



**Status and Plans to
Develop Regulatory
Guide Endorsing the
Advanced Non-LWR
PRA Standard**

February 23,
2021

Agenda

- NRC Staff Draft White Paper: Demonstrating the Acceptability of Probabilistic Risk Assessment Results Used to Support Advanced Non-Light Water Reactor Plant Licensing
- Staff observations of NEI 20-09, “Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR Standard”

Update on Staff Endorsement

- ANLWR PRA Standard ASME/ANS RA-S-1.4-2021 issued as an ANSI standard on February 8, 2021.
- Action Plan for review and endorsement of the standard has been updated – MLxxxxxx
- Comparison of the ANLWR PRA standard to other related LWR PRA standards and guidance progressing

Schedule

Milestone	Activity
June 2021	RG technically complete; start RG publication process
June/July 2021	Briefing to the ACRS Future Plant Designs Subcommittee
December 2021	RG issued for trial use
TBD	Revise trial-use RG to incorporate lessons learned and revision to ASME/ANS RA-S-1.4 (anticipated in 2023-2024)

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

Marty Stutzke

Senior Technical Advisor for Probabilistic Risk Assessment
Division of Advanced Reactors and Non-Power Production and
Utilization Facilities (DANU)
Office of Nuclear Reactor Regulation (NRR)

February 23, 2021

The NRC Staff Draft White Paper

- Released January 19, 2021 (ADAMS Accession No. ML21015A434).
- Provides staff views and perspectives for public discussion:
 - Content is subject to change
 - Should not be interpreted as official agency positions
 - Will help inform the staff's development of regulatory guidance

NRC Staff Draft White Paper Topics

- Applicability
- Graded PRA
- Applicable regulations and rulemaking
- Consideration and use of related guidance for LWR PRAs
- Uses of the PRA
- Definition of PRA acceptability
- Evolution of the PRA during various plant licensing stages
- PRA scope
- Risk metrics and the use of intermediate risk metrics
- Selection of the bounding site
- Use and endorsement of the ASME/ANS NLWR PRA standard
- Applicability of supporting requirements during various plant licensing stages
- Capability Categories
- Risk significance
- Nonmandatory appendices
- Peer reviews
- Quality assurance
- Documentation

Applicability

- Examples of NLWRs 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
 - Reactors that are cooled by heat pipes
- Applies to stationary NLWRs:
 - Reactors that may be constructed at a site
 - Reactors that are constructed at an offsite facility and subsequently transported and installed at a site
 - Does not address PRAs used to assess the risk of transporting NLWRs from an offsite facility to the site
 - Does not address mobile reactors, which may be relocated to different sites after initial criticality

Graded (Rightsized) PRA

- 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 decisionmaking, 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 NLWR 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.

Regulations and Rulemakings

- 10 CFR Part 52
 - Subpart B: Standard Design Certification (DC)
 - Subpart C: Combined License (COL)
 - Subpart E: Standard Design Approval (SDA)
 - Subpart F: Manufacturing License (ML)
- 10 CFR Part 50
 - Current regulations do not require applicants for Part 50 construction permits or operating licenses to provide PRA-related information
 - Commission policy statements:
 - Advanced reactor policy statement (73 FR 60612; October 14, 2008)
 - Severe accident policy statement (50 FR 32138; August 8, 1985)
 - Safety goal policy statement (51 FR 28044; August 4, 1986 as corrected and republished at 51 FR 30028; August 21, 1986)
 - PRA policy statement (60 FR 42622; August 16, 1995)
 - Rulemaking “Incorporation of Lessons Learned from New Reactor Licensing Process (Parts 50 and 52 Licensing Process Alignment),” Docket NRC-2009-0196, RIN-3150-AI66
- Proposed 10 CFR Part 53
 - Rulemaking “Risk Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors,” Docket NRC-2019-0062, RIN 3150-AK31
 - Being developed as required by the Nuclear Energy Innovation and Modernization Act (NEIMA)

Related Guidance

- RG 1.200, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities”
- NUREG-0800, Section 19.1, titled “Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”
- NUREG-0800, Section 19.0, titled “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors”
- NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking”
- RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis”
- RG 1.206, “Applications for Nuclear Power Plants”
- 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”
- 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”

The staff intends to maintain consistency with existing PRA-related staff positions to the extent possible.

