI 18-04 is commonly called the "LMP guidance" because it was developed as part of the Licensing Modernization Project that is being conducted by the Southern Nuclear Company, sponsored by the Nuclear Energy Institute, and cost-shared by the U.S. Department of Energy.

[2] This draft white paper is based on a non-publicly available pre-publication version of ASME/ANS RA-S-1.4-2021 which was provided to the NRC by the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) under the auspices of the Joint Committee on Nuclear Risk Management (JCNRM). The NRC is a voting member of the JCNRM and actively participated in the development of ASME/ANS RA-S-1.4-2021, which is expected to be published in February 2021.

This draft white paper has been developed to facilitate discussion among stakeholders on the ANLWR PRA standard and NEI 20-09, and to inform the NRC staff approach to endorsing these guidance documents while the regulatory guide is being developed. While an applicant is free to employ the final version of ASME/ANS RA-S-1.4-2021 and NEI 20-09 in developing a PRA, the applicant should justify the use of those guidance documents in any application filed before the NRC staff endorses them, so that the NRC staff can evaluate whether a PRA developed in accordance with the guidance is acceptable for the particular application.

Applicability

In this draft white paper, the term "non-light water reactor" means a nuclear fission reactor that does not use light water as a reactor coolant[3]. Examples of ANLWRs as addressed by this draft white paper include, but are not limited to, high-temperature gas-cooled reactors, liquid metal-cooled fast reactors, molten salt reactors, and reactors that are cooled by sodium heat pipes.

This draft white paper applies to applications for ANLWR licensing under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 5), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 6). Specifically:

- Construction Permit (CP): 10 CFR Part 50
- Operating License (OL): 10 CFR Part 50
- Standard Design Certification (DC): 10 CFR Part 52, Subpart B
- Combined License (COL): 10 CFR Part 52, Subpart C
- Standard Design Approval (SDA): 10 CFR Part 52, Subpart E
- Manufacturing License (ML): 10 CFR Part 52, Subpart F

The NRC staff notes that current regulations do not require applicants for Part 50 construction permits or operating licenses to provide PRA-related information; however:

- The Commission's severe accident policy statement (Ref. 7) articulates the Commission's determination that all new nuclear power plant designs can be shown to be acceptable for severe accident concerns, in part, by completing a PRA and considering the severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety.

[3]The NRC staff understands that the term "non-light water reactor" could be interpreted to include reactors that use heavy water as a neutron moderator. However, the definition of ANLWR used in this draft white paper is consistent with the NRC's and U.S. nuclear industry's general usage. To date, no designer has notified the NRC that it intends to license a heavy water reactor in the U.S.

- The Commission's advanced reactor policy statement (Ref. 8) articulates the Commission's expectation that all new nuclear power plant designs will meet the Commission's safety goals (Ref. 9).

- There is an ongoing rulemaking effort, "Incorporation of Lessons Learned from New Reactor Licensing Process (Parts 50 and 52 Licensing Process Alignment)," Docket NRC-2009-0196, RIN-3150-AI66[4] that is expected to add PRA-related requirements for Part 50 construction permit and operating license applications that will be similar to the existing requirements for Part 52 licenses, certifications, and approvals.

On January 14, 2019, the President signed the Nuclear Energy Innovation and Modernization Act (NEIMA) into law (Ref. 10). Consistent with Section 103 of NEIMA, the NRC staff has begun efforts to establish a "Risk Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors," Docket NRC-2019-0062, RIN 3150-AK31[5] for optional use by applicants for new commercial advanced nuclear reactor licenses by December 31, 2027. Specifically, this rulemaking activity will create a new 10 CFR Part 53, which is tentatively titled "Licensing and Regulation of Advanced Nuclear Reactors." The NRC staff notes that the phrase "risk-informed, technology-inclusive" implies that the acceptability of PRA results will be an important issue for all reactors (specifically including ANLWRs) licensed under 10 CFR Part 53.

The NRC staff views and perspectives provided in this draft white paper apply to stationary ANLWRs, which include (1) reactors that may be constructed at a site, and (2) reactors that are constructed at an offsite facility and subsequently transported and installed at a site. This draft white paper does not address PRAs used to assess the risk of transporting ANLWRs from an offsite facility to the site. Moreover, this draft white paper does not address mobile reactors, which may be relocated to different sites after initial criticality.

The staff notes that (1) the regulations in 10 CFR Part 52 requiring DC, SDA, ML, and COL applicants to provide a description of their PRAs and its results; (2) the regulations in 10 CFR Part 50 requiring COL holders to maintain and upgrade their PRAs; (3) the Commission's severe accident policy statement; and (4) the Commission's advanced reactor policy statement apply to all commercial nuclear power plants, regardless of their design or thermal power. However, in keeping with the philosophy of risk-informed decision-making, the staff recognizes that applicants may desire to tailor the PRA's scope and level of detail commensurate with the role that the PRA results play in establishing the safety case. The staff is considering what guidance to provide on rightsizing PRAs used to support ANLWR licensing. Applicants are encouraged to discuss the scope and level of detail that will be provided in their PRAs during preapplication interactions with the NRC staff.

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[4] Further information about this rulemaking (including the proposed schedule) is provided at https://www.nrc.gov/reading-rm/doc-collections/rulemaking-ruleforum/active/ruledetails.html?id=27

[5] Further information about this rulemaking (including the proposed schedule) is provided at https://www.nrc.gov/reading-rm/doc-collections/rulemaking-ruleforum/active/ruledetails.html?id=1108

Applicable Regulations

The following ~~are~~ regulations ~~that~~ are directly applicable to the use of PRA in licensing activities for ANLWRs. Other regulations may be more broadly applicable to ANLWR licensing activities~~,~~ but have not been included here for brevity.

- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The regulations in this part provide for the licensing of production and utilization facilities pursuant to the Atomic Energy Act of 1954, as amended and Title II of the Energy Reorganization Act of 1974.
    - 10 CFR 50.71(h)(1), requires that (1) each holder of a combined license under subpart C of 10 CFR Part 52 shall develop a Level 1 and a Level 2 PRA, and (2) the PRA must cover those initiating events and modes for which NRC-endorsed consensus standards on PRA exist one year prior to the scheduled date for initial loading of fuel.
    - 10 CFR 50.71(h)(2), requires that (1) each holder of a combined license shall maintain and upgrade the PRA required by 10 CFR 50.71(h)(1), (2) the upgraded PRA must cover initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect one year prior to each required upgrade, and (3) the PRA must be upgraded every four years until the permanent cessation of operations under 10 CFR 52.110(a).
    - 10 CFR 50.71(h)(3), requires that each holder of a combined license shall, no later than the date on which the licensee submits an application for a renewed license, upgrade the PRA required by 10 CFR 50.71(h)(1) to cover all modes and all initiating events.
- 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The regulations in this part govern the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities pursuant to the Atomic Energy Act of 1954, as amended and Title II of the Energy Reorganization Act of 1974.
    - 10 CFR 52.47(a)(27), requires that applicants for a standard design certification under Subpart B of 10 CFR Part 52 shall provide a description of the design-specific probabilistic risk assessment (PRA) and its results.
    - 10 CFR 52.79(a)(46), requires that applicants for a combined license under Subpart C of 10 CFR Part 52 shall provide a description of the plant-specific probabilistic risk assessment (PRA) and its results.
    - 10 CFR 52.79(c)(1), requires that applicants for a combined license under Subpart C of 10 CFR Part 52 who reference a standard design approval under Subpart E of 10 CFR Part 52 shall use and update the PRA information for the standard design approval to account for site-specific design information and any design changes or departures.
    - 10 CFR 52.79(d)(1), requires that applicants for a combined license under Subpart C of 10 CFR 52 who reference a standard design certification under Subpart B shall use and update

the PRA information for the standard design certification to account for site-specific design information and any design changes or departures.

- o 10 CFR 52.79(e)(1), requires that applicants for a combined license under Subpart C of 10 CFR Part 52 who reference the use of one or more manufactured nuclear power reactors licensed under Subpart F of 10 CFR Part 52 shall use and update the PRA information for the manufactured reactor to account for site-specific design information and any design changes or departures.

- o 10 CFR 52.137(a)(25) requires that applicants for a standard design approval under Subpart E of 10 CFR Part 52 shall include a description of the design-specific probabilistic risk assessment and its results.

- o 10 CFR 52.157(a)(31) requires that applicants for a manufacturing license under Subpart F of 10 CFR Part 52 shall include a description of the design-specific probabilistic risk assessment and its results.

Related Guidance

The NRC staff has reviewed the following sources while formulating the views and perspectives presented in this draft white paper. Many of these sources are specific to light-water reactors (LWRs); however, the NRC staff believes that the high-level concepts and principles that have been developed over the past 25 years in the context of LWR PRAs are also applicable to ANLWR PRAs. The citation of these sources in this draft white paper does not necessarily imply that these sources may be used as-is to support the demonstration of ANLWR PRA acceptability or ANLWR licensing in general.

- RG 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Ref. 11), describes one approach acceptable to the NRC staff for determining whether a base PRA, in total or the portions that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for LWRs.

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," provides guidance to the NRC staff in performing safety reviews of construction permit or operating license applications (including requests for amendments) under 10 CFR Part 50 and early site permit, standard design certification, combined license, standard design approval, or manufacturing license applications under 10 CFR Part 52 (including requests for amendments). NUREG-0800, Section 19.1, titled "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Ref. 12) guides the NRC staff in its evaluations of licensee requests for changes to the licensing basis that apply risk insights. Guidance developed in selected application-specific regulatory guides (RGs) and the corresponding chapters of NUREG-0800 also applies to these types of licensing basis changes. Furthermore, NUREG-0800, Section 19.0, titled "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors" (Ref. 13), pertains to the NRC staff review of the design-specific PRA for a DC and plant-specific PRA for a COL application, respectively.

- NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking" (Ref. 14), provides guidance on how to treat uncertainties associated with PRA in risk-informed decision-making. This guidance is intended to foster an understanding of the uncertainties associated with PRA and their impact on the results of PRA.

- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Ref. 15), provides guidance on an acceptable approach for developing risk-informed applications for a licensing basis change that considers engineering issues and applies risk insights.

- RG 1.206, "Applications for Nuclear Power Plants" (Ref. 16), provides guidance on the format and content of applications for nuclear power plants submitted to the NRC under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," which specifies the information to be included in an application. This RG applies to power reactors with LWR technology. The NRC staff also considers this RG to generally apply to other types of power reactors (e.g., ANLWRs). The NRC staff considers this guidance acceptable to support preparation of applications for early site permits, standard design certifications, and combined licenses under 10 CFR Part 52 and generally acceptable to support its review of other types of applications under 10 CFR Part 52.

- DC/COL-ISG-028, "Interim Staff Guidance on Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application" (Ref. 17), explains how to adapt the requirements[6] provided in the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) PRA standard ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Ref. 18), which was developed for currently operating reactors, to PRAs that support advanced LWR standard design certification and combined license applications.

- RG 1.233 provides guidance to inform the licensing basis and content of applications for ANLWRs, including, but not limited to, molten salt reactors, high-temperature gas-cooled reactors, and a variety of fast reactors of different thermal capacities. This guidance may be used by ANLWR applicants applying for permits, licenses, certifications, and approvals under 10 CFR Part 50 and 10 CFR Part 52. RG 1.233 endorses the licensing modernization project (LMP) guidance provided in NEI 18-04.

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[6] A national consensus PRA standard, as endorsed by the NRC staff with clarifications, qualifications, exceptions, or additions, provides one set of standards the NRC staff considers acceptable for developing an adequate base PRA. Because these PRA standards use the terms "requirement", "require," and other similar mandatory language, the NRC staff's endorsement, including NRC staff clarifications, qualifications, exceptions, and deletions, will mirror this language. However, the use of this language in this draft white paper is not intended to convey a regulatory requirement or suggest that these standards are the only way to meet the statutory and regulatory requirements. The staff will identify NRC regulatory requirements as such in this paper.

