

August 31, 2026

Subject: Breakthrough Institute Comment on Modernizing Reactor Licensing, Safety Oversight, and Siting Practices [Docket ID NRC-2025-0975].

The Breakthrough Institute (BTI) appreciates the opportunity to comment on the U.S. Nuclear Regulatory Commission's proposed rulemaking, *Modernizing Reactor Licensing, Safety Oversight, and Siting Practices*.¹ BTI is an independent, nonprofit 501(c)(3) global research center that advocates for appropriate regulation and oversight of nuclear reactors to enable the new and continued use of safe and clean nuclear energy. BTI acts in the public interest and does not receive funding from industry.

BTI generally supports the proposed rule's effort to modernize reactor licensing, construction, safety oversight, and siting through more risk-informed, performance-based, and technology-inclusive requirements. Many of the proposed changes remove unnecessary regulatory burden and should be finalized. The effectiveness of these reforms will depend on applying their underlying regulatory principles consistently across the NRC's broader framework. Parts 50, 51, 52, 53, 57, 70, 71, and 72 address common licensing concepts through different rules and concurrent rulemakings. Construction, generic finality, quality assurance, design-change control, and other regulatory issues should not produce materially different outcomes or unnecessary regulatory uncertainty for applicants, licensees, or the NRC staff.

Implementation is equally important. Once the NRC has reviewed and accepted a technical basis, later applicants should be able to rely on that determination where its underlying basis remains applicable. Implementing guidance must preserve the flexibility created by the rule rather than recreate prescriptive or duplicative review through staff practice.

More broadly, the proposed rule and all the regulatory frameworks should protect public health and safety and the common defense and security without unnecessarily limiting the benefits of civilian nuclear energy to society. For example, once adequate protection has been established, additional restrictions should not be simply added to cause conservatism or procedural burden. The recommendations below apply these principles across the proposed rule.

¹ Nuclear Regulatory Commission, *Modernizing Reactor Licensing, Safety Oversight, and Siting Practices*, Proposed Rule, Docket ID NRC-2025-0975, RIN 3150-AL44, [ML26181A098](#), July 2026. [Hereinafter LOS Proposed Rule]

I. SUMMARY OF RECOMMENDATIONS

BTI recommends that the NRC:

1. **Finalize the construction and repeat-deployment reforms while making them consistent across licensing pathways.** Align the construction boundary across Parts 50, 51, and 53 and coordinate it with Parts 57 and 70; keep construction-permit requirements tied to the findings needed before construction; and allow the general construction license to rely on demonstrated design maturity and construction verification rather than requiring full operation of a reference plant as the exclusive eligibility criterion.
2. **Complete the shift toward significance-based design-change control.** Extend the proposed Tier 2* flexibility where no design-specific basis warrants different treatment, provide Part 52 licensees the same bounded at-risk change process available under Part 53, and allow identical previously approved design changes to be reused without duplicative plant-specific approval.
3. **Make Appendix T and codes-and-standards modernization usable in practice.** Permit FOAK use of Appendix T based on demonstrated design and QMS maturity; establish technology-neutral acceptance criteria for alternative QA frameworks; grade QMS change control according to regulatory significance; and allow previously approved codes-and-standards alternatives to be referenced where their technical basis remains applicable.
4. **Ensure the new risk-informed and performance-based pathways create durable flexibility.** Allow previously approved alternative acceptance criteria to be reused; assess PRA adequacy based on fitness for the intended decision rather than making prior application-specific NRC approval the exclusive route; and allow approved VVUQ programs, vendor methodologies, and other programmatic technical work to be referenced by later licensees where technically applicable.
5. **Preserve determinate reactor-safety thresholds while bounding qualitative overrides and avoiding frequency-only credibility screening.** DG-1454 should identify when uncertainty, defense-in-depth, or engineering judgment may change the classification indicated by the numerical DBE/BDDBE thresholds; clarify how inherent and passive safety characteristics can inform the treatment of highly mitigated advanced-reactor scenarios; and ensure that potentially consequential events are not screened from safety

assessment solely because initiating-event frequency falls below the numerical credibility threshold.

6. **Finalize the higher-enrichment and accident-tolerant fuel reforms while addressing downstream storage and transportation.** The NRC should explain how fuel qualified under the new § 50.46a coolability criteria will be evaluated under Parts 71 and 72 and coordinate any necessary guidance before affected fuel reaches storage or transportation.
7. **Finalize the proposed license, permit, and approval-duration reforms with safeguards for long-term implementation.** Develop a programmatic technical basis and updated guidance for operation beyond 80 years, preserve focused review of plant-specific aging and operating experience, make pre-period-of-extended-operation aging-management obligations enforceable, and preserve ESP issue finality unless new information or changed conditions provide a basis for reopening.
8. **Keep emergency preparedness risk-informed and performance-based.** Make § 50.160 available across reactor technologies and retain the proposed risk-informed and performance-based changes to emergency planning, exercises, and program administration; move the 300 MWT benchmark from the regulatory text to the Statement of Considerations; and grade exercise and planning requirements according to demonstrated response needs and readiness. The implementing guidance should prescribe the screening calculation, present dose-distance results per source term rather than frequency-aggregated, and conform its no-EPZ trigger to the 96-hour period in the rule; and state the capability floor above which case-by-case grading operates.
9. **Finalize the graded Part 100 siting framework while removing requirements and guidance that can unnecessarily constrain otherwise acceptable sites.** DG-4036 should distinguish uncertainty that can affect Tier or NSC classification from uncertainty that cannot; § 100.21 should eliminate the additional societal-benefits showing and general low-population preference; RG 4.7 should be updated so that its population-density methodology does not operate as an independent barrier where the proposed reactor satisfies the applicable safety and emergency-planning requirements; and DG-1463 should provide a practical bounding pathway for qualified offsite meteorological information.
10. **Coordinate related NRC rules so that regulatory reforms remain usable across licensing pathways.** The NRC should implement construction, preconstruction, and early-work requirements as a coordinated cross-rule framework; establish a pathway-neutral

mechanism for recognizing prior NRC findings and preserving their substantive and procedural finality across Parts 50, 52, 53, and 57; and coordinate reactor licensing with the downstream storage and transportation requirements in Parts 71 and 72.

11. **Treat implementation and guidance as an integrated part of the reform so that new flexibility does not become de facto prescription in practice.** The NRC should provide clear implementation and transition pathways for new and existing projects, distinguish binding requirements from voluntary acceptable methods, establish a standard process for reusing prior NRC determinations, and conduct an early implementation review to identify and correct inconsistent interpretations, duplicative review, or unnecessarily prescriptive guidance.

II. CONSTRUCTION AND REPEAT DEPLOYMENT

BTI supports the NRC's proposed reforms to the definition of construction, construction-permit requirements, and general licensing for repeat deployment. The proposed rule appropriately moves toward a more function-based construction boundary by focusing on whether onsite activities may affect an SSC's ability to perform a safety or other credited regulatory function, rather than treating broad categories of work as regulated construction based primarily on their relationship to eventual plant operation. It also modernizes construction-permit information requirements and creates useful mechanisms for generic finality and repeat deployment. The final rule should preserve those reforms while applying the construction definition consistently across licensing pathways where the same regulatory function is involved, keeping construction-permit review tied to the findings that must be made before construction, and allowing generic determinations—including sufficiently standardized operational programs—to be relied upon in later deployments where the approved basis remains applicable. These changes would improve predictability, technology inclusivity, and reuse of prior NRC review without eliminating differences among licensing frameworks that serve distinct regulatory purposes.

A. Function-Based and Consistent Definition of “Construction”

BTI conditionally supports the proposed revision to § 50.10. The current construction framework, developed through the 2007 Limited Work Authorization rule, ties regulated construction to specified onsite activities involving SSCs with a reasonable nexus to radiological health and

safety or the common defense and security. The proposed rule would alter the §50.10 definition of construction to:

(a) Definitions. As used in this section, construction means the activities in paragraph (a)(1) of this section.