Uses of the PRA

Provide insights which support the conclusion that the NLWR plant design, construction, and operation provides reasonable assurance of no undue risk

Identify severe accident vulnerabilities

Demonstrate that the NLWR plant meets the Commission's safety goals

Support the regulatory oversight process

Support NEI 18-04 implementation

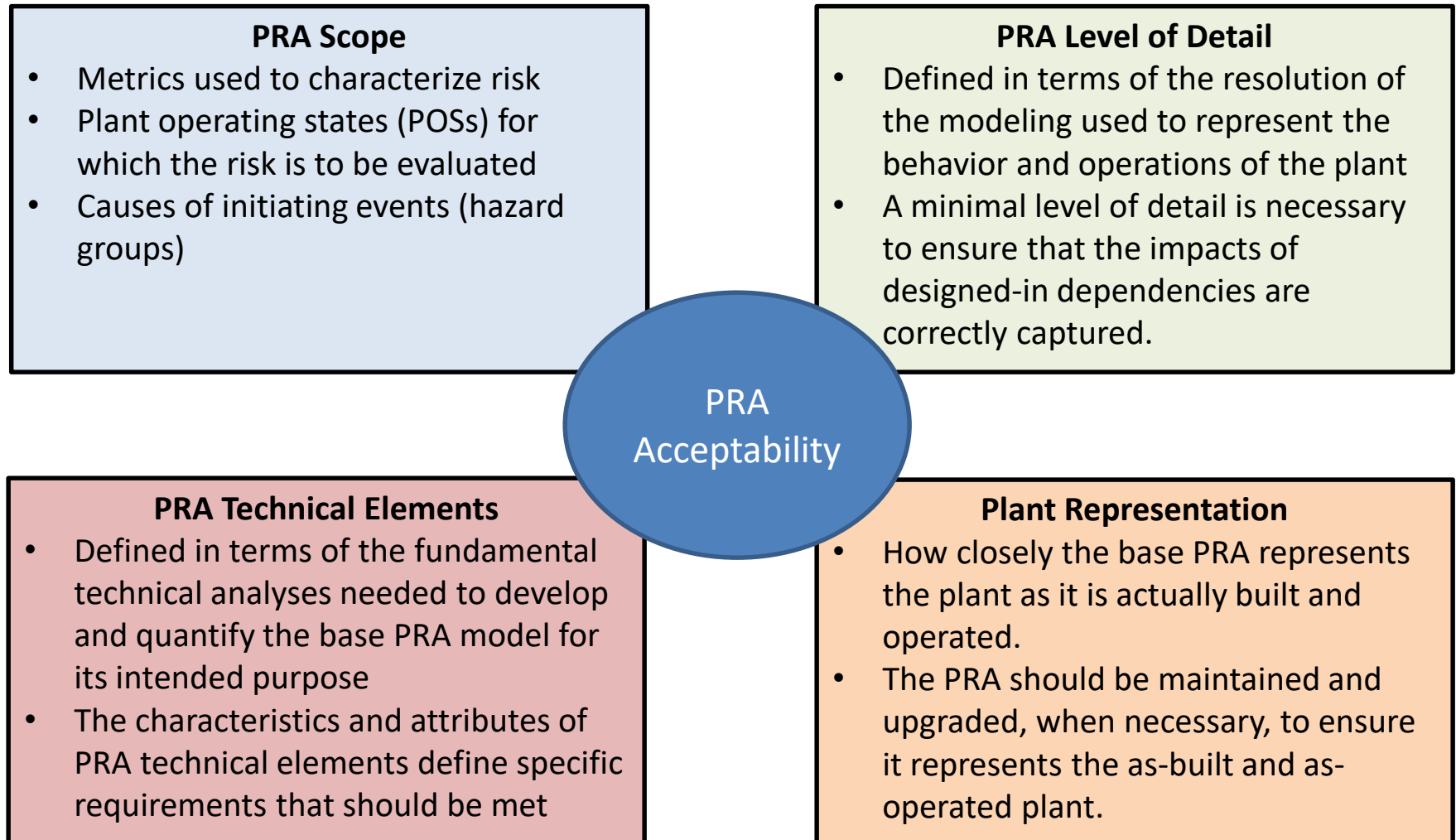
Support voluntary risk-informed applications

