American Nuclear Society (ANS) Standard ANSI/ANS-30.3-2022

July 15, 2024

India Banks, Project Manager

Licensing and Regulatory Infrastructure Branch

Division of New and Renewed Licensing

Office of Nuclear Reactor Regulation

Meeting Objective & Agenda

Objective: Discuss ANSI/ANS-30.3-2022, Light Water Reactor Risk-Informed, Performance Based Design

Agenda:

- Project kickoff: India Banks, Project Manager, (NRR/DNRL) – (9:00-9:10am)
- Opening remarks: Michele Sampson, Director (RES/DE) and Andrew Sowder, Chair (ANS Standards Board) – (9:10-9:15am)
- NRC staff observations – (9:15-10:00am)
- ANS presentation – (10:00-10:30am)
- Break – (10:30-10:40am)
- NEI remarks – (10:40-10:50am)
- Discussion – (10:50-11:30am)
- Public comments – (11:30-11:50am)
- Concluding remarks: Mike Franovich, Director (NRR/DRA) – (11:50-12:00pm)

NRC Staff Feedback on Jan 21, 2024, ANS Letter to NRC

Contributing Divisions

NRR/Division of Safety Systems
NRR/Division of Risk Assessment
NRR/Division of New and Renewed Licenses
NSIR/Division of Preparedness and Response
RES/Division of Engineering

Cognizant Divisions

NRR/Division of Engineering and External Hazards
NRR/Division of Advanced Reactors and Non-Power Production and Utilization Facilities

Background

- August 9, 2022 (ML23111A238): ANS requested NRC staff to review and endorse ANS 30.3-2022.

- June 27, 2023 (ML23121A283): NRC informed ANS that (a) NRC plans to perform a detailed review at the appropriate time, and (b) the review will determine whether the standard should be endorsed. The letter then provided a non-exhaustive list of general and preliminary technical observations.

- January 31, 2024 (ML24046A023): ANS-30.3 Working Group replied to NRC with preliminary comments for each of NRC's general and preliminary technical observations.

Comment #1 & ANS Response

NRC Comment #1 (General): ANSI/ANS-30.3-2022 provides broad and high-level guidance to designers of advanced light water reactors. While this objective is consistent with the standard's intended purpose as design guidance, standards endorsed by the NRC in the past have included substantially more detail.

ANS Response #1*:* ANSI/ANS-30.3-2022 is a performance-based standard; therefore, it would not normally be expected to contain the level of technical detail typically found in prescriptive documents...

...Section 11, “Performance-based decision making,” may be seen as groundbreaking because it explicitly draws from the Commission’s modernization efforts documented in Staff Requirements Memorandum (SRM) for SECY-98-144, “White Paper on Risk-Informed, Performance-Based Regulation.”

Staff Views on Response to Comment #1

- ANS states that publishing risk-informed performance-based guidance is consistent with Staff Requirements Memorandum (SRM) for SECY-98-144, “White Paper on Risk-Informed, Performance-Based Regulation” and refers to Section 11 to illustrate this point and justify lack of details.

- NRC staff recognizes the significance of SRM-SECY-98-144 and follows that Commission direction. However, clear and consistent implementation of performance-based rules (e.g., 10 CFR 50.65) relies on the development of detailed guidance for implementation.

- In the absence of detailed implementation guidance, the staff will not be able to provide the desired level of regulatory clarity and stability by endorsing ANS 30.3.

Staff View on Responses to Comment #1 (Contd.)

- Furthermore, insufficient level of detail in areas discussed in ANS 30.3 would require numerous interactions to align on language for use of the exemption process by potential applicants of this guidance.

- Consequently, significant staff resources will be required to review guidance; any endorsement of generic use of exemptions could be perceived as *de facto* rulemaking.

- This is an overarching concern applicable to most of the specific comments (#3 - #8) that follow.

Staff Views on Response to Comment #1 (Contd.)

- ANS 30.3 not only provides guidance relating to ensuring safety, but also, how the designs may be optimized to reduce costs. Therefore, NRC would need to expend significant resources to disambiguate a financially driven methodology from a safety driven methodology.