B. DISCUSSION

Background

In this draft white paper, the definition of the term "probabilistic risk assessment (PRA)" is consistent with Regulatory Position C.1 of RG 1.200. Specifically, for a method or approach to be considered a PRA, the method or approach (1) provides a quantitative assessment of the identified risk in terms of scenarios that result in undesired consequences (e.g., radioactive release) and their frequencies, and (2) is comprised of specific technical elements in performing the quantification.

The PRA is used to support ANLWR licensing in the following areas:

- As discussed in the Commission's severe accident policy statement, the PRA is used to identify severe accident vulnerabilities and to provide insights which support the conclusion that the ANLWR plant design, construction, and operation provides reasonable assurance no undue risk to public health and safety.
- As discussed in the Commission's advanced reactor policy statement, the PRA can be used to demonstrate that the ANLWR plant meets the Commission's safety goals.
- The PRA is used to support the environmental review required by 10 CFR Part 51, specifically, the evaluation of severe accident mitigation design alternatives (SAMDAs).
- For ANLWR applications that are based on the LMP guidance, the PRA is used to select licensing basis events, classify systems, structures and components (SSCs), and to inform the defense-in-depth evaluation.
- For ANLWR applications that are not based on the LMP guidance, the PRA may be used to support the process used to demonstrate whether the regulatory treatment of non-safety systems (RTNSS) is sufficient and, if appropriate, identify the systems, structures, and components (SSCs) included in RTNSS.
- The results and insights of the PRA are used to support the regulatory oversight process.
- The results and insights of the PRA are used to identify and support the development of specifications and performance objectives for the plant design, construction, inspection, and operation, such as: inspections, test, analyses, and acceptance criteria (ITAAC); technical specifications; and COL action items and interface requirements.
- The PRA may be used to support various voluntary risk-informed applications (e.g., risk-informed inservice inspection) that may be included in the ANLWR licensing application.

It is essential that ANLWR applicants demonstrate the acceptability of PRA results used to support ANLWR licensing. Consistent with RG 1.200, the NRC staff assesses acceptability of PRA results with respect to the scope, level of detail, conformance with PRA technical elements (i.e., technical adequacy), and plant representation of a PRA as related to the outcome of the NRC staff's review of a given ANLWR licensing application.~~:~~

- Scope of a PRA: The scope of a PRA is defined in terms of (1) the metrics used to characterize risk, (2) the plant operating states (POSs) for which the risk is to be evaluated, and (3) the causes of initiating events (hazard groups) that can potentially challenge and disrupt the normal operation of the plant and, if not prevented or mitigated, would eventually result in a radioactive release. The scope of a PRA is determined by its intended use for representing the as-built and as-operated plant.

- Level of detail of a PRA: The level of detail of a PRA is defined in terms of the resolution of the modeling used to represent the behavior and operations of the plant. A minimal level of detail is necessary to ensure that the impacts of designed-in dependencies (e.g., support system dependencies, functional dependencies, and dependencies on operator actions) are correctly captured. This minimal level of detail is implicit in the technical elements comprising the PRA and their associated characteristics and attributes.

- Technical elements of a PRA: The PRA technical elements are defined in terms of the fundamental technical analyses needed to develop and quantify the PRA model for its intended purpose (e.g., determination of a specific risk metric). The characteristics and attributes of the PRA technical elements define specific criteria for successfully performing those technical analyses and achieving a defined objective.

- Plant representation and PRA configuration control: Plant representation is defined in terms of how closely the PRA represents the plant as it is designed, built, and operated.

The NRC staff expects that a PRA that supports implementation of the LMP guidance or other voluntary risk-informed applications (such as implementation of 10 CFR 50.69 for applicants that do not use the LMP guidance), will have an increased level of detail and plant representation. In addition, the PRA will evolve as the plant is designed, constructed, and operated, as illustrated in Figure 1. Accordingly, the demonstration of PRA acceptability needs to evolve in parallel in order to support uses of the PRA.

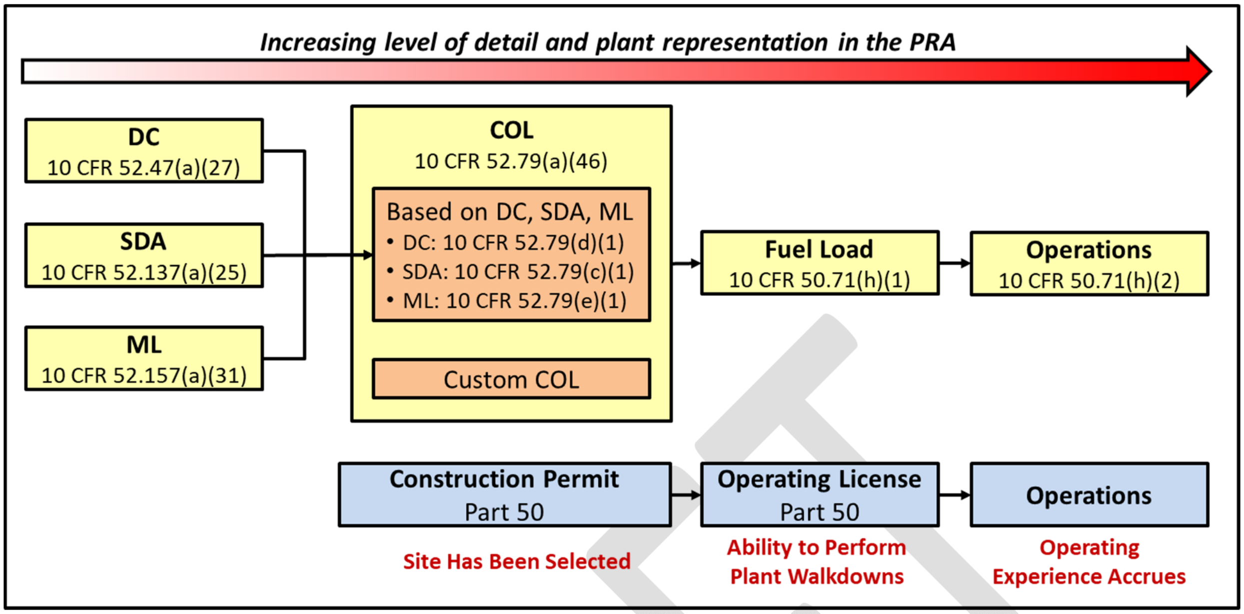

Figure 1. Evolution of the PRA as the Plant is Designed, Constructed, and Operated.

