UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

July 24, 2026

Honorable Ho K. Nieh
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: PROPOSED RULEMAKING ON MODERNIZING REACTOR LICENSING, SAFETY OVERSIGHT, AND SITING PRACTICES (RIN 3150-AL44)

Dear Chairman Nieh:

During its 737th meeting held from July 8 through 9, 2026, the Advisory Committee on Reactor Safeguards (ACRS) discussed with the U.S. Nuclear Regulatory Commission (NRC) staff this proposed rulemaking, which is part of the agency's broader effort to conduct a review and wholesale revision of its regulations to modernize its regulatory framework. We also benefitted from a number of closed Subcommittee Engagements with the staff over the past six months on the specific topics addressed in this letter and from the reference documents.

Conclusions and Recommendations

1. The proposed regulatory changes implement a graded approach that reflects the differing risk profiles of advanced non-light-water reactors (non-LWRs), small modular reactors (SMRs), and microreactors relative to large light-water reactors (LWRs).

2. These changes aim to improve regulatory flexibility and clarity by: (a) removing prescriptive and, in some cases, outdated requirements, and (b) emphasizing risk-informed, technology-inclusive, and performance-based approaches.

3. The rule continues to require adequate defense-in-depth and sufficient safety margins, two key cornerstones of nuclear safety. For advanced reactors, however, these attributes should be evaluated in the context of their design rather than how defense-in-depth and safety margins are established for the existing LWR fleet.

4. Because of the breadth of the rulemaking changes and the speed of development and implementation, the potential exists for contradictions or inconsistencies resulting from the changes that were made. During the public comment period, staff should use the time to look for inconsistent terminology, adverse interactions, gaps, or contradictions and correct them before the rule is finalized.

5. In implementing these changes, it will be important to ensure that the drive to eliminate prescriptive requirements does not inadvertently introduce new safety vulnerabilities. While we encourage justified reduction in the excessive conservatism that has sometimes been associated with overly prescriptive approaches, there are also pitfalls to be avoided when accommodating higher levels of flexibility. Precise navigation will be required to avoid the perils on either side.

6. We support the rulemaking at a high level and offer detailed comments for your consideration in the body of this letter.

Background

On May 23, 2025, the president signed Executive Order (EO) 14300, "Ordering the Reform of the Nuclear Regulatory Commission." Section 5 of the EO directs the NRC to conduct a comprehensive review and wholesale revision of its regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) and associated guidance documents. The proposed rule discussed herein aims to modernize reactor licensing, safety oversight, and siting practices addressing sections 5(f), 5(h), and 5(i) of the EO, and other items that support adding generation to the electrical grid. The proposed rule also includes a number of further changes to the NRC's regulations that will improve the efficiency and efficacy of its licensing process.

The Committee focused its review on the unique, novel, and noteworthy aspects of the following five key topics from the proposed rulemaking to modernize the NRC's regulatory framework, following the ACRS Review Flowchart in the ACRS Guidelines and Best Practices:

- Enhancing Flexibility of Reactor Site Criteria

- Determinate and Data-Backed Thresholds for Reactor Safety Assessments

- Alternative Risk-Informed and Performance-Based Acceptance Criteria for 10 CFR Parts 50 and 52

- Incorporation of Streamlined Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

- Risk Informing 10 CFR § 50.59 and Allowing Flexibility for Changes to Methods

The Committee previously provided advice to the Commission on the key topic, "Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors," via letter report dated February 22, 2025.

Enhancing the Flexibility of Reactor Site Criteria

The smaller physical footprint, reduced radiological risk, and enhanced safety features of advanced reactors—including SMRs and microreactors—relative to large LWRs support a graded approach to site hazard characterization. To evaluate the acceptability of a nuclear power plant site and enable flexibility in site characterization, the proposed rule would establish the hazard of the facility using an unmitigated dose consequence metric. This metric is based on both the potential source term of the nuclear facility from external hazards (which is a function of the nuclear technology) and the distance to its exclusion area boundary (EAB).