- Entire standard, or portions of the standard, are optional per the second paragraph of 1.3. This appears to supersede all "should" and "shall" designations within the standard. It is challenging for the NRC to endorse a wide-ranging design standard when each portion of it could be used independently of the other portions.

- ANS 30.3 appears to create the potential for substantial interwoven risk reasoning, i.e., changes to the PRA results would prompt an applicant to revisit conclusions based on PRA inputs. This creates a large potential for review rework and schedule threat for regulatory review for both Construction and Operation License reviews.

Comment #2 & ANS Response

NRC Comment #2: On several topics, the standard contains guidance that is noticeably different information from established NRC regulations, policy, guidance, and endorsed documents (e.g., guidance for Title 10 of the Code of Federal Regulations (10 CFR) 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors," and 10 CFR 50.47 "Emergency plans").

ANS Response #2: The standard represents established state-of-practice techniques (e.g., NuScale Design Certification and Standard Design Approval Application) and is expected to be fully compliant with existing NRC light water reactor (LWR) regulations. In addition, the standard provides guidance in some areas (e.g., risk-informed single failure criterion) whereby designers may take **exceptions** to specific regulations or guidance with appropriate justification on a case- by-case basis. Section 11, "Performance-based decision making," on regulatory conformance describes how a designer might develop such justification for departures or exceptions, which are allowed under the existing regulations. Additional specific comments are addressed via responses to NRC Comment #5 (10 CFR 50.69) and #7 (10 CFR 50.47).

Staff Views on Response to Comment #2

- ANS 30.3 first asserts that the standard is expected to be fully compliant with existing NRC light water reactor regulations and then goes on to state that designers may take **exceptions** to specific regulations or guidance with appropriate justification on a case- by-case basis. Staff would like to receive clarification on ANS 30.3's intent of using the term "exception" in their response.

- NRC cannot generically endorse an approach that would require exemptions, which require special circumstances and review on a case-by-case basis.

Comment #3 & ANS Response

NRC Comment #3: Classification of events based on event sequence instead of initiating event frequency could incorrectly result in events being classified inconsistent with current regulatory requirements and staff guidance.

ANS Response #3: It is not clear to what the phrase "classification of events" refers, since this term is not used in the standard. Assuming the comment is referring to the "categorization" of initiating events by frequency and functional event type found in Section 5.1, "Initiating event selection,"…

...This section is consistent with chapter 15 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (formerly issued as NUREG-75/087). Assuming the comment is referring to the identification of design basis events (DBEs) found in Section 5.2, "Identification of DBEs,"

...

Section 5.2.1, "DBE identification using a deterministic approach with incorporation of insights from the PRA," describes a deterministic approach consistent with the manner in which DBEs were identified for the current generation of plants and is entirely consistent with regulatory requirements, staff guidance (e.g., NUREG-0800 and NUREG-75/087), and industry standards (ANS-51.1-1983 [withdrawn], Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, and ANS-52.1-1983 [withdrawn], Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants). In addition, this section requires enhancement of the traditional approach to identifying DBEs by incorporating insights developed in the PRA, allowing for the consideration of design features and operating characteristics that may be unique to the plant design in the progression of event sequences.

Comment #3 & ANS Response (Contd.)

Section 5.2.2, “Identification of DBEs by adjusting the scope of the PRA,” describes an alternative approach that uses the PRA as a primary source of DBE development, similar to NEI 18-04 (Rev. 1), Risk-Informed Performance-Based Technology Guidance for Non-Light Water Reactors, and industry standards for other advanced reactor designs (e.g., ANSI/ANS-53.1-2011 [R2021], Nuclear Safety Design Process For Modular Helium-Cooled Reactor Plants). In beginning with the event sequences of the PRA, this section also requires adjustments to be made to the PRA to incorporate assumptions that would be made in more traditional deterministic analyses to ensure the completeness of the selection of DBEs. NEI 18-04 and the process described in ANSI/ANS-53.1-2011 (R2021) are technology-neutral and applicable to LWRs.