The following sections provide NRC staff views and perspectives on each of these attributes of PRA acceptability in the context of PRAs used to support ANLWR licensing.

Scope of an Acceptable PRA

The NRC staff believes that a PRA that is used to support ANLWR licensing should:

- Address all radiological sources at the plant (e.g., reactor cores, spent fuel, fuel reprocessing facilities for molten salt reactors), including accident scenarios that lead to a radioactive release from multiple radiological sources.
- Address all internal hazards (such as, but not limited, to internal initiating events, internal floods, and internal fires) and all external hazards (such as, but not limited to, seismic events, external floods, and high wind events). Seismic events should always be included; other external hazards should also be included if they cannot be screened out with appropriate justification. With the exception of seismic events, external hazards may be screened from further consideration in the PRA with appropriate justification.
- Address all plant operating states (e.g., at-power, low-power, shutdown).
- Develop the frequencies of accident scenarios from the occurrence of an initiating event until the release of radioactive materials to the environment and estimate the consequences that result from the release (i.e., an ANLWR PRA should be a Level 3 PRA).

This recommended PRA scope is consistent with the expected scope of ASME/ANS RA-S-1.4-2021; the LMP guidance (NEI 18-04); NUREG-2150, "A Proposed Risk Management Regulatory Structure" (Ref. 19); and NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing" (Ref. 20). The NRC staff acknowledges that the above recommended scope of a PRA used to support ANLWR licensing exceeds the traditional scope of a PRA used to support LWR licensing, which typically only estimates core-damage frequency (CDF) and large release frequency (LRF). There are several important issues that should be considered when contemplating the development of a PRA that uses risk surrogates such as CDF and LRF to support ANLWR licensing:

1. The NRC staff anticipates that Section 1.9.1 of ASME/ANS RA-S-1.4-2021 will discuss the development of PRAs for ANLWRs that use CDF and LRF as follows:

   > The LWR risk metric core damage frequency (CDF) is not used because it is not applicable to non-LWRs as defined in LWR PRA standards. Large release frequency (LRF) and large early release frequency (LERF) are not used because they are not necessary in this Standard given the requirement to quantify source terms and radiological consequences. User defined intermediate metrics may be used if justified by the user.

2. LRF was originally proposed by the Commission in its safety goal policy statement as a general performance guideline. In SRM-SECY-89-102 (Ref. 21), the Commission emphasized that:

   > The Commission believes that the safety goal objectives should be applied to all designs, independent of the size of containment or character of a particular design approach to the release mitigation function. Accordingly, for the purpose of implementation, the staff may establish subsidiary quantitative core damage frequency and containment performance objectives through partitioning of the Large Release Guideline.

3. In SRM-SECY-90-016 (Ref. 22), the Commission approved the use of LRF as a risk surrogate for evaluating evolutionary LWRs. The NRC has not issued a formal definition of a "large release" for LWRs or ANLWRs. The Joint Committee on Nuclear Risk Management (JCNRM), which is composed of ASME and ANS representatives and is tasked with developing PRA-related industry consensus standards, has developed the following definition of large release:

   > large release: the release of airborne fission products to the environment such that there are significant off-site impacts. Large release and significant off-site impacts may be defined in terms of quantities of environmental fission product releases, status of fission product barriers and scrubbing, or dose levels at specific distances from the release, depending on the specific design objectives and regulatory requirements.

   The NRC staff understands that this definition is expected to be incorporated in the next edition of the L1/LERF PRA standard (currently anticipated by mid-2021) as part of a nonmandatory appendix that addresses advanced LWRs.

4. In SRM-SECY-90-016, the Commission approved the use of conditional containment failure probability (CCFP) as a probabilistic containment performance goal for evolutionary LWRs. In SECY-18-0096, "Functional Containment Performance Criteria for Non-Light Water Reactors" (Ref. 23), the staff proposed a methodology for establishing functional containment performance criteria for ANLWRs which does not use the concept of CCFP. In SRM-SECY-18-0096, "Staff Requirements – SECY-18-0096 – Functional Containment Performance Criteria for Non-Light

Water Reactors" (Ref. 24), the Commission approved the NRC staff's proposed methodology for establishing functional containment performance criteria for ANLWRs.

5. In SRM-SECY-93-087 (Ref. 25), the Commission approved the use of margin-type assessments of seismic events for evolutionary and advanced LWRs. However, ASME/ANS RA-S-1.4-2021 is not expected to provide high-level requirements or supporting requirements for conducting seismic margin analyses; rather, it requires the development of a seismic PRA.

The NRC staff recognizes that ASME/ANS RA-S-1.4-2021 is not expected to provide high-level requirements and supporting requirements for internal fire PRAs during low-power and shutdown (LPSD) plant operating states (POSs). However, the NRC staff believes that a complete set of PRA results and insights (specifically including results and insights from internal fire PRAs for LPSD POSs) should be developed as early as possible to support risk-informed decisionmaking related to design, licensing, procurement, construction, operation, and maintenance of ANLWRs. Moreover, in November 2019, the National Fire Protection Association (NFPA) issued a revision to NFPA 806, "Performance-Based Standard for Fire Protection for Advanced Nuclear Reactor Electric Generating Plants Change Process" (Ref. 26). In its discussion of the origin and development of NFPA 806, the NFPA states:

> The need for fire protection in nuclear power facilities has been demonstrated in a number of incidents, including the Browns Ferry Fire in 1975 and other more recent incidents in the United States and abroad. Probabilistic risk assessments of existing plants have shown that fire is one of the largest single contributors to the possibility of reactor damage. This document represents a comprehensive consensus of baseline fire protection requirements for all aspects of change process for advanced nuclear reactor electric generating plants, including their construction and all phases of operations, such as shutdown, degraded conditions, and decommissioning.