Two different tiers are defined based on this metric: a Tier 1 reactor is a power reactor having an unmitigated dose consequence less than 25 rem total effective dose equivalent (TEDE) at the site EAB, and a Tier 2 reactor is a power reactor having an unmitigated dose consequence greater than 25 rem TEDE at the site EAB.

Draft regulatory guide (DG)-4036 provides guidance on performing the unmitigated dose consequence analysis with an appendix describing relevant considerations necessary to model airborne radioactive releases. The guide establishes five nuclear siting categories based on the numerical values of the unmitigated release consistent with guidance in American National Standards Institute (ANSI)/American Nuclear Society (ANS)-2.26-2004. These categories are tied to associated seismic design categories.

In addition, DG-4036 provides hazard-specific approaches to graded site characterization in the areas of geotechnical engineering, geology, seismology, climatology, hydrology and human-induced external hazards. Other draft regulatory guides on site specific ground motion (DG-1462), meteorological monitoring (DG-1463), and flooding (Regulatory Guide (RG) 1.59) have been updated to incorporate graded approaches consistent with the overall siting approach.

Cumulatively, this graded approach defines design basis hazard levels for structures, systems, and components (SSCs) to achieve acceptable and balanced risk across the plant by applying optimized design requirements commensurate with the consequence of failure. Thus, SSCs with higher consequences of failure need to be designed to withstand lower frequency events.

While some of the pieces for a graded approach currently exist in relevant regulatory guides, no comprehensive NRC approach exists. In establishing this graded approach, the draft guidance drew upon current guidance on site characterization and external hazards used for LWRs, non-power production and utilization facilities, and research and test reactors.

ACRS Comments

1. The graded approach to siting and site characterization is a worthy objective as it fundamentally attempts to account for the different hazard/risk profiles associated with advanced reactors, SMRs, and microreactors compared to large LWRs.

2. The graded approaches to geology, hydrology, geotechnical engineering and other site characteristics are reasonable and appropriate. The new guidance on use of off-site meteorological data and the updates to flooding guidance are also reasonable and appropriate.

3. The material-at-risk approach has worked well for Department of Energy non-reactor facilities but may not work as well for the broader range of commercial reactor technologies and sizes without additional guidance on what constitutes an unmitigated event. The draft guidance (DG-4036) currently notes that widespread failure of SSCs is to be assumed (based on a severe external hazard) and only inherent chemical and physical characteristics of the material-at-risk can be considered. Maximum temperature due to loss of cooling can also be considered in the evaluation. However, the guidance is less clear related to reactivity events that can be very important in some microreactor designs. For example, a reactivity addition unmitigated by scram could lead to an energetic event with consequences significantly more severe than those caused by loss

of cooling. Such anticipated transient without scram (ATWS) type events are required to be addressed per 10 CFR Part 50, and it is not clear how or whether they are to be assessed per the new 10 CFR Part 100 requirements. We suggest that the guidance be clarified to explain whether and how reactivity transients without scram, or severe reactivity excursions, are to be assessed as part of an unmitigated dose consequence evaluation.

4. The actual words used to describe the unmitigated dose value need careful consideration from a public perspective. It is merely a metric calculated under unrealistic conditions with the goal of establishing the level of robustness of SSCs necessary to reduce the hazards of the facility to a safe level. However, the perception by the public may be quite different when seeing potentially large numerical values compared to the common perception about the safety of microreactors, advanced non-LWRs, and SMRs.

5. Alignment of the nuclear siting categories to seismic design categories (SDCs) allows optimization of the seismic design relative to the hazard at the beginning of the design process well in advance of any detailed seismic analysis or seismic probabilistic risk assessments (PRAs) being performed. This gradation is appropriate given the impact of seismic requirements on the complexity of the facility design. The use of 'nesting' lower SDC SSCs inside a number of more robust higher SDC SSCs as described in DG-4036, Appendix B, will also help in overall seismic design optimization. This approach is appropriate. It is worth considering the impact of this optimization on other external hazards such as tornados and high wind missiles.