(1) Activities constituting construction are the driving of piles, subsurface preparation, placement of backfill, concrete, or permanent retaining walls within an excavation, installation of foundations, or in-place assembly, erection, fabrication, or testing, which may impact the required functions of:

- (i) Safety-related structures, systems, or components (SSCs) of a facility, as defined in § 50.2;*
- (ii) SSCs that perform safety significant functions; and*
- (iii) SSCs necessary to comply with part 73 of this chapter.*

BTI is supportive of the proposed revision to § 50.10 that focuses the construction definition on whether specified onsite activities may affect the required function of a safety-related, safety-significant, or security-related SSC, with the caveat that the word “may” is too vague. Overall, the new definition provides a more appropriate boundary than treating an activity as regulated construction merely because operation of the SSC has a reasonable nexus to nuclear safety. However, the use of the word “may” threatens to render the new definition moot because there is no stopping point where something has no pathway to impact.

Take, for example, pile driving for a nonsafety-related office or parking structure located some distance from the reactor. Pile driving produces ground vibration. In principle, that vibration propagates through the soil and therefore “may impact” a safety-related foundation, embedded piping, electrical equipment, or other SSC. As distance increases, the vibration becomes progressively less capable of producing any meaningful effect, but there is no obvious point at which one can say the activity no longer “may impact” the SSC. The same problem arises with excavation: excavation could damage a utility, which could disrupt another system, which could affect construction operations, which could ultimately affect an SSC. Once indirect causal chains count, “may” becomes infinitely inclusive.

The word “may” should be replaced with something more concrete that is less likely to be subjected to a slippery slope of indirect impact. BTI suggests altering the definition to read: “... in-place assembly, erection, fabrication, or testing, *for which there is a credible causal pathway to*

adversely affect the required functions of.” or “. . .*that is reasonably likely to adversely affect* the required functions of.” Either of these provides a basis by which the chain of connections between action and impact could be adequately qualified without the vagueness and broad inclusivity of “may.”

The NRC should apply a consistent construction boundary across licensing pathways to the fullest extent permitted by the Atomic Energy Act (AEA), with differences retained only where required by statute. For reactors proceeding under Parts 50 and 52, § 50.10 establishes the construction-authorization boundary. Part 53 instead uses the definition in § 53.020. Those provisions pursue a similar objective but do not use the same formulation: proposed § 50.10 focuses on whether an activity may affect the required function of an SSC, while § 53.020 defines construction by reference to activities associated with specified categories of SSCs and their special-treatment requirements. The proposed revisions substantially improve alignment between the Part 50 and Part 53 construction frameworks, but the final rule should clarify whether the two provisions are intended to establish the same substantive construction boundary and align the operative language. An applicant should not face a materially different construction/pre-construction constraint simply because an otherwise comparable reactor proceeds under Part 50 rather than Part 53.

The same coordination issue extends to Part 57. Proposed § 57.3 defines construction by reference to activities performed for safety-related and specified security SSCs, while the proposed rule reaches both safety-related SSCs and SSCs that perform safety-significant functions. The two rules also use different operative tests: Part 57 focuses on whether the activity *is for* a covered SSC, whereas proposed § 50.10 focuses on whether the activity *may* affect a covered SSC’s required function. For example, an advanced reactor could rely on a non-safety-related electrical, cooling, or support system to perform a safety-significant function. Under the proposed Part 50, onsite work that may affect that required function can fall within the construction boundary even though the SSC is not classified as safety-related. Under Part 57, the same type of SSC would not necessarily fall within the construction definition if it is neither safety-related nor one of the specified security SSCs. These differences could produce different construction boundaries for functionally comparable SSCs depending on the licensing pathway. The NRC should align the Part 57 construction definition with the functional, safety-significance-based approach adopted in this rulemaking, or explain why the distinct Part 57 framework warrants different treatment.

The materials licensing framework presents a related but distinct problem. Part 70 separately addresses “construction” and “commencement of construction” and retains a framework based in part on the reasonable nexus of an activity to radiological health and safety or common defense and security. BTI’s materials comment recommends moving that framework toward the same functional principle reflected in the proposed rule: the regulatory boundary should depend on the significance of the affected SSC and its credited function, while work outside that boundary should be permitted to proceed at the applicant’s risk where the AEA allows it.² The NRC should apply the same underlying principle across these frameworks: construction restrictions should attach to work that is material to a credited safety or security function, while differences in the regulatory boundary should be retained only where the governing statute or the distinct safety finding requires them.

BTI supports aligning the definition of “construction” across the regulatory parts rather than maintaining an independent construction definition for each Part.³ Such an approach can prevent the environmental regulations from becoming disconnected from changes in the licensing rules. It also means that differences among Parts 50, 53, 57, and the materials framework can carry directly into the determination of what activity requires NRC construction authorization. The NRC should align construction provisions across licensing pathways and explain in implementing guidance which activities may proceed without NRC authorization and when authorization is required. Misalignment of definitions will reduce the efficiency of license reviews and prevent innovations by developers to meet NRC regulations.

In alignment with the function-based principle, BTI also supports the proposed treatment of onsite emergency response facilities. Where emergency preparedness is the only reason the facility would otherwise fall within the construction boundary, the NRC can verify the required functionality through the applicable emergency-planning requirements rather than regulating construction of the entire facility.

The remaining implementation issue is how applicants and NRC staff will apply these definitions to actual SSCs. Applicants may need to classify large numbers of SSCs, including dual-function SSCs and SSCs shared among multiple units or modules. Those determinations directly impact

² Breakthrough Institute, *Comment on Modernizing Materials Licensing*, Docket ID NRC-2025-1370, Comment ID NRC-2025-1370-0038, August 10, 2026, <https://www.regulations.gov/comment/NRC-2025-1370-0038>.

³ Breakthrough Institute, *Comment on Implementation of the National Environmental Policy Act*, Docket ID NRC-2025-0478, Comment ID NRC-2025-0478-0893, August 21, 2026, <https://www.regulations.gov/comment/NRC-2025-0478-0893>.

whether work may proceed as preconstruction or instead requires NRC construction authorization. The NRC should rely on the proposed applicant-developed classification methodologies, including generic methodologies that could be reviewed once and used by multiple applicants, rather than allowing the construction/preconstruction boundary to depend primarily on repeated project-specific staff determinations.

“Construction” should be limited to activities that install or permanently incorporate an SSC, or may affect the required function of an SSC credited for radiological safety, nuclear criticality safety, physical protection, or another safety-significant function. The definition should focus on the regulatory significance of the affected function. Activities outside that scope should not constitute construction and should be permitted to proceed entirely at the applicant’s risk.

The NRC should adopt a comparable functional approach in the current rulemaking and align the definition across all licensing pathways as AEA permits. Applying the same functional principle across the reactor and materials frameworks would reduce unnecessary differences among NRC licensing regimes and provide applicants with a clearer boundary between preliminary work and regulated construction. It would also preserve the NRC’s ability to oversee safety-significant construction while allowing applicants to undertake activities that do not prejudice the agency’s review or compromise its ability to require design changes.

B. Construction-Permit Requirements Should Reflect the Findings Required Before Construction

BTI supports the proposed revisions to § 50.34, which clarify that the technical information in a preliminary safety analysis report should be sufficient to enable the NRC to make the findings required by §§ 50.35(a), 50.40, and 50.50. The proposed change also removes unnecessarily prescriptive language, revises LWR-specific terminology to accommodate a broader range of reactor technologies, and makes corresponding changes to Part 52. The NRC should carry that principle consistently through § 50.34 by tying the information required in a construction-permit application to the findings necessary to authorize construction, rather than requiring greater design maturity solely because additional information will ultimately be needed at the operating-license stage. That approach would improve flexibility and technology inclusivity without reducing the information necessary for the NRC’s construction-permit findings.