While 10 CFR 50.48(c)(4) allows for a risk-informed or performance-based alternative to compliance with NFPA 805, " Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", ~~for LWRs,~~ the NRC staff has no current plans to endorse NFPA 806. NUREG/CR-7114, "A Framework for Low Power/Shutdown Fire PRA" (Ref. 27), provides an analysis framework for developing a LPSD fire PRA that can be applied to ANLWRs. It should be noted that NUREG/CR-7114, which is based on experience gained in developing at-power fire PRAs for LWRs, does not provide a comprehensive methodology because the approaches described do not yet represent a full, complete, and tested set of analysis tools. As an alternative to performing a LPSD fire PRA, the methods and approaches in NUREG-1855, Section 5, "Stage C – Assessing Completeness Uncertainty," provide acceptable approaches for performing qualitative and quantitative screening analyses (including bounding quantitative analysis; conservative, but not bounding analysis; and realistic, but limited quantitative analysis).

When developing the portions of the PRA that address external hazards (e.g., seismic, external floods, high winds) and radiological consequence assessment, the NRC staff anticipates ASME/ANS RA-S-1.4-2021 to specify the use of a bounding site[7] if the actual site has not been selected. This situation

[7] The ANLWR PRA standard defines a bounding site as "a hypothetical site that is defined to bound the characteristics of a range of sites for use in design of a standard plant. The site characteristics may be selected from site parameters from actual sites. For this bounding site, site-related parameters are defined using a set of

applies to PRAs that support DC, SDA, and ML applications. The NRC staff notes that it may be necessary to use a separate bounding site for each external hazard and for the radiological consequence assessment, i.e., it may not be possible to find a single physical site that bounds all of the external hazards and all of the inputs to the radiological consequence assessment (e.g., meteorological data, demographic data). Applicants should describe and justify the selection of the bounding site(s) used in the PRA. The NRC staff believes that COL applicants who reference a DC, SDA, or ML should upgrade the DC, SDA, or ML PRAs to reflect site-specific external hazards and inputs to the radiological consequence analysis.

Risk Metrics

As previously discussed in the section titled "Scope of an Acceptable PRA," the NRC staff believes that an ANLWR PRA should be a Level 3 PRA. Accordingly, the following risk metrics should be produced:

- Individual prompt (early) fatality risk (per plant-year[8]), for an average individual within 1 mile of the exclusion area boundary (EAB).

- Individual latent cancer fatality risk (per plant-year), for an average individual within 10 miles of the EAB.

- If the LMP guidance (NEI 18-04) is implemented, the total mean frequency at which the 30-day total effective dose equivalent (TEDE) at the EAB exceeds 100 mrem.

- Risk metrics that support the evaluation of SAMDAs, such as population dose risk (person-rem per plant-year) and offsite economic risk ($ per plant-year).

Applicants who elect to develop a smaller scope PRA that uses intermediate risk metrics in lieu of a Level 3 PRA need to define and justify all user-defined intermediate risk metrics (e.g., CDF, LRF), to explain how the user-defined intermediate risk metrics capture the risk from all radiological sources, and to show how the user-defined intermediate risk metrics relate to the quantitative health objectives defined in the Commission's safety goal policy statement.

Use of the ANLWR PRA Standard

As discussed in RG 1.200, national consensus PRA standards provide one set of minimum criteria that can be met, as endorsed by the NRC staff with clarifications, qualifications, exceptions, or additions, for a PRA to be considered acceptable. For PRAs that support ANLWR licensing, the NRC staff currently anticipates that the high-level requirements (HLRs) and supporting requirements (SRs) in ASME/ANS RA-S-1.4-2021 (the ANLWR PRA Standard) will be generally acceptable for development

scenarios that are chosen to provide appropriately high external hazard design parameter values and the most adverse meteorological conditions and population data for assessing off-site radiological impact." It should be noted that a bounding site, as defined by the ANLWR PRA standard, is different that the set of site parameters used in a design certification or early site permit application.

[8] In this draft white paper, the term "plant" means a collection of one or more NLWRs and their supporting facilities (e.g., spent fuel storage, fuel reprocessing facilities for molten salt reactors) that are located at the same site. As defined in ASME/ANS RA-S-1.4-2021, a "plant-year" is a calendar year in the operating life of a plant.

of a PRA. Nonetheless, it should be noted that the NRC may develop clarifications, qualifications, exceptions, or additions to these requirements when it develops its staff position in the draft regulatory guide on whether to endorse ASME/ANS RA-S-1.4-2021 and NEI 20-09.

As previously discussed in the section titled "Background" and illustrated in Figure 1, the PRA evolves as the plant is designed, constructed, and operated. For ANLWRs that are licensed under 10 CFR Part 50, there are three plant licensing stages as described below:

1. CP application stage: The site has been selected; the PRA represents the conceptual design.

2. OL application stage: The plant has been constructed and is ready to begin pre-operational testing; the PRA generally represents the as-built and as-to-be-operated plant but does not reflect plant-specific operating experience.

3. Commercial operations stage: The plant accrues operating experience; the PRA reflects the as-built, as-operated plant including plant-specific operating experience.

For ANLWRs licensed under 10 CFR Part 52, there are four potential plant licensing stage as described below:

1. DC, SDA, ML application stage: The site has not been selected; the PRA is based on postulated site parameters and represents an essentially complete design (for a DC or ML application) or the final design of an entire facility or major portions thereof (for an SDA application).

2. COL application stage: The site has been selected; the PRA generally represents the as-to-be-built plant (i.e., the plant as described in the final safety analysis report) but does not reflect plant-specific operating experience.