Determinate and Data-Backed Thresholds for Reactor Safety Assessments

The purpose for using determinate and data-backed thresholds in reactor safety assessments is to ensure that, where feasible, safety assessments focus on credible and realistic risks. Establishing clear thresholds for initiating events across a spectrum of event frequencies together with qualitative screening criteria is expected to provide a sound framework for categorizing initiating events, identifying design basis events, and applying a graded approach to beyond design basis events. Establishing what are credible, realistic risks also involves consideration of operating experience, PRA insights, and guidance from sources including the increased enrichment rulemaking, the Licensing Modernization Project (LMP) approaches, non-LWR PRA standard, international guidance and practices as well as previous NRC precedents.

DG-1454 provides such numerical thresholds as well as qualitative screening criteria. Specifically, DG-1454 defines design basis events (DBEs) as those caused by initiating events with frequencies of occurrence that are greater than or equal to 10^{-4} per reactor-year (based upon the upper 95% confidence bound on frequency uncertainty) and include both postulated accidents and anticipated operational occurrences. For these events, the traditional conservative safety analysis methods, such as those described in Chapter 15 of the NUREG-0800 Standard Review Plan, are to be used to demonstrate the capability of safety-related SSCs during design basis events.

Beyond design basis events (BDBEs) are those with a frequency below the threshold for a DBE but greater than or equal to the "threshold of credibility." DG-1454 defines BDBEs as those caused by initiating events with frequencies of occurrence less than 10^{-4} and greater than or equal to 10^{-6} per reactor-year (based on the upper 95% confidence bound on frequency uncertainty) or are required by regulation to be considered in the licensing basis (e.g., ATWS, and station blackout). Safety assessments in this range: (1) assume realistic conditions and use best estimate methods along with evaluations of key uncertainties; (2) take credit for non-safety-related SSCs in the assessment of plant response if they are not affected by the initiating event (similar to the LMP); (3) demonstrate adequacy of defense-in-depth and address cliff-edge effects; and (4) justify that safety margins are sufficient to support a reasonable assurance of safety finding. DG-1454 defines non-credible events using qualitative criteria and those events with initiating event frequencies below 10^{-6} per reactor-year (with consideration of uncertainties); these events are not required to be evaluated by this proposed regulation, unless specifically required by a different regulation to be considered in the licensing basis.

ACRS Comments

We consider this overall approach to be viable and offer the following for consideration:

1. While the thresholds bring welcome analytical discipline to the frequency dimension of risk, we observe that consequence-related acceptance criteria applied within each event category, consistent with LMP, may help ensure that risk is accurately characterized.

2. The new definition of DBE and the existing term design basis accident (DBA) in 10 CFR Part 50 are not consistent with other regulatory uses of these terms. For example, the LMP defines and uses these two terms in a different way than the new 10 CFR Part 50 definition. The DG-1454 also uses initiating event frequencies (and associated cut off values) while LMP uses quantified event sequence frequencies (and different cut off values), potentially leading to confusion. Additionally, the term "DBA" would remain undefined in 10 CFR Part 50, making unclear what distinction is being drawn for instances where DBA (vs DBE) is used within 10 CFR Part 50. We suggest the staff consider actions that may be warranted, such as additional guidance, to avoid confusion in implementing the new definitions.