The proposed rule removes language that could be read to require a fully developed LOCA analysis methodology at the CP stage and revises containment-related terminology to accommodate functional containments and other non-LWR designs. BTI supports those changes. The NRC should make conforming revisions to § 50.67 where terminology or assumptions would otherwise preserve an LWR-specific requirement that the § 50.34 revisions are intended to eliminate.

Like the proposed revisions to § 50.34, § 53.1309 begins from the principle that a construction-permit application must contain information sufficient for the NRC to resolve the safety matters that must be decided before issuance of the construction permit. Part 53 does not require a completed design at that stage. Section 53.1309(a)(2)(ii) allows aspects of the design to remain preliminary, and § 53.1309 permits additional analysis, research, testing, and experience where design features have not yet been fully developed or demonstrated. Those provisions recognize that construction-permit review is a stage in the licensing process and allow technical work that is not necessary to the construction-permit findings to continue as the design develops.

A design certification and a construction permit need not require the same degree of design maturity. Part 53 may reasonably require information different from Part 50 because its safety framework and licensing methodology differ. But additional design maturity required under § 53.1309 should correspond to a Part 53 CP finding or another determination that must be made before construction, rather than arise simply because the provision incorporates information required for a standard design certification.

But where Part 53 provides a more flexible and technology-inclusive means of accomplishing the same regulatory function, it is a logical reference for modernizing the older requirements. In particular, construction-permit information requirements across the reactor licensing frameworks should reflect the findings the NRC must make before construction and should not require greater design maturity without a corresponding regulatory need. Greater consistency on that principle would preserve the distinct licensing pathways while reducing unnecessary differences in the information required at the same stage of reactor development.

Responses to Specific Requests for Comment

Question 5. Are there any additional requirements in 10 CFR 50.34 that the NRC should consider changing to be more technology-inclusive or provide additional clarity? Should conforming changes to 10 CFR 50.67 be made to be consistent with 10 CFR 50.34 wording updates?

BTI supports the proposed technology-inclusive revisions to § 50.34 and recommends that the NRC continue applying the same principle to any remaining requirements whose prescriptive wording is unnecessary for the findings required at the construction-permit stage. BTI does not identify an additional discrete § 50.34 requirement in this comment that should be removed, but the final rule and implementing guidance should make clear that application content is governed by the information necessary to support the applicable CP findings rather than by legacy LWR-specific expectations that are not necessary for those findings.

Yes, the NRC should make conforming changes to § 50.67 where necessary to maintain consistency with the revised § 50.34 terminology and substantive approach. The two provisions should not impose different terminology or information expectations for the same CP-stage determination.

Question 6. Are there any requirements in 10 CFR 50.34 that the NRC should consider changing that are posing undue barriers to timely and efficient submittal of CP applications, or which are unnecessary at the CP stage? Please provide a basis for your response.

BTI does not identify additional § 50.34 requirements that should be removed at this time. The NRC should finalize the proposed revisions and ensure that CP applicants are required to provide only the information necessary to support the findings required for issuance of the construction permit, rather than information that need not be resolved until a later licensing stage.

C. Generic Finality and the General Construction License Should Support Repeat Deployment

BTI supports the proposed general construction license and generic-finality framework, including applicant-at-risk construction, NRC inspection authority, and the requirement that matters afforded generic finality generally remain resolved in subsequent proceedings. These features

should be retained and implemented to maximize the value of prior NRC review for repeat deployment.

The principal issue is when the general license becomes available. Proposed § 50.10(h) requires a subsequent applicant to reference a design for which the NRC has issued an operating license, or a combined license for which the § 52.103(g) finding has been made, and for which generic finality has been granted. The proposal ties eligibility not only to resolution of the relevant generic design and construction issues, but also to completion of the reference plant's licensing and construction sequence. The proposed rule itself describes eligibility in terms of a design that has been afforded generic finality, successfully constructed under NRC oversight, and placed into operation.

BTI agrees that the general license should principally support NOAK deployment rather than provide an unrestricted FOAK construction pathway. But full operation of the reference plant need not be the exclusive evidence that the SSCs covered by the general license are sufficiently mature. The NRC should provide an alternative eligibility route where the relevant portions of the reference design have been constructed and inspected under NRC oversight, applicable acceptance or verification criteria have been satisfied, the design configuration relevant to the proposed work is stable, and the subsequent site falls within the approved site-parameter envelope. Full reference-plant operation could remain a sufficient pathway without being the only one.

Generic finality should also extend to standardized operational programs where those programs are sufficiently mature to permit generic NRC review. As BTI recommended in its Part 57 comment, a manufacturing license should be capable of resolving operational-program elements with meaningful finality so that later applicants referencing the standardized design can rely on those approvals rather than resubmit the same program for substantive review.⁴ The latter applicant would remain responsible for demonstrating that the approved program, including the assumptions, conditions, and limitations underlying the NRC's determination, remains applicable to the proposed facility and for addressing any material applicant-specific, site-specific, or operational differences. The same principle should apply wherever an operational program is sufficiently standardized to support generic resolution and the later applicant

⁴ Breakthrough Institute, Comment on Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles, Docket ID NRC-2025-0379, Comment ID NRC-2025-0379-0123, June 15, 2026, <https://www.regulations.gov/comment/NRC-2025-0379-0123>.

demonstrates that the approved basis remains applicable, whether or not the later deployment proceeds under the same licensing pathway. Prior findings under Parts 50, 52, 53, or 57 should remain available to a later applicant where the underlying determination is technically applicable and the applicant satisfies any requirements unique to its licensing pathway. Changing licensing pathways should not itself require reconsideration of an otherwise applicable generic determination. Generic finality should attach to the matter the NRC has actually resolved, while subsequent review should focus on differences that could affect the continued applicability of that determination.

Implementation guidance should also address shared SSCs at multi-module sites. A shared SSC may serve modules operating or proceeding under different licensing statuses. The NRC should identify which authorization governs work on the shared SSC and how applicants should evaluate activities affecting multiple modules. The NRC should also provide an applicant-facing pathway map showing how design approval, generic finality, reference-plant verification, docketing, general-license eligibility, inspections, and subsequent CP or COL review are sequenced.

Responses to Specific Requests for Comment

Question 1. Should the NRC consider granting one or more general licenses to begin construction activities as part of an existing or future standard design certification rule? If the NRC were to issue such a general license, should the NRC require that the general license holder reference another plant that has completed construction (resulting in the general license excluding first of a kind applicants)? If not, what conditions should the NRC include in the general license to ensure that the construction activities authorized by the general license can be completed safely? Please provide a basis for your response.

Yes, with conditions. The NRC should allow an existing or future standard design certification to provide the generic design basis for a general construction license, but design certification alone should not make the general license available for an unconstructed FOAK design. A design certification resolves design issues generically; it does not by itself demonstrate that the relevant design features can be constructed as approved.

The NRC also should not require, as the exclusive eligibility criterion, that another plant using the design have completed its entire construction and licensing sequence. Instead, the final rule should provide an alternative eligibility route once the portions of the reference design relevant to the proposed construction have been constructed and inspected under NRC oversight,

applicable acceptance or verification criteria have been satisfied, the relevant design configuration is stable, and the subsequent site falls within the approved site-parameter envelope. Full completion of the reference plant could remain a sufficient basis for eligibility without being the only one. This approach would preserve the general license principally as a NOAK tool while allowing its use when the relevant construction maturity has actually been demonstrated.

Question 2. Would it be useful for the NRC to issue regulations to provide for general licenses related to production and utilization facilities for other activities described under section 109 of the AEA, with the exception of import, export, construction, or operation? Please provide a basis for your response.