3. Fuel-load stage: The plant has been constructed and ready to begin pre-operational testing; the PRA generally represents the as-built and as-to-be-operated plant but does not reflect plant-specific operating experience.

4. Commercial operation stage: The plant accrues operating experience; the PRA reflects the as-built, as-operated plant including plant-specific operating experience.

The PRA standards that apply to LWRs (e.g., ASME/ANS RA-Sa-2009) were developed to support the currently operating fleet of commercial nuclear power plants and, accordingly, only address the commercial operation stage discussed above. The NRC staff developed DC/COL-ISG-028 to address the DC/SDA/ML and COL stages for LWRs licensed under Part 52. In contrast, ASME/ANS RA-S-1.4-2021 was developed to support ANLWR plant design, construction, and operation and, accordingly, addresses all licensing stages. The NRC staff notes that some of the SRs expected to be provided in ASME/ANS RA-S-1.4-2021 only apply to certain plant licensing stages, as illustrated by the examples provided in Table 1 (the examples are not exhaustive).

Table 1. Examples of Supporting Requirements in ASME/ANS That Only Apply to Certain Plant Licensing Stages.

No.	SR	Capability Category[9] I (CC-I)	Capability Category II (CC-II)	Remarks
1	POS-A1	IDENTIFY a representative set of plant evolutions to be analyzed. INCLUDE, at a minimum, plant evolutions from at-power operations. See Note POS-N-1, POS-N-2, POS-N-3, POS-N-4	IDENTIFY a representative set of plant evolutions to be analyzed, including refueling outages, other controlled shutdowns and forced outages See Note POS-N-3	Applies to all licensing stages.
2	POS-A5	For PRAs performed during the pre-operational stage, ENSURE the level of detail in delineating the POSs is consistent with the level of detail of the design information available to support, and referenced by the PRA sufficient to identify potential risk-significant contributors. See Note POS-N-13		Applies during plant design and construction; replaced by POS-A4 after construction has been completed.
3	POS-A4	For operating plants, ENSURE the level of detail in the delineation of POSs is consistent with the as-built and as-operated plant sufficient to identify potential risk-significant contributors. See Note POS-N-12		Applies after design and construction; replaces POS-A5.
4	POS-C2	For PRAs performed during the pre-operational stage, PROVIDE the basis for the assumed mean duration and time in the plant operating cycle of each modeled POS. See Note POS-N-13		Applies before operating experience is accrued; replaced by POS-C1 after the start of commercial operation.
5	POS-C1	Within the selected plant evolutions, CALCULATE the mean duration and the mean time after shutdown for each POS based on a review of applicable plant- or design-specific record. See Note POS-N-7, POS-N-8, POS-N-20		Applies after operating experience accrues; replaces POS-C2.
6	SHA-A1	For the seismic hazards analysis, either: (a) IDENTIFY the site at which the reactor being analyzed is located or (b) DESCRIBE a bounding site and JUSTIFY that the bounding site bounds the list of sites in the scope of the PRA. See Note S-N-1		Part (b) applies prior to site selection; Part (a) applies after site selection.

[9] In the ANLWR PRA standard, Capability Category (CC) is used to differentiate SRs according to scope and level of detail, plant specificity, and realism. Some SRs apply to only one CC and some extend across both CCs. When a SR spans both CCs, it applies equally to each CC.

Table 1. Examples of Supporting Requirements in ASME/ANS That Only Apply to Certain Plant Licensing Stages.				
No.	**SR**	**Capability Category[9] I (CC-I)**	**Capability Category II (CC-II)**	**Remarks**
7	WFR-A3	For PRAs conducted on a specific site, ENSURE that the wind fragilities are site-specific. See Note W-N-37		Only applies after site selection.
8	RCAD-A8	For PRAs performed on a bounding site, IDENTIFY assumptions made due to the lack of site details that influence the atmospheric transport and dispersion conditions. See Note RC-N-5		Only applies prior to site selection.

It is important to determine when a given SR applies to a given licensing stage. The wording of each SR in ASME/ANS RA-S-1.4-2021 may contain qualifiers such as “pre-operational,” “operating plants,” “bounding site,” and “specific site” that indicate how the SR relates to the plant licensing stages. If no qualifier is provided, then the SR applies to all plant licensing stages. It should be noted that some SRs contain the “operating plants” qualifier but pertain to the use of plant-specific operating experience, which does not accrue until the plant enters commercial operation. Table 2 provides a guide for determining the applicability of SRs to various plant licensing stages based on the qualifiers used in the wording of the SRs.

Table 2. Guide for Determining the Applicability of Supporting Requirements for Various Plant Licensing Stages.						
		Plant Licensing Stage				
SR Type	**Qualifier**	**DC, SDA, or ML Application**	**COL or CP Application**	**COL Holder Fuel-Load PRA or OL Application**	**Commercial Operations**	**Example SRs**
1	<none>	yes	yes	yes	yes	POS-A1
2	Contains the phrase “pre-operational stage” and does not involve operating experience	yes	yes	no	no	POS-A5
3	Contains the phrase “operating plants” and does not involve operating experience	no	no	yes	yes	POS-A4
4	Contains the phrase “pre-operational stage” and uses generic data or requires assumptions	yes	yes	yes	no	POS-C2
5	Contains the phrase “operating plants” and uses plant-specific operating experience	no	no	no	yes	POS-C1
6	Two-part SR that contains the phrases “bounding site” and “site-specific”	yes(BS)[a]	yes(SS)[b]	yes(SS)[b]	yes(SS)[b]	SHA-A1

Table 2. Guide for Determining the Applicability of Supporting Requirements for Various Plant Licensing Stages.						
		Plant Licensing Stage				
SR Type	**Qualifier**	**DC, SDA, or ML Application**	**COL or CP Application**	**COL Holder Fuel-Load PRA or OL Application**	**Commercial Operations**	**Example SRs**
7	Contains the phrase "For PRAs conducted on a specific site"	no	yes	Yes	yes	WFR-A3
8	Contains the phrase "For PRAs performed on a bounding site"	yes	no	no	no	RCAD-A8
[a]The portion of the SR that pertains to the use of a bounding site applies. [b]The portion of the SR that pertains the use of a specific site applies.						