The two ends of the spectrum for identifying DBEs are described above. In between these ends of the spectrum are blended approaches that are acceptable and include deterministic methods with only limited expansion of the design-specific PRA for unique initiators or selected transient and accident scenarios.

Staff Views on Response to Comment #3

- As ANS noted, staff intended to use the term categorization.

- High-level view on 5.2.1: Staff disagrees with several assertions such as, “This section is consistent with chapter 15 of NUREG-0800, ...” Multiple staff and applicant interactions will be needed to identify deviations of the proposed approach from Chapter 15 of the SRP and associated regulations.

- High-Level view on 5.2.2: Similarities to approaches used in NEI 18-04, in itself, is not sufficient for staff to review the approach proposed in ANS 30.3, in part, because staff endorsement of NEI 18-04 considered integrated aspects of the LMP process (e.g., a FC curve and a rigorous treatment of uncertainties are used to select the DBEs) within the context of non-LWR designs.

Staff Views on Response to Comment #3 (Contd.)

- Also, with respect to 5.2.2, note that even though RG 1.233 endorsed NEI 18-04, excerpts from RG 1.233 & RG 1.253 (staff guidance on Technically Inclusive Content of Application to Inform the Licensing Basis, Certifications & Applications for NLWRs) explicitly identify an applicant's need to demonstrate compliance with any innovative approaches to demonstrate compliance or justify exemptions.

- With respect to the ANS statement regarding *"that the two ends of the spectrum for identifying DBEs are described above. In between these ends of the spectrum are blended approaches that are acceptable and include deterministic methods with only limited expansion of the design-specific PRA for unique initiators or selected transient and accident scenarios,"* it is important to note that the staff's review of the proposed approach for acceptability requires significant additional details and consequently significant additional review resources.

Comment #4 & ANS Response

NRC Comment #4: How the risk-informed approach to single failure criteria meets the regulations in 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” and 10 CFR Part 50 Appendix A.

ANS Response #4: 10 CFR 50.46 does not mention the single failure criteria; however, ANSI/ANS-30.3-2022 requires consideration of the potential for single failures in the design and analysis of the plant. For example:

- ANSI/ANS-30.3-2022, Sections 5.2.1.1, “Consideration of single failures and coincident occurrences,” (deterministic approach supplemented by PRA insights) and 5.2.2.2, “Review of the PRA for deterministic insights,” (use of PRA as a primary source supplemented by deterministic considerations) both require consideration of potential single failures in the identification of DBEs.

- ANSI/ANS-30.3-2022, Section 9.1, “Classification,” references the use of ANSI/ANS-58.14-2011 (R2022), Safety and Pressure Integrity Classification Criteria for Light Water Reactors, which requires consideration of single failures during DBEs in classification of systems, structures, and components (SSCs).

Comment #4 & ANS Response (Contd.)

Given such input, Section 6.6, "Risk-informed single-failure criterion consideration," allows for use of the PRA in a review of the plant design to identify the appropriate failures to consider in the safety analysis of the plant. This is consistent with staff positions documented in SECY-05- 138, "Risk-Informed and Performance-Based Alternatives to the Single-Failure Criterion." As indicated in this section, inclusion or exclusion of single failures would be based on event sequences under consideration, the impact on the loss of functioning of components in question on system response to the event, and the cost of addressing postulated failures versus the magnitude of the risk being addressed.

In Section 11, *Performance-based decision-making*," a review for conformance with regulatory requirements is required to be performed. Any departure from regulation would require justification and possibly a request for exemption. The outcome of risk-informed, performance-based design (including treatment of single failures) is intended to encourage alternatives to regulatory requirements or guidance that would otherwise result in design decisions that could cause excessive or unnecessary design or operational complexity and cost with minimal to no safety benefit. This draws from the experience documented in the SRM for SECY-19-0036, "Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves," on the NuScale application.