BTI does not identify an additional activity in this rulemaking for which a new general license under AEA § 109 is presently necessary. The NRC should first implement the proposed construction general license and develop experience with its use. If recurring classes of other activities later emerge for which the NRC can define a bounded class, objective eligibility criteria, and generally applicable safety and security conditions, a general license may be appropriate. The agency should not create additional general licenses merely for procedural symmetry where the regulatory record has not identified a recurring activity that would benefit from generic authorization.

Question 3. Does the proposed regulatory change achieve its intended objective of allowing applicants to make business decisions on the appropriate balance between expedited construction and the regulatory risk that changes will be necessary to address safety issues? Are there other approaches that can both provide the needed flexibility while better allowing applicants to structure the degree of regulatory risk they take on through interactions with the NRC staff (e.g., hold-points during construction for NRC inspection)?

Yes. The proposed rule appropriately allows an applicant to decide whether the schedule benefits of beginning construction justify the financial and licensing risk that later NRC findings may require modification or rework. That allocation of risk should be preserved.

The NRC should not, however, routinely require construction hold points that function as additional prior approvals for work already authorized by the general license. Applicants should be able to use targeted NRC inspection or verification milestones where doing so would reduce regulatory uncertainty or demonstrate satisfaction of an eligibility criterion, including the alternative design-maturity shown discussed above. Such milestones should address defined safety or verification issues rather than recreate an LWA-type approval process within the general

license. This would preserve applicant flexibility while providing a means for applicants that prefer less regulatory risk to obtain earlier NRC verification.

Recommendations

1. **Adopt a consistent, function-based definition of construction across reactor licensing pathways.** BTI recommends that the NRC finalize the proposed construction and repeat-deployment reforms while making targeted changes to ensure that they operate as a coherent framework across licensing pathways. The final rule should retain the proposed shift away from categorical treatment of onsite activities and define construction by whether the activity may affect an SSC's ability to perform a safety or other credited regulatory function. Sections 50.10, 51.4, and 53.020 should implement that substantive principle consistently, while allowing differences where required by the distinct statutory or safety functions of the applicable framework. The NRC should coordinate those provisions with Parts 57 and 70 so that materially different construction or preconstruction boundaries exist only where there is a regulatory reason for the difference. Implementing guidance should also establish or endorse a reusable SSC-classification methodology for applying the definition to dual-function, shared, and multi-module SSCs.
2. **Tie construction-permit information requirements to the findings required at that licensing stage to provide licensing flexibility.** The NRC should finalize the proposed § 50.34 revisions and make conforming changes to § 50.67 and related Part 52 provisions so that construction-permit application requirements remain technology-inclusive and are tied to the findings the NRC must make before authorizing construction. The NRC should apply the same principle to § 53.1309: information required in a Part 53 CP application should reach the level of maturity necessary to support the applicable CP-stage findings, rather than standard-design-certification maturity unless that additional detail is needed for a determination that must be made before construction. Where the NRC has already made an equivalent generic finding under Part 50, 52, or 53, that finding should also remain usable in a later Part 57 proceeding when technically applicable and subject to Part 57's additional requirements.
3. **Make generic finality and the general construction license usable for repeat deployment.** The NRC should retain the proposed general construction license, applicant-at-risk structure, and generic-finality provisions, but should not make full operation of a

reference plant the exclusive means of demonstrating readiness for repeat deployment. The final rule should permit an alternative eligibility showing where the relevant portions of the reference design have been constructed and inspected under NRC oversight, applicable acceptance or verification criteria have been satisfied, the relevant design configuration is stable, and the subsequent site falls within the approved site-parameter envelope. The NRC should also allow standardized operational programs to obtain generic finality through a manufacturing license and be referenced by later applicants where the approved basis, assumptions, conditions, and limitations remain applicable. The same principle should apply to other sufficiently standardized operational programs and generic determinations, including across Parts 50, 52, 53, and 57.

III. DESIGN DEPARTURES AND CHANGE CONTROL DURING CONSTRUCTION

BTI supports the NRC's broader effort to establish a more performance-based and technology-inclusive framework for regulating advanced reactors and changes to approved designs. These reforms should be implemented as an integrated framework. Within that framework, the final rule should apply its change-control principles consistently across reactor technologies and focus regulatory attention on the safety significance of a proposed change, rather than on design-specific categories, historical classifications, or the NRC's prior construction experience with a particular technology. The framework should clearly identify when prior NRC approval is required, preserve appropriate regulatory controls for changes that could affect safety, and provide sufficient flexibility for licensees to implement lower-risk changes efficiently during construction.

A. Significance-Based Design Change Control

BTI supports the NRC's proposed shift toward change control requirements based on the significance of a departure rather than the regulatory category in which the affected information was placed. In particular, BTI supports the proposed § 50.59-like process for Tier 1 design-description changes, removal of standardization as an independent change control

criterion, elimination of duplicative exemptions for ITAAC changes, and more flexible treatment of AP1000 Tier 2* information.

The proposed treatment of Tier 2* information, however, does not fully carry that principle through in a technology-inclusive principle. The rule would effectively treat AP1000 Tier 2* information as ordinary Tier 2 information during construction, while continuing to require prior NRC approval for comparable ABWR and ESBWR Tier 2* departures until the plant first achieves full power. The NRC bases that distinction primarily on the construction and licensing experience available for the AP1000. The Vogtle Units 3 and 4 experience illustrates that Tier 2* changes of minimal safety significance that nevertheless required license amendments, and Southern Nuclear ultimately obtained approval to use a Tier 2-like significance-based process for certain Tier 2* departures.⁵

However, a change control threshold should instead depend on the significance of the proposed change, not on whether a particular certified design has accumulated construction experience. The AP1000 experience supports applying that principle more broadly, not limiting the resulting flexibility to the AP1000. The NRC now recognizes that some information designated Tier 2* was of insufficient safety significance to warrant that treatment, and later design certifications for the APR1400 and NuScale designs contain no Tier 2* information at all.⁶ The proposed rule also adopts the principle that Tier 2* information should qualify for Tier 1-level treatment and that information of minimal safety significance should not be placed in categories that automatically require prior approval.

The final rule should apply the Tier 2 departure process to ABWR and ESBWR Tier 2* information during construction, same as the proposed change to AP1000. Particular Tier 2* matters could remain subject to prior NRC approval where the NRC identifies a design-specific technical basis showing that the ordinary departure criteria would not provide adequate regulatory oversight of safety-significant design changes. That approach would preserve heightened review where

⁵ Nuclear Regulatory Commission, *Southern Nuclear Operating Company, Inc., Vogtle Electric Generating Plant, Units 3 and 4; Changes to Tier 2 Departure Evaluation Process*, 83 FR 54622 (October 30, 2018).
<https://www.federalregister.gov/documents/2018/10/30/2018-23627/southern-nuclear-operating-company-inc-vogtle-electric-generating-plant-units-3-and-4-changes-to>.

⁶ Nuclear Regulatory Commission, *Modernizing Reactor Licensing, Safety Oversight, and Siting Practices*, Proposed Rule, Docket ID NRC-2025-0975, RIN 3150-AL44, 91 Fed. Reg. 44560, 44582–83 (July 16, 2026), [ML26181A098](#).

technically justified while making the change control framework technology-inclusive and consistent with the rule's broader shift toward significance-based regulation.

Responses to Specific Requests for Comment

Question 7. Will there be adequate regulatory clarity regarding the category of Tier 1 information in the generic DCD such that the change control process can be readily identified?