The observation that some of the SRs anticipated~~expected~~ to be provided in ASME/ANS RA-S-1.4-2021 only apply to certain plant licensing stages has important ramifications with respect to PRA upgrade and peer reviews. ASME/ASNS RA-S-1.4.-2020 provides the following definitions:

- PRA maintenance: a change in the PRA that does not meet the definition of PRA upgrade.

- PRA upgrade: a change in the PRA that results in the applicability of one or more SRs or CCs (e.g., the addition of a new hazard model) that were not previously assessed in a peer review of the PRA, an implementation of a PRA method in a different context, or the incorporation of a method not previously used.

According to these definitions, the PRA is being upgraded as it progresses through the various licensing stages because one or more SRs that were not previously assessed in a peer review have become applicable. Specifically, referring to Table 2:

- Transition from the DC/SDA/ML stage to the COL stage:
  - Type 6 SRs change from a bounding site to a specific site
  - Type 7 SRs become applicable
  - Type 8 SRs become not applicable

- Transition from the COL stage to the fuel-load stage (for plants licensed under Part 52), and transition from the CP stage to the OL stage (for plants licensed under Part 50):
  - Type 3 SRs become applicable
  - Type 2 SRs become not applicable

- Transition from the fuel-load stage (for plants licensed under Part 52) or from the OL stage (for plants licensed under Part 50) to the commercial operation stage:
  - Type 5 SRs become applicable when plant-specific operating experience is first incorporated into the PRA
  - Type 4 SRs become not applicable

Section 6 of ASME/ANS RA-S-1.4-2021 provides requirements for PRA peer reviews. The NRC staff notes that Section 6 allows the use of a focused-scope peer review in lieu of a full-scope peer review "…to address changes to the PRA model as a result of upgrades." Further NRC staff views and perspectives on PRA peer reviews are provided elsewhere in this draft white paper.

After an ANLWR plant enters commercial operation and all relevant SRs have been peer reviewed, the PRA should be maintained by periodically incorporating plant-specific operating experience. For COL holders, 10 CFR 50.71(h)(2) requires periodic PRA maintenance and upgrade on a four-year period.

The NRC staff's perspective on the use of CCs for various plant licensing stages is as follows:

1. Consistent with DC/COL-ISG-028, CC-I is generally acceptable if the ANLWR licensing application is not based on the LMP guidance and does not involve concurrent voluntary risk-informed applications.

2. Consistent with RG 1.174, CC-II is generally expected if the ANLWR licensing application is based on the LMP guidance or involves concurrent voluntary risk-informed applications.

The identification of risk-significant items (e.g., basic events, human failure events, initiating events, event sequences, event sequence families, plant operating states, release categories) is an essential part of PRA. First, risk-significant items are part of the "description of the PRA and its results," and are used to identify severe accident vulnerabilities as specified in the Commission's severe accident policy statement. Second, knowledge of the risk-significant items is used to refine the PRA as it is developed; accordingly, some SRs in ASME/ASN RA-S-1.4-2021 only apply to the risk-significant items.

ASME/ANS RA-S-1.4-2021 is expected to include the option to use either relative or absolute criteria for establishing risk significance depending on the PRA application. The NRC staff agrees that, for the purpose of refining the PRA, either relative or absolute criteria may be used. The NRC staff expects PRA results to include risk-significant items that have been identified by using relative criteria and, if used to develop the PRA, risk-significant items that have been identified by using absolute criteria.

The nonmandatory appendices expected to be provided in ASME/ANS RA-S-1.4-2021 may be binned into two groups: (1) Notes that support the understanding of various SRs, and (2) Commentaries. The NRC staff generally accepts the Notes and has no opinion about the Commentaries. It should be noted that the NRC may develop clarifications, qualifications, exceptions, or additions to the nonmandatory appendices when it develops its NRC staff position in the regulatory guide on ASME/ANS RA-S-1.4-2021 and NEI 20-09.

Peer Reviews

Peer reviews play an important role in demonstrating the acceptability of PRAs used to support ANLWR licensing. Specifically, peer reviews are expected to help reduce the need for an in-depth review of the PRA by NRC reviewers, allowing them to focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the licensing application. However, the NRC staff may elect to review any PRA technical element at its own discretion.

ASME/ANS RA-S-1.4-2021 is expected to identify two categories of peer reviews. A full-scope peer review examines all high-level requirements and applicable supporting requirements. In contrast, a focused-scope peer review is a subset of a full-scope peer review that involves specific PRA technical elements and their associated high-level and supporting requirements. Peer reviews should be conducted at each stage of the licensing process as follows:

- PRAs that support DC, SDA, and ML applications: A full-scope peer review should be conducted

- PRAs that support COL applications that are based on a DC, SDA, or ML: Generally, a focused-scope peer review should be adequate unless special or unique circumstances exist. If the applicant is implementing LMP, a full-scope peer review should be conducted.

- PRAs that support custom COL or CP applications: A full-scope peer review should be conducted.

- Fuel-load PRAs required by 10 CFR 50.71(h)(1) for COL holders or PRAs that support OL applications: Generally, a focused-scope peer review should be adequate.

- PRAs that have been upgraded, as required by 10 CFR 50.71(h)(2) for COL holders, to incorporate the first four years of commercial operating experience: Generally, a focused-scope peer review should be adequate.

In general, the NRC staff expects that focused-scope peer reviews will mainly concentrate on SRs that have become applicable due to the change in the plant licensing stage. For example, a focused-scope peer review of a PRA that supports COL applications that are based on a DC, SDA, or ML will concentrate on the incorporation of site-specific external hazard and radiological consequence assessment inputs (e.g., meteorology, demographics). However, the NRC notes that some of the previously reviewed SRs may need to be re-examined in subsequent peer reviews to determine whether PRA changes meet the criteria for PRA upgrade.