Identify and support the development of specifications and performance objectives

Support severe accident mitigation design alternative (SAMDA) evaluations

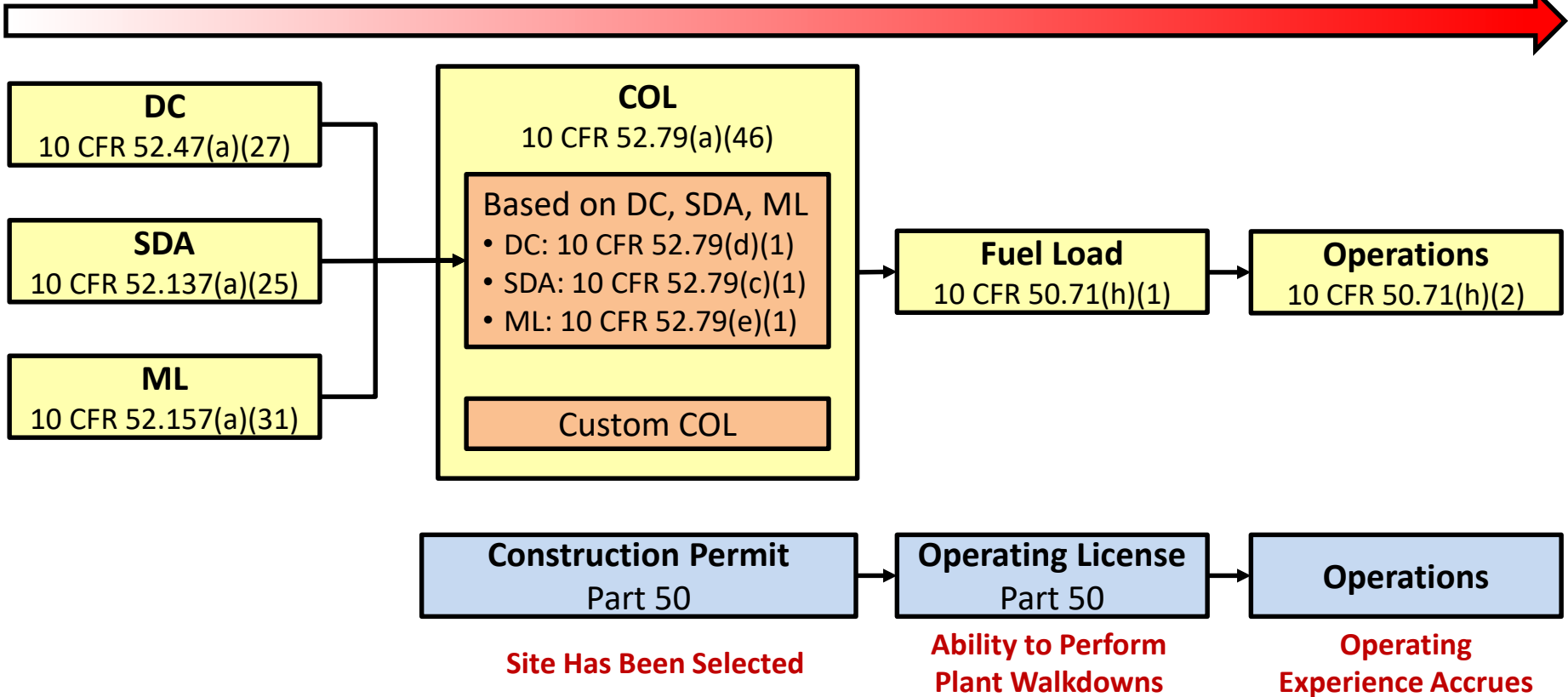


The Definition of PRA Acceptability



Evolution of the PRA During Various Plant Licensing Stages

Increasing level of detail and plant representation in the PRA →



Proposed NLWR PRA Scope

- Address all radiological sources at the plant:
 - Reactor cores
 - Spent fuel
 - Fuel reprocessing facilities for molten salt reactors)
 - Accident scenarios that lead to a radioactive release from multiple radiological sources.
- Address all hazards:
 - All internal hazards such as, but not limited, to internal initiating events, internal floods, and internal fires.
 - 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 .
- Address all plant operating states (e.g., at-power, low-power, shutdown)
- An NLWR PRA should be a Level 3 PRA:
 - Develop the frequencies of accident scenarios from the occurrence of an initiating event until the release of radioactive materials to the environment
 - Estimate the consequences that result from the release