Staff Views on Responses to Comment #4

- Staff's comment states "...the regulations in 10 CFR 50.46..." This includes the incorporated requirements of 10 CFR 50 Appendix K and General Design Criterion 35, both of which require consideration of single failures. The staff comment is also relevant to regulatory requirements to design SSCs to the single failure criterion, which is intended to ensure sufficient reliability, redundancy, independence, and testability of safety functions. Additional examples include 10 CFR 50.34(f)(3)(vi), 50.49(e)(3), GDC 17, GDC 21.

- Staff agrees that several Commission policies relating to the application of single failure endorse use of risk insights in the application of single failures and the NRC has endorsed the use of risk-informed insights in the application of single failure in RG 1.233 specifically for non-LWRs following NEI 8-04 in its entirety. However, incorporating the single failure to LWRs using 10 CFR 50 or 10 CFR 52 will, at a minimum, require deviations from NRC’s current procedures and consequently significant staff review resources.

Comment #5 & ANS Response

NRC Comment #5: Changes to the categorization process from established NRC regulations, policy, guidance, and endorsed documents for 10 CFR 50.69, including (1) allowing for classification of individual structures, systems, and components (SSCs) as opposed to entire systems, (2) the omission of the risk sensitivity study to assess the potential cumulative impact of the categorization of the SSCs, (3) the omission of constraints on changes from the preliminary classification by the independent panel of experts, and (4) allowing the use of absolute thresholds instead of relative importance. The changes identified above, among others, call into question the potential alternative treatment on SSC reliability and plant risk.

ANS Response #5: The categorization and classification scheme chosen for ANSI/ANS-30.3-2022 was purposefully chosen to be consistent with the 10 CFR 50.69 "four box" approach to support efficient handoff from the designer to the constructor/operator/owner. In addition, NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," in some instances, is highly prescriptive and goes beyond what is needed or required for the design phase, since it was developed for different risk profiles than those expected for advanced passive LWRs.

Comment #5 & ANS Response (Contd.)

Specific responses to the four subparts are provided below :

(1) ANSI/ANS-30.3-2022, Section 9.1, "Classification," references ANSI/ANS-58.14-2011 (R2022) in performing classification of SSCs as safety related, non-safety related, or non-safety related with special treatment. ANSI/ANS-58.14-2011 (R2022) not on of entire systems but SSCs at all levels of the plant design.

(2) Section 9.2, "Categorization," of ANSI/ANS-30.3-2022 requires the performance of risk-sensitivity studies. The risk-sensitivity study demonstrating the cumulative effect of the categorization the bullets under ANSI/ANS-30.3-2022, Section 9.2:...

(3)Current industry guidance on the categorization of SSCs (NEI 00-04) does not address their classification but accepts that classification as it exists for each facility implementing 10 CFR 50.69…

(4) ANSI/ANS-30.3-2022 allows for the use of both absolute and relative importance in the determination of risk significance…

Staff Views on Responses to Comment #5

The staff understands that ANS 30.3 guidance is targeted for the design stage of advanced LWR; current guidance for 10 CFR 50.69 was promulgated for the operating fleet. Staff finds that the ANS 30.3 outlines a potential categorization approach but does not provide specifics to define a categorization methodology.

Response #5 (1)*: *allowing for classification of individual SSCs as opposed to entire systems

Staff understands that classification would apply to entire plant.

Response #5 (2)*: *omission of the risk sensitivity study to assess the potential cumulative impact of the categorization of the SSCs

Response unclear. Are cumulative risk sensitivity studies being considered?

Staff Views on Responses to Comment #5 (Contd.)

Response #5 (3)*: *omission of constraints on changes from the preliminary classification by the independent panel of experts

Staff understands a different process would be used for the final decision making on categorization, different from the Integrated Decision-making Panel (IDP) described in NEI 00-04 guidance for 50.69. The process is not clearly outlined.

ANS Response #5 (4)*: *allowing the use of absolute thresholds instead of relative importance.

Staff appreciates the clarification. The staff understands that the use of absolute risk metrics may be preferred for advanced reactors with lower risk profiles.

Significant staff resources will be necessary to request for additional detail and assure consistency with 10 CFR 50.69.