3. DG-1454, Section C.8, states that compliance with 10 CFR Parts 50 (or 52) inherently provides adequate defense in depth. This is true for the large LWRs for which 10 CFR Part 50 was originally written. However, it is not evident how this inherently remains true for advanced reactors that use 10 CFR Part 50 (or 52). For example, while 10 CFR Part 50 includes specific beyond-design-basis requirements, such as station blackout and ATWS, which provide meaningful regulatory guardrails, those guardrails were constructed from LWR operational experience and reflect vulnerabilities of that specific technology. For advanced reactors with a limited operational experience base, characterized by novel coolant chemistries, passive safety systems, and integrated protection and safety system architectures, evaluating these events, while necessary, may not be obviously sufficient. For this reason, non-LWRs using LMP require a separate assessment of defense-in-depth even though such LMP-based applications today generally use 10 CFR Part 50 (or 52). Advanced reactor applicants who choose to use 10 CFR Part 50 (or 52) without LMP would appear to also need a separate assessment of defense-in-depth. We suggest the staff clarify the conditions under which 10 CFR Part 50 (or 52) compliance inherently ensures adequacy of defense-in-depth and add guidance for defense-in-depth assessments for plant designs that do not meet these conditions.

4. DG-1454 Section C.1, provides limits on when an independent failure of a non-safety system can be included within an initiating event. We suggest this section be clarified to better explain the intent of the restrictions.

Quality Assurance

Quality assurance (QA) requirements in Appendix B of 10 CFR Part 50 were established in 1970 to ensure that SSCs are designed, fabricated, constructed, and tested to perform their intended safety functions. Over time, a need has developed for a modernized QA framework that retains the safety rigor of Appendix B. Proposed Appendix T of 10 CFR Part 50 was created to address this identified need. It is streamlined relative to Appendix B, and use of the Appendix is voluntary. It is intended to provide a technology-inclusive graded approach for implementation and may eliminate the need for NRC oversight of suppliers and vendors. There are three entrance conditions to use Appendix T: it must be for an N^{th} of a kind (NOAK) plant; there must be no change in safety SSCs compared to the first-of-a-kind (FOAK) plant; and the program must include procedures and work processes for implementation of Appendix T.

Appendix T defines requirements for a Quality Management System (QMS) and a Quality Assurance Program (QAP). The QAP requirements are expressed as 11 separate criteria grouped around three topics: management, performance, and assessment. It also includes QA requirements for software. The requirements align with Appendix B and are more consistent with international quality standards. However, any gap between industry QA standards and the Appendix T QA requirements must be addressed by the applicant.

Two questions are included in the *Federal Register* Notice (FRN) on restriction of Appendix T to NOAK plants and on the change process for the QMS under 10 CFR 50.54(a)(5).

<u>ACRS Comments</u>

Quality assurance at nuclear power plants and fuel reprocessing facilities creates an underpinning foundation for nuclear safety by ensuring that planned and systematic actions necessary to design, construct, and operate nuclear facilities provide confidence that SSCs will perform satisfactorily and reliably in service. The proposed Appendix T for NOAK facilities adequately fulfills this objective in lieu of 10 CFR Part 50, Appendix B.

Because of the broad integration of quality assurance with safe operation of nuclear facilities, it is important to have consistency and clarity in quality assurance requirements. We note the following comments as opportunities to improve the consistency and clarity of requirements in the proposed Appendix T:

1. Appendix T has three entry requirements, one of which is that the facility adopting Appendix T is NOAK with sufficient fidelity to the FOAK baseline facility. Staff discussed that Appendix T had been mapped with and fully supports all attributes of quality described in Appendix B. There are additional benefits in Appendix T such as guidance on software quality assurance. Responses to one of the FRN questions may provide an opportunity to reconsider this limitation and allow Appendix T as a viable option for FOAK applicants. This approach would eliminate potential inconsistencies between operators of a specific design and suppliers that may be required to have multiple quality assurance programs to support FOAK and NOAK applications.

2. The applicability of the draft Appendix T to SSCs that are designated as non-safety-related but safety-significant (NSRSS) is inconsistent throughout the draft and warrants clarification. For example, its applicability to SSCs deemed to be "non-safety-related with special treatment" (NSRST) per the RG 1.233-endorsed LMP, or those categorized under "Regulatory Treatment of Non-Safety Systems" (RTNSS) for passive advanced LWRs per Standard Review Plan (NUREG-0800) Chapter 19.3 guidance, is unclear even though these SSCs are selected using similar criteria to those stated for NSRSS. Additionally, the software QA requirement for digital control systems is limited to ensuring "that the safety functions will be performed under design basis conditions." Limiting the digital instrumentation and control (DI&C) requirement in this manner means the requirement would not apply for NSRSS type DI&C SSCs because such SSCs are "relied on to achieve adequate defense in depth or perform risk significant functions," not directly to mitigate design basis conditions.