NEW



NEW

Risk Metrics

- Individual prompt (early) fatality risk (per plant-year), 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).
- The use of intermediate risk metric such as core-damage frequency (CDF) or large release frequency (LRF) may be acceptable:
 - Define and justify all user-defined intermediate risk metrics (e.g., CDF, LRF)
 - Explain how the user-defined intermediate risk metrics capture the risk from all radiological sources
 - Show how the user-defined intermediate risk metrics relate to the quantitative health objectives defined in the Commission's safety goal policy statement.

Bounding Sites

- The ASME/ANS NLWR PRA standard specifies the use of a bounding site if the actual site has not been selected:
 - A bounding site is “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 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 ASME/ANS NLWR PRA standard, is different than the set of site parameters used in a design certification or early site permit application.
- Bounding sites are used for PRAs that support DC, SDA, or ML applications.
- 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.

Use of the ASME/ANS NLWR PRA Standard, ASME/ANS RA-S-1.4-2021, “Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants,” 2/8/2021

- The NRC staff currently anticipates that the high-level requirements (HLRs) and supporting requirements (SRs) in ASME/ANS RA-S-1.4-2021 will be generally acceptable for development of PRAs that support NLWR licensing.
- Note that the NRC may develop clarifications, qualifications, exceptions, or additions to these documents when it develops its NRC staff position in the regulatory guide.

Applicability of Supporting Requirements During Various Plant Licensing Stages

- The ASME/ANS NLWR PRA standard was developed to support NLWR plant design, construction, and operation.
- Supporting requirements (SRs) may contain qualifiers such as “pre-operational,” “operating plants,” “bounding site,” and “specific site” that can be used to determine how each SR relates to the plant licensing stages.
 - If no qualifier is provided, then the SR applies to all plant licensing stages.
 - 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 in the draft staff white paper provides a suggested approach for determining which SRs apply to specific plant licensing stages.

Draft White Paper Table 2

Guide for Determining the Applicability of Supporting Requirements for Various Plant Licensing Stages (1 of 2)

SR Type	Qualifier	Plant Licensing Stage				Example
		DC, SDA, or ML Application	COL or CP Application	COL Holder Fuel-Load PRA or OL Application	Commercial Operations	
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