Comment #6 & ANS Response

NRC Comment #6: The use of a risk metric as a cut-off value for the determination of design basis events without consideration of uncertainty, key assumptions, or cliff edge effects.

ANS Response #6: It is not clear to which section this comment refers unless it is reference to the 10^{-7}/year threshold referenced in Section 5.2, "Identification of DBEs." Our interpretation of this comment is that the staff wants to understand the basis for the cut-off value. This threshold for defining DBEs is consistent with the suggested threshold for the current generation of plants found in NUREG-75/087 and WASH-1270, *Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors*, which is the NRC's suggested cut-off for individual contributions exceeding 10 CFR 100, "Reactor Site Criteria," limits. Therefore, the 10^{-7}/year threshold is consistent with existing NRC guidance. Note that it is significantly less than the threshold between DBEs and beyond design basis events (BDBEs) proposed by NEI 18-04 (10^{-4}/year).

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Staff Views on Response to Comment #6

- The staff concern is not simply focused on the use of the 10^{-7}/year threshold referenced in Section 5.2, “Identification of DBEs.” Staff understand that that it is significantly less than the threshold between DBEs and beyond design basis events (BDBEs) proposed by NEI 18-04 (10^{-4}/year) and endorsed in RG 1.233. Rather, staff’s concern applies to all regulatory matters where quantitative values generated from PRA models are used as thresholds for decision making. In such cases, consistent with NRC safety goal policy and, also reflected by lower tier regulatory guidance such as RG 1.200, staff requires analysts to describe how uncertainties, key assumptions, and cliff edge effects are treated.

- Also, based on the language in ANS 30.3 (e.g.,...statement in Section 5.2 which states that “DBEs consist of initiating events by themselves as well as initiating events with concurrent malfunctions...”), staff cannot ascertain whether an applicant would use initiating event frequencies or event sequence frequencies to identify DBEs.

- Due to insufficient details in ANS 30.3, the NRC staff cannot conclude whether an applicant who uses the standard would be meeting NRC PRA related guidance (RG 1.200) with respect to treatment of uncertainty and key assumptions.

Comment #7 & ANS Response

NRC Comment #7 : The discussion on emergency planning zone sizing does not reference NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," the technical basis for 10 CFR 50.47, "Emergency plans," and the proposed emergency preparedness rule for small modular reactors and other new technologies. The scope of accidents does not state that a spectrum of accidents should be considered as stated in NUREG-0396.

ANS Response #7: Section 8.2, “EPZ sizing,” includes a reference to 10 CFR 50.47. While NUREG-0396 provides the technical basis for 10 CFR 50.47, its addition (or omission) has no impact on the guidance on EPZ (emergency planning zone) sizing in Section 8.2. This is also the case for recent rulemaking on emergency preparedness for small modular reactors and other technologies. Also, because Section 8.2.1.2, “Scope of accidents,” describes the scope of accidents as those from the PRA—including DBEs, BDBEs, and design basis accident sequences—this captures the intent of “a spectrum of accidents.” The standard provides a modern performance-based approach to risk-informed EPZ sizing that goes beyond the guidance in NUREG-0396.

Staff Views on Response to Comment #7

- The NUREG-0396 approach can be generalized—and PRA information used—as appropriate, as was done for the EP rule for SMRs and other new technologies (ML18064A317; RG 1.242)
- The modern planning basis concept is described in NUREG-0654/FEMA-REP-1, Revision 2. The phrase “spectrum of incidents” describes the types of events that inform EP. For example, hostile action and aircraft impact are considered in the planning basis (SECY-03-0165; Decommissioning Rulemaking 87 FR 12254).
- High Level View on Section 4.2: EP is an independent layer of defense-in-depth. EP/EPZ is informed by the design but is not used to demonstrate design safety.
- High Level View on Section 8.2.1: NRC regulations do not require a zero or site boundary EPZ to be set as a design requirement or goal during the design phase.