Alternative Risk-Informed and Performance Based Acceptance Criteria

Many regulations include both numerical and non-numerical limits as acceptance criteria that restrict the ability of applicants to propose alternative approaches without an exemption. The rule would allow expanded use of risk-informed and performance-based criteria enabling the focus to be on safety outcomes and risk significance instead of rigid rules. It also would allow deviations from General Design Criteria with justification in the application instead of using the exemption process. The rule still requires the design to maintain adequate defense-in-depth and sufficient safety margins. The applicant is urged to vet such proposals through pre-application engagement if they choose to use these new flexibilities.

Associated new draft guidance has been developed by the staff (DG-1464). The guidance notes that the applicant should discuss what conditions will be addressed with the criteria and specify the proposed alternative criteria including quantitative values or performance targets. Any sufficiency and adequacy evaluations need to be developed in the context of an integrated decision-making panel using guidance in RG 1.201. The draft guide also discusses the need to consider the cumulative impact of the proposed alternative criteria.

<u>ACRS comments</u>

1. We support the approach as described in DG-1464. Importantly, the rule language requires maintenance of adequate defense-in-depth and sufficient safety margins, key cornerstones of nuclear safety. We also observe that advanced non-LWRs, small modular reactors, and microreactors present different risk profiles than large LWRs. Having a sound technical basis for alternative criteria substantiated by the different risk profiles should enable efficient review and approval.

Risk Informing 10 CFR § 50.59

The changes to the rule will allow the use of quantitative risk metrics such as core damage frequency (CDF) and large early release frequency (LERF) and their changes (Δ-CDF and Δ-LERF) when performing assessments. Whereas early implementations of 10 CFR § 50.59 did not use such quantitative criteria in their assessments because of the lack of experience with risk assessments, today risk assessments are more mature. Thus, such quantitative assessments are consistent with decades of evolving practice on risk-informed regulation at NRC. As part of the process, the changes must not result in large changes to the risk metrics and must maintain adequate defense-in-depth and sufficient safety margins. The Δ-CDF must be less than $5x10^{-7}$ per reactor-year; the Δ-LERF must be less than $5x10^{-8}$ per reactor-year; and the cumulative risk must be less than 10^{-4} per reactor-year for CDF and less than 10^{-5} per reactor-year for LERF. The staff will consider other metrics for non-LWRs. The use of the PRA directly in these evaluations raises issues relating to the quality of the PRA, the associated uncertainty assessment, and the impact of any updates to the PRA from the approved licensing baseline; however, decades of experience from recurring peer reviews of site specific PRAs of the current operating fleet demonstrate that such concerns can be mitigated.

The rule also allows for some changes to models used in evaluations without requiring NRC review and approval if the applicant has an approved verification, validation and uncertainty quantification (VVUQ) program under proposed 10 CFR § 50.221, “Credibility requirements for modeling and simulation.” Such approaches are developing in other parts of the government (e.g. Food and Drug Administration, Federal Aviation Administration) and represent an emerging consensus type of standard for computational modeling. This change in regulation results in a shift away from NRC directly approving each individual model and toward NRC approval of the process, through a structured VVUQ program, by which models deemed to be of low or medium risk are determined to be credible (trusted). It also allows applicants greater flexibility to evolve their models to incorporate advanced modeling methods and updates to operating platforms without NRC approval.

The staff has developed a DG-1468 that describes the high-level attributes of such an approach. The program is centered around establishing risk levels for model and simulation activities. The rigor of the VVUQ assessment should reflect the importance of the model’s role in decision making and the risk associated with errors in the results.