BTI agrees that the proposed revisions would provide substantially greater regulatory clarity regarding the categories of Tier 1 information. By identifying the specific generic DCD sections and tables corresponding to definitions and general provisions, design descriptions, ITAAC, significant site parameters, and significant interface requirements, Section II.D would allow the applicable change process in Section VIII to be identified directly from the regulatory appendix. The proposal further clarifies the treatment of referenced figures and non-ITAAC tables within design descriptions and specifies the portions of ITAAC tables that constitute ITAAC information.

The NRC should finalize this approach and clarify that the category assignments in Section II.D are for the purposes of applying the Section VIII change processes. If information is referenced from multiple sections or could reasonably fall within more than one Tier 1 category, the appendix should make clear which category and change process governs. This would preserve the clarity created by the proposed mapping and avoid case-by-case disputes over classification.

Question 8. Should the flexibility provided to the AP1000 design in appendix D during construction also be extended to the ABWR and ESBWR designs in appendices A and E?

Yes. The NRC should extend the AP1000 Tier 2* flexibility to ABWR and ESBWR unless it identifies particular information for which a design-specific technical basis warrants prior approval. The applicable change control process should depend on the significance of the departure, not solely on whether the NRC has prior construction experience with the design.

B. Changes During Construction

The NRC should also make Part 52 and Part 53 consistent for changes implemented at risk during construction. The proposed 53.1535(b) expressly allows a COL holder, before the § 53.1452(g) finding, to begin construction implementing a change that requires NRC approval and submit

the required amendment or exemption within 45 days, with the licensee bearing the risk that the NRC may ultimately reject the change. Part 52 should provide the same express option. There is no clear reason for otherwise comparable COL holders to face different sequencing rules merely because one project is licensed under Part 52 and another under Part 53. The NRC can preserve its authority over changes requiring approval without making prior filing a mandatory construction hold point.

The same significance-and-reuse principle should govern changes the NRC has already approved. The proposal would allow an OL or COL referencing a manufacturing license to implement an FSAR change without obtaining another license amendment when the change is identical to one previously approved through an amendment to the manufacturing license and the basis for the NRC's approval remains applicable. The NRC should provide equivalent treatment for COLs referencing a design certification. A subsequent licensee should still demonstrate that the previously approved change is technically applicable to its plant and does not introduce another plant-specific change requiring approval, but the NRC should not repeat the underlying technical review solely because the same approved change is being implemented at another plant.

The NRC should also preserve clear Tier 1 categorization in the final design-certification appendices. Because the proposal assigns different change control processes to design descriptions, ITAAC, significant site parameters, interface requirements, and other Tier 1 information, each item should be traceable to its applicable category and change process without requiring case-by-case staff interpretation. The proposed identification of relevant DCD sections and tables is useful and should be retained.

Responses to Specific Requests for Comment

Question 9. Should the provisions in 10 CFR part 52 be revised to also explicitly support the change process reflected in § 53.1535(b)?

Yes. Part 52 should expressly permit the same at-risk construction-change process provided in § 53.1535(b), with the licensee bearing the risk that the NRC may deny the required amendment or exemption.

Question 10. Should the flexibility allowing OL or COL holders referencing a manufacturing license to implement an identical NRC-approved ML change without another license amendment also be provided to COLs that reference a design certification?

Yes. A COL holder referencing a design certification should be able to implement an identical NRC-approved change without duplicative plant-specific approval where the basis for the prior approval remains applicable, and no additional change independently requires a license amendment.

Recommendations

1. **Extend significance-based Tier 2 change control to ABWR and ESBWR unless a design-specific technical basis warrants prior approval.** The NRC should finalize the proposed Tier 1 design-description process, elimination of standardization as an independent change-control criterion, removal of the duplicative exemption requirement for ITAAC changes, and more flexible treatment of AP1000 Tier 2* information, and apply the same significance-based principle to comparable ABWR and ESBWR Tier 2* departures.
2. **Allow Part 52 licensees to implement changes at risk under the same bounded process available under Part 53.** Part 52 should expressly permit the process in § 53.1535(b), including the 45-day filing period and allocation of implementation risk to the licensee, rather than making prior filing a construction hold point solely because the project is licensed under Part 52.
3. **Allow subsequent licensees to reuse previously approved design-certification and manufacturing-license changes when the prior technical basis remains applicable.** The later licensee should demonstrate applicability and address any plant-specific difference that independently requires approval, but the NRC should not repeat the underlying technical review solely because the same approved change is being implemented at another plant.

IV. QUALITY ASSURANCE AND CODES AND STANDARDS

BTI supports the NRC's proposed Appendix T as a voluntary, graded, performance-based alternative to Appendix B and supports the proposed expansion of § 50.55a(z) to reduce unnecessary reliance on exemptions. Appendix T appropriately directs quality-assurance controls

toward the safety and risk significance of structures, systems, and components (SSCs), while the 50.55a(z) revisions retain established quality-and-safety findings while making the alternative process available across the section.

The principal concern is the conditions the proposed rule places on their use. Appendix T would make graded quality assurance available only after a first-of-a-kind (FOAK) plant has already been designed, constructed, and operated under the existing framework, and the proposed change control provisions would require prior NRC approval for every change to an approved Quality Management System (QMS). Those restrictions risk postponing the principal benefits of the new framework until after applicants have already incurred the supply-chain and administrative costs the framework is intended to reduce. The final rule should instead tie eligibility and change control to the maturity, safety significance, and regulatory consequence of the activity at issue.

A. Appendix T Should Not Be Categorically Limited to NOAK Plants

The NRC should retain Appendix T's graded structure but revise Section I so that first-of-a-kind plants may use Appendix T when the applicant demonstrates sufficient design maturity and quality-program maturity, with additional oversight where remaining uncertainty warrants it.

As proposed, Appendix T is available only to an nth-of-a-kind (NOAK) applicant that identifies a FOAK reference plant. Departures from that reference plant may not change the classification, design, or methods of manufacture, construction, and operation of SSCs within Appendix T's scope, and the relevant procedures and work processes must already be established. The NRC defines a NOAK plant for these purposes as a subsequent implementation after the FOAK plant has been designed, constructed, and operated.

The NRC's rationale is that a completed FOAK provides mature design information, established manufacturing and construction methods, operational experience, known suppliers, and demonstrated implementation processes.⁷ The NRC also explains that FOAK licensing and construction may require vendor inspections because detailed design and QA procedures are not always sufficiently developed during the licensing review, whereas those inspections may be less

⁷ Nuclear Regulatory Commission, *Modernizing Reactor Licensing, Safety Oversight, and Siting Practices*, Proposed Rule, Docket ID NRC-2025-0975, RIN 3150-AL44, 91 Fed. Reg. 44560, 44576—77 (July 16, 2026), [ML26181A098](#).

necessary for a repeated design and previously inspected suppliers. Those considerations support additional guardrails for less mature applications. They do not require a categorical rule that graded QA becomes available only after an entire reactor design has already progressed through design, construction, and operation.

NRC must have reasonable assurance that the SSCs within the graded QA program will perform their required safety functions. “FOAK” and “NOAK” are proxies for the information available to make that determination. They are not themselves measures of quality assurance. A FOAK applicant may have mature SSC designs, qualified suppliers, completed procedures, extensive component testing, manufacturing experience, and an auditable QMS before the plant enters operation. Conversely, a nominally NOAK project may introduce a new supplier, manufacturing process, material, software platform, or other departure that requires additional oversight even though the project satisfies the rule’s formal NOAK definition.

The final rule should regulate those differences directly. A more durable eligibility framework would allow Appendix T where the applicant demonstrates that the SSCs within its scope are sufficiently mature for graded QA treatment. That showing should establish that the SSCs’ safety classifications and required functions are defined; the design information needed to provide sufficient oversight for the manufacture, construction, inspection, and testing is sufficiently developed; the QMS procedures and work processes needed to implement Appendix T are in place; and relevant suppliers and manufacturing processes have been appropriately qualified. Where material FOAK uncertainty remains, the NRC could address it through additional oversight commensurate with the significance of that uncertainty rather than by categorically requiring completion and operation of a reference plant.