Comment #8 & ANS Response

NRC Comment #8: The standard does not address Commission expectations for advanced LWR design that have been issued through SECY papers (such as the "Regulatory Treatment of Non-Safety Systems [RTNSS]") and Commission policy statements (such as the "2008 Advanced Reactor Policy Statement"). Section 8 of ANSI/ANS-30.3-2022, "Severe Accident Considerations," references SECY-01-0009, "Modified Reactor Safety Goal Policy Statement." However, the SRM for SECY 01-0009 states, "The Commission has disapproved issuance of the revised Reactor Safety Goal Policy Statement at this time.

ANS Response #8: The last bullet in Section 3.2, "DID principles," was intended to address the use of RTNSS at a high level: Use reliability-enhancing concepts in the design of non-safety systems so as to reduce risk to the extent practicable.

Additional guidance is provided in Section 4.4, "Performance-based safety objectives," on establishing performance-based safety objectives. The reference in the standard states: "As discussed in the NRC severe accident policy statement (SECY-01-0009), new designs should achieve a higher standard of severe accident performance compared to prior designs." Although the SECY-01-0009 is referenced in the standard, it is not required to be followed and was referenced to highlight the NRC's goal of improving the safety of prior designs. Reference to this paper points the user to a position held by the staff at a certain point in time and enriches the knowledge base offering insight into the process at the NRC to arrive at Commission decisions.

Staff Views on Response to Comment #8

- NRC staff agrees with the statements made by ANS regarding SECY 01-0009.
- Staff noted that SECY in our comment to convey the overarching concern that an applicant who uses ANS 30.3 would be expected to demonstrate how the design meets the key Commission guidance relating to the 2008 Advanced Reactor Policy statement. RTNSS was an example to illustrate that need.
- Statement in Section 3.2, "DID principles," does not provide sufficient details for the staff to determine wither an applicant would address Commission expectations relating to RTNSS. Due to insufficient details in ANS 30.3, the NRC staff cannot conclude whether an applicant who uses the standard would be meeting the Commission's expectation for advanced LWR designs.
- Therefore, applicants who choose to use ANS 30.3 will be required to provide additional details when they submit their designs or construction permit applications for NRC review.

Additional Considerations on Resource Implications

- ANS 30.3 uses IAEA Glossary whose definitions can differ from how the same or similar terms are defined in NRC policies, regulations, and other regulatory documents. This poses a significant challenge, and consequently influences the magnitude of staff resources required to review this standard, since NRC reviewers must ensure that staff positions taken with respect to this standard consider differences in the definition of terms in this standard against NRC's definitions of terms as they appear in 10 CFR 50.2, NRC Glossary, and NUREG-2122, which the staff uses to interpret terms.

- ANS 30.3 indicates that "independent" panels should be used at various stages of design, at times conflating these panels with "integrated" and design reliability assurance panels. Clarity in the precise definition of these panels and their roles would be helpful. It is unclear how practical "independent" panels would be during design. What would constitute independence for the purposes of review?

- ANS 30.3 lacks a clear explanation or examples of risk-informed performance measures. As these are fundamental to the conception of the approach. Additional detail would be critical in assessing the utility of this standard wherein performance measures are used to help ensure design/plant remain within design analyses. Clear description of performance measures, and programmatic responses if performance measures indicate insights outside of PSAR/FSAR analyses lacks description in the standard.

Concluding Remarks

- Due to NRC's limited resources, endorsement efforts are prioritized based on need. Our ongoing reviews and preapplication discussions, including the review of safety strategies of applications such as NuScale SDAA, BWRX-300, and AP300 applications, provide the staff with insights on how we can use the current regulatory framework to review risk-informed performance-based designs.

- Furthermore, whereas an NRC endorsement of a standard is desired, such an endorsement is not essential for a designer to use it during the design process.

- The following information would be useful to inform NRC's decision to review ANS 30.3 for potential endorsement:
  - How many vendors have committed to using ANS 30.3 and the time frame for their applications?
  - Has there been any considerations for the development of implementing guidance?

NRC Next Steps

- Consider feedback from stakeholders from this meeting.

- Use feedback to inform NRC's path forward on the request for endorsement.