Draft White Paper Table 2

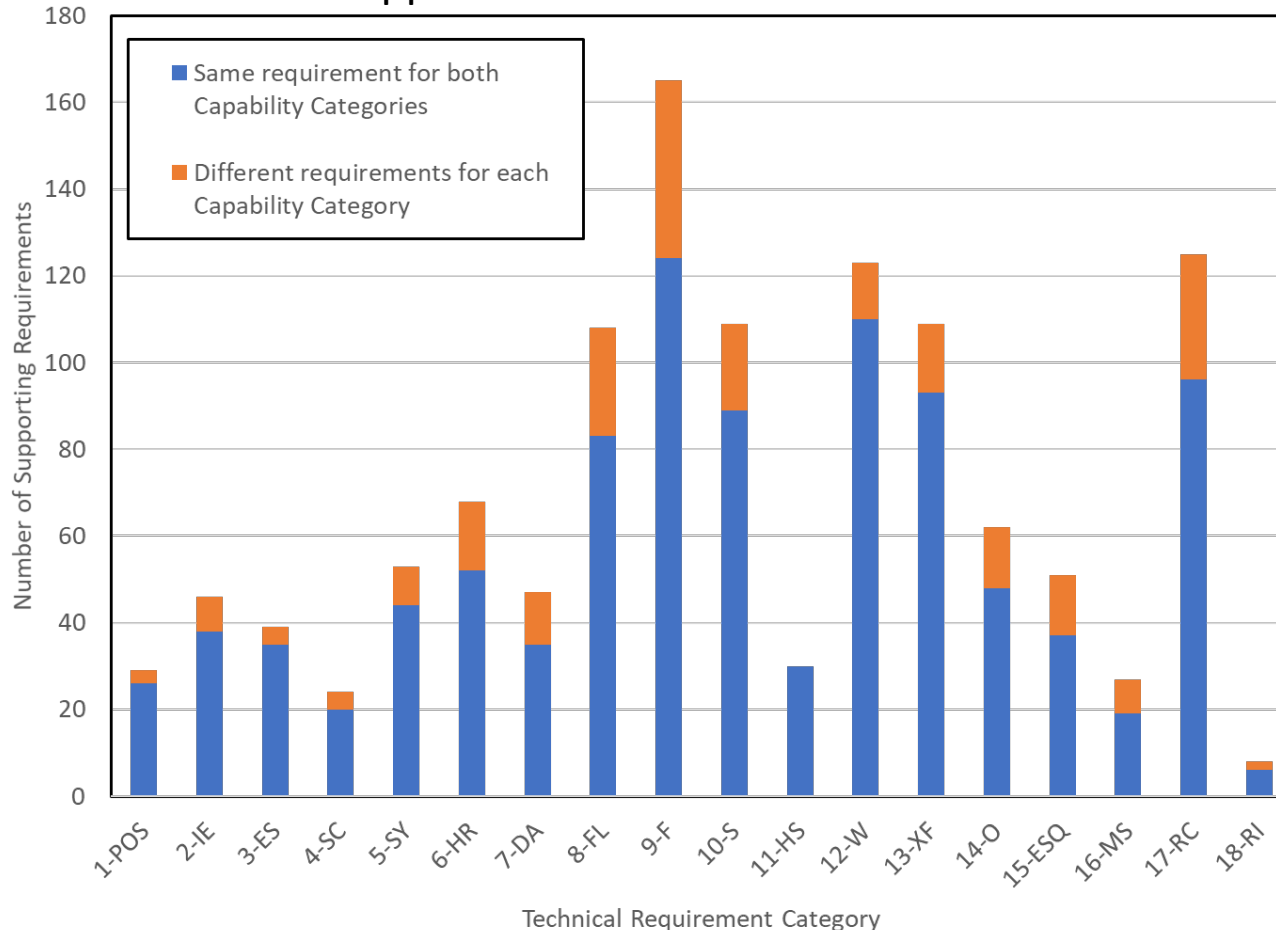
Guide for Determining the Applicability of Supporting Requirements for Various Plant Licensing Stages (2 of 2)

SR Type	Qualifier	Plant Licensing Stage				Example
		DC, SDA, or ML Application	COL or CP Application	COL Holder Fuel-Load PRA or OL Application	Commercial Operations	
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
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

^aThe portion of the SR that pertains to the use of a bounding site applies.
^bThe portion of the SR that pertains the use of a specific site applies.

Capability Categories

- Consistent with DC/COL-ISG-028, CC-I is generally acceptable if the NLWR licensing application is not based on the LMP guidance and does not involve concurrent voluntary risk-informed applications.
- Consistent with RG 1.174, CC-II is generally expected if the NLWR licensing application is based on the LMP guidance or involves concurrent voluntary risk-informed applications.



- About 20% of the supporting requirements distinguish between CC-I and CC-II
- Not very sensitive to the plant life cycle stage