For a FOAK plant, that could mean retaining vendor inspections, inspection hold points, targeted audits, enhanced supplier oversight, or other verification until the relevant processes have demonstrated satisfactory performance. For a later plant using an established supplier and unchanged process, those additional controls could be reduced. A new supplier introduced on the tenth plant should not automatically receive less scrutiny than a mature supplier on the first solely because one project is labeled NOAK and the other FOAK. This approach would preserve the safety rationale underlying the NRC’s proposal while avoiding an unnecessarily binary eligibility rule.

The timing problem is also important. If a developer must first establish an Appendix B program, qualify its supply chain under that program, construct the FOAK plant, and place it into operation before Appendix T becomes available, much of the institutional and commercial lock-in will already have occurred. Suppliers will have developed Appendix B- or NQA-1-based processes, procurement specifications will have been written around them, vendor qualification will have occurred, and applicant procedures will have been built around that system. Changing the QA framework only after that infrastructure is established imposes transition costs and reduces the incentive to take advantage of the more flexible framework.

That result would weaken one of Appendix T's stated purposes. The NRC describes Appendix T as a means of incorporating terminology and methodologies used across safety-critical industries and international quality-management standards so applicants can make greater use of cross-industry and global supply chains. It also expressly identifies reduced supplier and vendor oversight, where appropriate, as a benefit. A framework that cannot be used until the initial nuclear supply chain has already been established under the legacy system addresses the problem only after a substantial portion of the relevant cost has been incurred.

In the July 2026 ACRS meeting, the staff stated that Appendix T has been mapped with and fully supports all attributes of quality described in Appendix B.⁸ In its letter on the proposed rule, the ACRS noted that staff had mapped Appendix T to the attributes of quality in Appendix B and identified additional benefits of the proposed framework, including its treatment of software quality assurance.⁹ The ACRS recommended reconsidering the NOAK entrance condition so that Appendix T could be available as a viable option for FOAK applicants. It also observed that allowing FOAK use could avoid inconsistencies for operators and suppliers that might otherwise need to maintain different quality-assurance programs for the FOAK and subsequent NOAK applications. That recommendation reinforces the case for regulating the underlying maturity and quality-assurance considerations directly rather than using NOAK status as an absolute prerequisite. A FOAK applicant that can demonstrate sufficient design, QMS, supplier, and implementation maturity should be able to use Appendix T, with additional NRC oversight where

⁸ U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, *Transcript of Meeting on Reactor Licensing, Safety Oversight, and Siting Practices Related to Executive Order 14300*, July 8, 2026, ML26197A187, <https://www.nrc.gov/docs/ML2619/ML26197A187.pdf>.

⁹ Advisory Committee on Reactor Safeguards, Letter to Hon. Ho K. Nieh, Chairman, U.S. Nuclear Regulatory Commission, *Proposed Rulemaking on Modernizing Reactor Licensing, Safety Oversight, and Siting Practices (RIN 3150-AL44)*, July 24, 2026, [ML26205A160](#).

remaining uncertainty warrants it. Thus, Appendix T should be applicable to FOAK plants without reduction in safety. A potential pitfall of the draft rule is that if any design changes are made on a NOAK plant for the purposes of efficiency and innovation, the updated design could be reclassified as FOAK, and the developer would have to go through Appendix B once again. Such an outcome, or potential for said outcome, would remove the incentive for developers to make improvements and innovations in their designs even if those improvements would contribute materially to safety and cost reduction. Therefore, the NRC must define a way to move from Appendix B to Appendix T, even if there are small changes in design. One potential avenue for the FOAK Appendix B QA program to the NOAK Appendix T QA program is the idea of the “same core design,” meaning that a reactor design keeps its NOAK status throughout small innovations as long as the core design remains functionally the same. Another option is to drop the requirement of “NOAK” from Appendix T.

The NRC should also revise proposed Section I.B so that justified design and manufacturing improvements do not automatically eliminate Appendix T eligibility. As drafted, a NOAK applicant may use Appendix T only if departures from the FOAK reference plant do not change the classification, design, or method of manufacture, construction, or operation of covered SSCs. That restriction could discourage improvements made during repeat deployment. The better approach is to determine whether a departure materially changes the basis for the graded QA treatment. A change that does not alter the SSC’s safety classification or required function, or otherwise undermine the QMS, supplier qualification, manufacturing controls, or other basis supporting Appendix T eligibility, should not by itself force the applicant back to Appendix B. This would preserve appropriate control over meaningful departures without discouraging iterative design and manufacturing improvements.

The NRC should revise Appendix T Section I to permit use by FOAK plants and existing licensees based on demonstrated design and QMS maturity and risk-appropriate oversight. NOAK status and prior operating experience can remain relevant evidence supporting reduced oversight, but they should not be categorical prerequisites. The NRC should also permit justified departures from a reference design where the applicant demonstrates that the change does not materially undermine the basis for Appendix T treatment.

B. The NRC Should Establish Technology-Neutral Acceptance Criteria for Alternative QA Frameworks

The NRC should define the substantive criteria an acceptable QA framework must satisfy and allow applicants to demonstrate how Appendix T, an endorsed consensus standard, or another framework meets those criteria.

Appendix T already provides a useful foundation for this approach. Its requirements address management, performance, assessment, software used in design and analysis, and digital items important to safety, while directing applicants to grade QA controls according to the safety and risk significance of affected SSCs. The regulatory objective should be reasonable assurance that SSCs will perform their required safety functions, supported by QA controls commensurate with their safety significance.

The NRC should identify the substantive functions necessary to make that showing, including appropriate control of design, procurement and suppliers, inspection and testing, corrective action, records, and independent assessment. The required rigor can vary with safety significance, uncertainty, novelty, and supplier performance, but the applicant should be able to identify the applicable criteria and demonstrate how its selected QA framework satisfies them.

BTI took the same approach in its materials-licensing comment.¹⁰ There, BTI recommended retaining Appendix B as an accepted means of compliance while allowing Appendix T, applicable consensus standards, or applicant-developed alternatives where they provide equivalent reasonable assurance. That comment identified design control, supplier oversight, inspection and testing, corrective action, records, availability of credited functions, and independent assessment as the substantive functions that an alternative program should address, with rigor graded to safety significance. The NRC should apply the same principle here.

This would also avoid making Appendix B the de facto requirement simply because it is the framework with the most established review precedent. A nominally flexible rule provides little practical flexibility if applicants cannot predict what showing will be sufficient for a different program. Part 57 illustrates the problem. It permits tailored QA programs rather than mandating Appendix B, but BTI previously identified the need for implementation guidance specifying how

¹⁰ Breakthrough Institute, *Comment on Modernizing Materials Licensing*, Docket ID NRC-2025-1370, Comment ID NRC-2025-1370-0038, August 10, 2026, <https://www.regulations.gov/comment/NRC-2025-1370-0038>.

the NRC will evaluate those programs, including the expected rigor at different levels of safety significance.¹¹ Without such criteria, applicants and reviewers may converge informally on the familiar framework even where the regulation nominally permits something else.

The NRC should not automatically deem ISO 19443 or another consensus standard equivalent to Appendix B or Appendix T. Nor should a framework be rejected solely because it is organized differently from Appendix B. The NRC should evaluate whether the standard, together with any necessary nuclear-specific supplements or applicant commitments, satisfies the substantive QA criteria required for the SSCs and activities within its scope. Where a consensus standard repeatedly satisfies those criteria, NRC endorsement could provide a predictable pathway for subsequent applicants.

The final rule and implementing guidance should establish technology-neutral acceptance criteria for QA and explain how applicants may demonstrate that Appendix T, an NRC-endorsed consensus standard, or another QA framework satisfies them. Appendix B should remain an established compliance pathway, but historical familiarity with Appendix B should not determine whether another framework is acceptable.