<u>ACRS Comments</u>

1. For the changes proposed related to 10 CFR § 50.59 criteria (c)(2)(i) and (c)(2)(ii), the approach expands the options available to applicants to incorporate quantitative risk results in their assessments when making changes in plant design and operation. Such changes require a risk assessment, supplemented by an evaluation of adequacy of safety margins and defense-in-depth. While the regulation and guidance changes provide significant detail on the PRA acceptability and allowable frequency levels to use for the risk-informed decision, little guidance is provided to define evaluation criteria that the licensee is expected to use in determining whether a risk-informed change has maintained acceptable safety margins and defense-in-depth. For example, the draft regulatory guide calls out the following:

   > "Documentation and justification of how defense-in-depth is maintained.
   > Documentation and justification of how safety margins are maintained."

   Determining whether these criteria are met requires significant qualitative judgment, which is currently validated by the NRC staff in their review of risk-informed license amendment requests submitted per RG 1.174. We recommend that the staff focus on adequacy of these safety margin and defense-in-depth assessments in their early regulatory audits or inspections of licensee self-approvals granted using this new provision, and plan to adjust the guidance as warranted based on these audits.

2. For the parts of the rule associated with computational modeling, the lack of detailed NRC review of some model changes is counterbalanced by the need for an approved VVUQ program. While legacy vendors would probably be attracted to such an approach

given their significant operating experience and test data, this may be difficult for advanced reactors to consider given their lack of operating experience. Thus, the guidance appears largely focused on the operating fleet. The guidance may need to be updated and broadened to consider evaluation models for advanced reactors in the context of the safety margin and defense-in-depth attributes of the design.

3. The use of “modeling and simulation” in the draft guide is purposely quite broad. A sentence or footnote in the guide that would explicitly define the term as applying to a range of computational work for the reactor including, for example, traditional safety evaluation models, PRAs, digital twins, and even structural design and component performance analyses would improve the clarity.

Path Forward

As the NRC advances this modernization effort, we are encouraged by the meaningful progress reflected in this proposed rulemaking. We congratulate the staff who developed these changes in a short amount of time. Furthermore, we observe that the senior staff is supportive and engaged. After the battle at El Alamein in World War II, Winston Churchill said that it was not the end, nor even the beginning of the end, but that perhaps it was the end of the beginning. We have the same cautious hope. We remain interested in this activity and would like to be kept aware of the progress.

We are not requesting a response to this letter.

Sincerely,

Gregory H. Halnon

Gregory H. Halnon
Chairman

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LIST OF ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ANSI	American National Standards Institute
ANS	American Nuclear Society
ATWS	Anticipated Transient Without Scram
BDBE	Beyond Design Basis Event
CFR	Code of Federal Regulations
CDF	Core Damage Frequency
DBA	Design Basis Accident
DBE	Design Basis Event
DG	Draft Regulatory Guide
DI&C	Digital Instrumentation and Control
EAB	Exclusion Area Boundary
E.O.	Executive Order
FOAK	First-of-a-kind
FRN	Federal Register Notice
IEEE	Institute of Electrical and Electronics Engineers
LERF	Large Early Release Frequency
LMP	Licensing Modernization Project
LWR	Light Water Reactor
NRC	U.S. Nuclear Regulatory Commission
NOAK	Nth-of-a-kind
Non-LWR	Non-Light Water Reactor
NSRSS	Non-safety-related but safety-significant
NSRST	Non-safety-related with special treatment
PRA	Probabilistic Risk Assessment
QA	Quality Assurance
QAP	Quality Assurance Program
QMS	Quality Management System
REM	Roentgen Equivalent Man
RG	Regulatory Guide
RTNSS	Regulatory Treatment of Non-Safety Systems
SDC	Seismic Design Categories
SMR	Small Modular Reactor
SSC	Structures, Systems, and Components
TEDE	Total Effective Dose Equivalent
VVUQ	Verification, Validation and Uncertainty Quantification