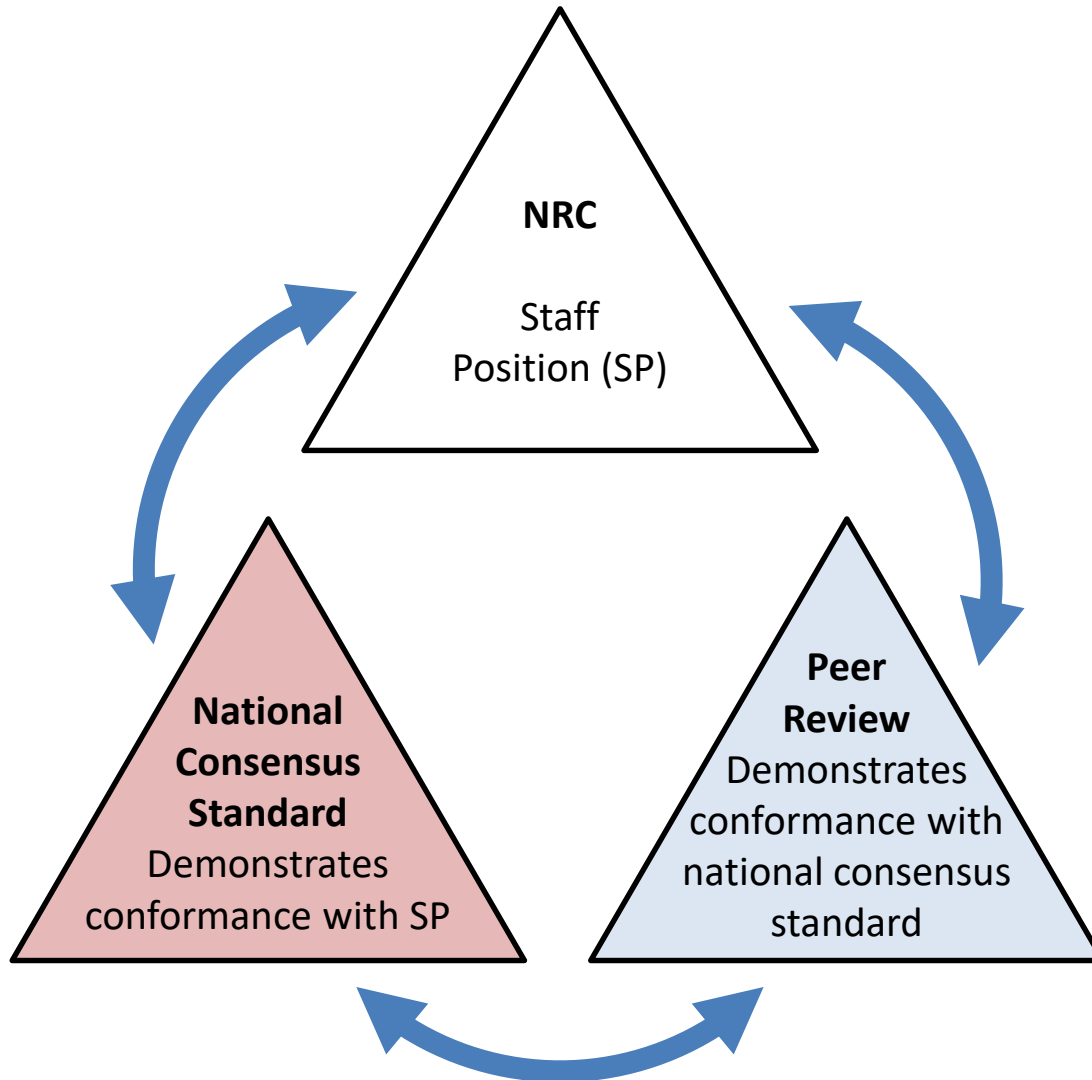
Risk Significance

- 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.
 - 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.
 - 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.
- The ASME/ANS NLWR PRA standard includes 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.
- NRC staff expects the PRA results to always include risk-significant items that have been identified by using relative criteria (e.g., Fussell-Vesely, risk achievement worth).
- In addition, if absolute risk significance criteria have been used to develop the PRA or to support risk-informed applications such as the LMP guidance, then the NRC staff expects the PRA results to also include risk-significant items that have been identified using absolute criteria.

Use of Nonmandatory Appendices

- The nonmandatory appendices in ASME/ANS NLWR PRA standard may be binned into two groups:
 - Notes that support the understanding of various SRs, and
 - Commentaries
- The NRC staff generally accepts the Notes.
- The NRC staff has no opinion about the Commentaries.
- Note 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.