Responses to Specific Requests for Comment

Question 39. Appendix T of 10 CFR is intended to address nuclear supply chain constraints by providing a graded approach for additional QA flexibilities and aligning with international standards. The proposed rule limits the applicability of appendix T to nth-of-a-kind (NOAK) plants referencing a first-of-a-kind (FOAK) plant constructed and operated under appendix B. How can the NRC expand the applicability of appendix T to FOAK plants and existing licensees, including any guardrails or restrictions that may be desirable (e.g., vendor inspections for FOAK and new suppliers for NOAK)? Should the NRC consider alternative approaches to appendix T such as endorsement of ISO 19443 or changes to appendix B? Please provide the basis for your recommendations, including any unintended consequences

BTI recommends expanding Appendix T beyond NOAK plants. BTI does not recommend revising Appendix B as the principal means of creating this flexibility; Appendix T is the better vehicle for

¹¹ The Breakthrough Institute, *Comment on Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles*, Docket ID NRC-2025-0379, Comment ID NRC-2025-0379-0123, June 15, 2026, <https://www.regulations.gov/comment/NRC-2025-0379-0123>.

a distinct graded pathway while leaving Appendix B available for applicants and licensees that choose the established framework. FOAK applicants and existing licensees should be permitted to use Appendix T where they can demonstrate sufficient design, QMS, supplier, and implementation maturity to support graded quality assurance. Remaining FOAK uncertainty can be addressed through targeted vendor inspections, supplier oversight, hold points, or other additional NRC verification rather than through a categorical requirement that a reference plant first be constructed and operated.

The NRC should not automatically treat ISO 19443 or another consensus standard as equivalent to Appendix B or Appendix T. Instead, it should define the safety-assurance functions an acceptable QA framework must perform and allow applicants to demonstrate how their selected framework satisfies those functions. Appendix B should remain an available compliance pathway, but Appendix T should be usable when an applicant can substantively demonstrate readiness to implement it.

C. QMS Change Control Should Be Graded

The NRC should revise proposed §§ 50.54(a)(5), 50.55(f)(5), and the corresponding Part 53 provisions so that prior NRC approval is required for changes that materially reduce approved QA commitments or otherwise affect the basis for the NRC's safety finding, rather than for every change to an approved QMS.

The proposed rule currently takes a broader approach. Proposed §§ 50.54(a)(5) and 50.55(f)(5) would require QMS changes to be submitted to the NRC and approved before implementation. The proposed Part 53 conforming amendments do the same: proposed § 53.1565(d)(1)(iii) and (d)(2)(ii) would require QMS changes to be submitted and approved before implementation for applicable Part 53 license and permit holders. That requirement sits uneasily with the purpose of a graded, performance-based QA system.

A QMS is necessarily a living management system. Procedures change as organizations correct deficiencies, adopt improved processes, change suppliers, update software, reorganize responsibilities, incorporate operating experience, or improve inspection and assessment methods. Not every such change reduces safety assurance. Some increase it. Others are administrative or affect activities of relatively low safety significance. Requiring prior NRC

approval for every QMS change gives all such changes the same regulatory consequence which is the opposite of grading.

It could also create a perverse incentive. If improving a procedure, reorganizing a process, or adopting a more effective control requires a prior approval cycle simply because the change alters the approved QMS, licensees have a reason to minimize changes to the program even where experience indicates that a change would improve performance. A performance-based framework should encourage correction and continuous improvement while reserving prior approval for changes that alter the basis on which the NRC accepted the program.

The NRC itself identifies the better alternatives in Question 40, which asks whether the change process should be graded or should mirror § 50.54(a)(4), under which prior approval is focused on changes that reduce commitments. BTI recommends a graded QMS change process. Prior NRC approval should be required when a change reduces an approved QA commitment, weakens controls for safety-related or safety-significant SSCs, lowers the QA treatment applied to an SSC, or materially alters the basis for supplier qualification, inspection, testing, independent assessment, corrective action, or another element that supported NRC approval of the QMS. Changes that do not reduce the level of assurance should be handled under the licensee's approved change process, documented, and available for NRC inspection.

This is particularly important for Part 53. Part 53 expressly uses flexible and graded treatment tied to the safety significance of SSCs, and the final Part 53 rule does not incorporate the fixed codes-and-standards model used in § 50.55a. Instead, it allows generally accepted standards to be tailored to the assessed safety significance of SSCs. Importing a categorical prior-approval requirement for all Appendix T QMS changes would make one of Part 53's principal performance-based programs less flexible than the broader framework in which it operates. The NRC should preserve Appendix T's substantive controls while making its change process consistent with the same graded regulatory logic.

Responses to Specific Requests for Comment

Question 40. The change process for the Quality Management System (QMS) in proposed 10 CFR 50.54(a)(5) requires changes to the QMS be submitted to the NRC for NRC approval prior to implementation. Should the change process have a graded approach or mirror 10 CFR 50.54(a)(4) in

which only changes that reduce the commitments be submitted to the NRC for prior approval? Please explain if there is a preferred method.

BTI recommends a graded QMS change process modeled on the principle in § 50.54(a)(4). Prior NRC approval should be required when a change reduces an approved commitment or otherwise materially weakens the basis for the NRC's QA finding, including changes that reduce oversights for safety-related or safety-significant SSCs or materially alter supplier qualification, inspection, testing, corrective action, or independent assessment.

Changes that maintain or increase the level of assurance should be controlled under the approved QMS, documented, and available for NRC inspection without prior approval. This approach better aligns Appendix T with its graded, performance-based structure and avoids creating an unnecessarily prescriptive change control regime, particularly for Part 53.

D. Preserve the § 50.55a(z) Expansion and Make Approved Alternatives Reusable

BTI supports expanding § 50.55a(z) to all requirements in § 50.55a. The existing limitation has produced exemption requests where the established § 50.55a(z) criteria would already provide an adequate safety finding; removing that artificial boundary preserves the requirement to demonstrate an acceptable level of quality and safety, or hardship without a compensating increase in quality and safety, while avoiding unnecessary use of § 50.12.

The NRC should go one step further and ensure that a technically identical alternative need not be re-established from the beginning each time it is used.

The value of an alternative pathway increases when regulatory learning accumulates. The first applicant using an alternative may appropriately bear the burden of establishing its technical basis and the conditions necessary for NRC approval. Once the NRC has made that determination, however, later applicants proposing the same alternative under materially equivalent circumstances should be able to rely on the prior approval rather than recreate the full demonstration.

Later applicants would still need to show that the earlier technical basis remains applicable to their proposed use and identify any material differences that could affect the NRC's prior finding.

Where those differences exist, the NRC should focus its review on them rather than reopen issues that have already been resolved.

This principle is consistent with the direction already emerging elsewhere in the NRC's modernization effort. Part 53 does not incorporate specific § 50.55a codes and instead permits generally accepted codes and standards to be tailored to the assessed safety significance of SSCs.¹² Proposed Part 57 similarly allows applicants to select appropriate consensus standards rather than incorporating § 50.55a's specific standards, but BTI identified an implementation problem if every selected standard must be reconsidered independently in every subsequent licensing proceeding.¹³ BTI recommended that the NRC explain how previously approved code selections can be relied upon in later applications and how those determinations interact with manufacturing-license and generic-finality mechanisms.

For Part 50 and Part 52 facilities governed by § 50.55a, the final rule or implementation guidance should establish a process by which an applicant or licensee may reference a previously approved alternative and demonstrate that the technical basis, conditions, and limitations remain applicable. The NRC should retain authority to review plant-specific differences and should require a new request where those differences are material.