The NRC staff is familiar with the pre-publication version of ASME/ANS RA-S-1.4-2021 that the ASME/ANS committee concurred on in August 2020 and the proposed guidance in NEI 20-09. The overall approaches in the ASME/ANS standard, as concurred on in August 2020, and NEI 20-09 are generally acceptable to the NRC staff. It should be noted that the NRC may develop clarifications, qualifications, exceptions, or additions to these documents when it develops its NRC staff position in the regulatory guide on ASME/ANS RA-S-1.4-2021 and NEI 20-09.

Quality Assurance Requirements

As stated in DC/COL-ISG-028, the NRC staff has determined that a PRA used to support LWR licensing need not be included within a formal quality assurance program that meets the provisions of Appendix B to 10 CFR Part 50. The NRC staff believes that this determination should also apply to PRAs that support ANLWR licensing. However, consistent with Section 5 of RG 1.174, the NRC staff expects that the PRA will be subjected to quality control. To the extent that a licensee elects to use PRA information to enhance or modify activities affecting the safety-related functions of systems, structures, and components (SSCs), the following (in conjunction with the other views and perspectives presented in this draft white paper) describes methods acceptable to the NRC staff to ensure that the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 are met and that the PRA is sufficient for use in regulatory decisions:

- Use personnel qualified for the analysis.
- Use procedures that ensure control of documentation, including revisions, and provide for independent review, verification, or checking of calculations and information used in the analyses. (An independent peer review is an important element in this process.)
- Provide documentation and maintain records.
- Use procedures to ensure appropriate attention and corrective actions if assumptions, analyses, or information used in previous decision-making are changed (e.g., licensee voluntary action) or determined to be in error.
- When performance monitoring programs are used to support or maintain the licensing basis, those programs should include quality assurance provisions commensurate with the safety significance of affected SSCs.

Documentation

Consistent with Section 6.2 of RG 1.174, archival documentation should include a detailed description of engineering analyses conducted and the results obtained, irrespective of whether they were quantitative or qualitative, or whether the analyses used traditional engineering methods or probabilistic approaches. The applicant/licensee should maintain this documentation as part of its quality assurance program, so that it is available for examination.

The HLRs and SRs for PRA configuration control expected to be provided in Section 5 of ASME/ANS RA-S-1.4-2021 are generally acceptable to the NRC staff. It should be noted that the NRC may develop clarifications, qualifications, exceptions, or additions to these requirements when it develops its NRC staff position in the regulatory guide on ASME/ANS RA-S-1.4-2021 and NEI 20-09.

In 2007, the Nuclear Regulatory Commission (NRC) amended its regulations by revising the provisions applicable to the licensing and approval processes for nuclear power plants in Part 52 (Ref. 28). These amendments clarified the applicability of various requirements to each of the Part 52 licensing processes by making necessary conforming amendments throughout the NRC's regulations to enhance the NRC's regulatory effectiveness and efficiency in implementing its licensing and approval processes. Consistent with the Statements of Consideration for this rulemaking:

- For standard design certifications, the entire PRA does not need to be included in Tier 2 information (information that is approved but not certified by the NRC) because it is not part of the design basis information.
- For standard design certifications, the description of the PRA and its results that is required by 10 CFR 52.47(a)(27) is part of the Tier 2 information.

- The NRC expects that, generally, the information that it needs to perform its reviews of licensing applications from a PRA perspective is that information that will be contained in an applicant's FSAR, and that the complete PRA (e.g., logic models, supporting information and data, codes) would be available for NRC inspection or audit at the applicant's offices, if needed.

There are several sources of information that may be consulted when developing the description of the PRA and its results that will be submitted to the NRC as part of an ANLWR licensing application:

- RG 1.206 provides guidance on the content for standard design certification and combined license applications.

- The industry-led technology-inclusive content of application project (TI-CAP) is developing ~~a~~ proposed content for specific portions of the safety analysis report (SAR) that would be used to support an advanced reactor application. The TI-CAP portion of the SAR will be informed by the guidance found in the LMP guidance (NEI 18-04).

- The NRC-led advanced reactor content of application project (ARCAP) is developing technology-inclusive, risk-informed and performance-based application guidance that is intended to be used for an advanced reactor application for a combined license, construction permit, operating license, design certification, standard design approval, or manufacturing license. The ARCAP is broader than the TI-CAP and encompasses it. ARCAP is a longer-term effort that will support the 10 CFR Part 53 rulemaking effort.

Documentation of the fuel-load PRA required by 10 CFR 50.71(h)(1) for COL holders and the four-year PRA upgrades required by 10 CFR 50.71(h)(2) for COL holders is required to be provided as updates to the final safety analysis report as required by 10 CFR 50.71(e).

Acronyms and Initialisms

ANLWR	advanced non-light water reactor
ANS	American Nuclear Society
ARCAP	advanced reactor content of application project
ASME	American Society of Mechanical Engineers
COL	combined license (10 CFR Part 52, Subpart C)
CCFP	conditional containment failure probability
CP	construction permit (10 CFR Part 50)
DC	standard design certification (10 CFR Part 52, Subpart B)
EAB	exclusion area boundary
HLR	high-level requirement
ITAAC	inspections, test, analyses, and acceptance criteria
LMP	licensing modernization project
LPSD	low-power and shutdown
LWR	light water reactor
ML	manufacturing license (10 CFR Part 52, Subpart F)
MSPI	Mitigating Systems Performance Index
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OL	operating license (10 CFR Part 50)
POS	plant operating state
PRA	probabilistic risk assessment
RAP	reliability assurance program
RG	regulatory guide
RTNSS	regulatory treatment of non-safety systems
SAMDA	severe accident mitigation design alternative (10 CFR Part 51)
SDA	standard design approval (10 CFR Part 52, Subpart E)
SDP	significance determination process
SR	supporting requirement
SSC	systems, structures, and components
TEDE	total effective dose equivalent
TI-CAP	technology-inclusive content of application project

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