NRC General Framework for Achieving PRA Acceptability



Peer Reviews

- The observation that some of the SRs in the ASME/ANS NLWR PRA standard only apply to certain plant licensing stages has important ramifications with respect to PRA upgrade and peer reviews.
- Definitions in the ASME/ANS NLWR PRA standard:
 - 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.
- Types of peer reviews:
 - A full-scope peer review examines all high-level requirements and applicable supporting requirements.
 - 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 Implied by the ASME/ANS NLWR PRA Standard

Application	Peer Review
PRAs that support DC, SDA, and ML applications	Full-scope
PRAs that support COL applications that are based on a DC, SDA, or ML	Focused-scope
PRAs that support custom COL or CP applications	Full-scope
Fuel-load PRAs required by 10 CFR 50.71(h)(1) for COL holders or PRAs that support OL applications	Focused scope
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	Focused-scope

Use of NEI 20-09

“Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard,” pending.

- The overall approaches to peer review in the ASME/ANS NLWR PRA standard and NEI 20-09 are generally acceptable to the NRC staff.
- Note that the NRC may develop clarifications, qualifications, exceptions, or additions to these documents when it develops its NRC staff position in the regulatory guide.

Quality Assurance

- Consistent with DC/COL-ISG-028, the NRC staff believes that a PRA used to support NLWR licensing need not be included within a formal quality assurance program that meets the provisions of Appendix B to 10 CFR Part 50.
- Consistent with Section 5 of RG 1.174, the NRC staff expects that the PRA will be subjected to quality control.
 - 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 (1 of 2)

- Consistent with Section 6.2 of RG 1.174, maintain PRA archival documentation as part of a quality assurance program so that it is available for examination.
- The requirements in the ASME/ANS NLWR PRA standard for configuration control are generally acceptable to the NRC staff.
- 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.

Documentation (2 of 2)

- 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.

Acronyms and Initialisms (1 of 2)

ANS	American Nuclear Society
ARCAP	Advanced Reactor Content of Application Project
ASME	American Society of Mechanical Engineers
CC	Capability Category
CDF	core-damage frequency
COL	combined license
CP	construction permit
DC	standard design certification
EAB	exclusion area boundary
FR	Federal Register
FSAR	final safety analysis report
JCNRM	Joint Committee on Nuclear Risk Management
ISG	interim staff guidance
LMP	Licensing Modernization Project
LRF	large release frequency
LWR	light water reactor

Acronyms and Initialisms (2 of 2)

ML	manufacturing license
NEI	Nuclear Energy Institute
NEIMA	Nuclear Energy Innovation and Modernization Act
NLWR	Non-light water reactor
NRC	Nuclear Regulatory Commission
OL	operating license
POS	plant operating state
PRA	probabilistic risk assessment
RG	regulatory guide
SAMDA	severe accident mitigation design alternative
SDA	standard design approval
SP	staff position
SR	supporting requirement
SSC	systems, structures, and components
TEDE	total effective dose equivalent
TI-CAP	Technology-Inclusive Content of Application Project

Feedback on NEI 20-09 dated November 2020 (ML20339A485)

- **Page 5:** It is unclear that back referencing between supporting requirements in the ANLWR PRA Standard is different from the Level 1/LERF LWR PRA Standard. It will be beneficial if NEI 20-09 either includes an explanation of the unique back referencing for ANLWR Std or retains the language from NEI 17-07 that was deleted from NEI 20-09.
- **Page 7:** It is unclear that back referencing between supporting requirements in the ANLWR PRA Standard is different from the Level 1/LERF LWR PRA Standard. It will be beneficial if NEI 20-09 either includes an explanation of the unique back referencing for ANLWR Std or retains the language from NEI 17-07 that was deleted from NEI 20-09.
- **Page 10:** The proposed deletion of the minimum peer-review team size appears to be pre-mature because it is based on a potential future version of the ANLWR Standard that does not exist and has not been considered for endorsement by the NRC.

Feedback on NEI 20-09 (continued)

- **Page 11:** The proposed deletion of reference to ANLWR PRA Std for the peer-review team size appears to be pre-mature because it is based on a version of the ANLWR Std that does not exist and has not been considered for endorsement by the NRC.
- **Page 13:** Inclusion of “economic impact modeling” will be beneficial because it would be consistent with the severe accident mitigation design alternatives (SAMDA) analysis.
- **Page 24:** It would be desirable to re-introduce language in NEI 20-09 on the assignment of peer-review findings based on “preponderance of evidence” with reference to “NRC ANLWR PRA acceptability regulatory guidance.”