For Part 53, the NRC should preserve the technology-inclusive codes-and-standards framework already established in § 53.440(b) while applying the same principle of regulatory reuse. Part 53 does not incorporate the specific codes and standards in § 50.55a and instead permits the use of generally accepted consensus standards that the NRC has endorsed or otherwise found acceptable. Where the NRC has already accepted a code or standard for a particular function and set of service conditions, a later Part 53 applicant should be able to reference that determination and demonstrate that its technical basis, conditions, and limitations remain applicable. The staff's review should then focus on material differences rather than repeat the underlying determination of the standard's acceptability.

¹² Nuclear Regulatory Commission, *Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors*, Final Rule, Docket ID NRC-2019-0062, RIN 3150-AK31, 91 Fed. Reg. 15696, 15700 (Mar. 30, 2026), <https://www.govinfo.gov/content/pkg/FR-2026-03-30/pdf/2026-06048.pdf>.

¹³ The Breakthrough Institute, *Comment on Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles*, Docket ID NRC-2025-0379, Comment ID NRC-2025-0379-0123, June 15, 2026, <https://www.regulations.gov/comment/NRC-2025-0379-0123>.

The better organizing principle across Parts 50, 52, 53, and 57 is that applicants must justify the codes and standards appropriate to the SSC, its required function, its safety significance, and the service conditions it must withstand. Where the NRC has already accepted the same standard for the same function under materially equivalent conditions, later reviews should be able to credit that determination.

Recommendations

1. **The NRC should base Appendix T eligibility on demonstrated design and QMS maturity rather than categorical NOAK status.** FOAK applicants should be permitted to use Appendix T with additional oversight where remaining uncertainty warrants it.
2. **The NRC should establish technology-neutral acceptance criteria for alternative QA frameworks.** Appendix B should remain an available compliance pathway, while Appendix T, appropriate consensus standards, and other frameworks should be acceptable where they provide equivalent reasonable assurance.
3. **The NRC should grade QMS change control under §§ 50.54, 50.55, and 53.1565 according to the significance of the change.** Prior NRC approval should be required for reductions in approved commitments or other changes that materially affect the basis for the NRC's QA finding.
4. **The NRC should preserve the proposed expansion of § 50.55a(z) and facilitate reuse of previously approved alternatives.** Later applicants should be able to rely on prior NRC determinations where the technical basis, conditions, and limitations remain applicable, while Part 53 should retain its technology-inclusive approach to codes and standards.

V. RISK-INFORMED AND PERFORMANCE-BASED ACCEPTANCE CRITERIA AND ANALYTICAL METHODS

BTI supports the NRC's proposed expansion of risk-informed and performance-based decision-making in Parts 50 and 52. New §§ 50.220 and 52.220 provide a structured means of replacing unnecessarily conservative acceptance criteria without relying on exemptions, while the revisions to § 50.59 and new § 50.221 would allow greater use of quantitative risk information and modern analytical methods. These reforms can reduce unnecessary regulatory burden while preserving the safety functions that the existing requirements are intended to protect.

The reforms should nevertheless be implemented with recognition of the architecture of Parts 50 and 52. Those frameworks were developed principally through deterministic requirements, prescribed acceptance criteria, and overlapping layers of protection, and have been progressively risk-informed over time. Unlike Part 53, they were not originally constructed around an integrated risk-informed safety framework. As a result, an individual deterministic criterion may be demonstrably overconservative and appropriately replaced without necessarily providing a clear system-level benchmark for how many such changes, taken together, remain consistent with the safety assurance provided by the existing framework.

DG-1464 recognizes this issue by requiring proposals that seek alternatives to more than one regulation to address their cumulative impact and by cautioning against using §§ 50.220 and 52.220 to reconstruct a substantial portion of a plant's licensing basis. The final guidance should preserve that discipline. Where multiple alternatives interact, the NRC should consider their combined effects on risk, defense-in-depth, safety margins, uncertainty, and performance monitoring rather than rely solely on separate findings that each change is acceptable in isolation. The NRC should apply the same principle when reviewing the combined effects of the multiple risk-informed reforms adopted through this rulemaking.

This does not counsel against continued risk-informing of Parts 50 and 52. It instead distinguishes between correcting individual requirements that no longer represent an appropriate measure of safety and using successive alternatives as a substitute for addressing structural problems in the underlying framework. Where the same prescriptive requirement repeatedly necessitates alternative treatment, generic rulemaking or guidance may provide a more durable solution; however, the tool provided in 50.220 should not be used to modernize Part 50 and 52 in a piecemeal fashion. Part 53 should remain the integrated technology-inclusive pathway for applicants seeking a fundamentally different licensing architecture.

Against that background, the principal implementation question is whether the new pathways will produce durable and transferable regulatory flexibility. The final rule and guidance should distinguish between the technical demonstration necessary to establish a new alternative or analytical framework and the narrower showing necessary when a later applicant or licensee relies on a determination the NRC has already made.

A. Previously Approved Alternative Acceptance Criteria Should Be Reusable

BTI supports proposed §§ 50.220 and 52.220. The provisions appropriately create a voluntary pathway for applicants and licensees to propose risk-informed or performance-based alternatives to prescriptive acceptance criteria without satisfying the special-circumstances requirement for an exemption. At the same time, the proposal retains substantial safety guardrails: an alternative must provide an equivalent level of safety, maintain defense-in-depth and safety margins, account for relevant operating experience, include performance monitoring, and address cumulative effects where multiple alternatives are used.

The NRC should preserve those requirements while allowing greater reliance on alternatives it has already approved.

The proposed rule currently requires an applicant or licensee using § 50.220 to include the alternative in an initial application or submit a license amendment request. That approach is appropriate when the NRC is evaluating a new alternative for the first time. The first proponent should establish the technical basis for the alternative, demonstrate that the risk-informed criteria in § 50.220 are satisfied, and identify the conditions and limitations necessary to maintain the NRC's safety finding.

Once the NRC has completed that review, a later applicant proposing the same alternative under materially equivalent circumstances should not have to recreate the original demonstration. The later applicant should instead establish that the approved technical basis, conditions, limitations, and performance assumptions remain applicable to its facility. Any material difference that could affect the prior safety finding would remain subject to NRC review, but the NRC should review that difference rather than reconsider the alternative from the beginning.

The proposed rule already anticipates that prior approvals will have value beyond the original application. The NRC states that it intends to maintain a record of approved alternative acceptance criteria and their bases to support streamlined use by future applicants. Question 41 goes further and expressly asks whether applicants and licensees should be allowed to adopt previously approved alternatives without submitting another application when the basis for approval remains applicable. The final rule should implement that concept rather than leaving reuse dependent primarily on informal efficiencies in subsequent reviews.

A reuse mechanism can preserve the safeguards in § 50.220. A subsequent adopter should document why the previously approved alternative applies to its facility and confirm that the assumptions supporting the prior determination remain valid. The adopter should also evaluate the alternative together with other approved alternatives already used at the facility because § 50.220 properly requires consideration of cumulative effects. If a plant-specific difference changes the technical basis, safety margin, defense-in-depth conclusion, or performance-monitoring assumptions, the difference should receive NRC review.

This structure would also resolve the potential interaction with § 50.59. Adoption of a previously approved alternative should not itself require a new substantive § 50.220 approval where the prior basis remains applicable. Section 50.59 should continue to determine whether implementation of the alternative involves an additional facility, procedure, or analytical-method change that independently requires a license amendment. In other words, the prior § 50.220 determination should resolve the acceptability of the alternative criteria; § 50.59 should address any separate plant-specific change not already resolved by that determination.

The NRC should establish this reuse pathway in the final rule rather than rely only on guidance. The ability to adopt a previously approved alternative without submitting another application determines when prior NRC approval has binding procedural effect and should be clear in the regulatory text. DG-1464 can then explain the documentation necessary to demonstrate applicability and the circumstances in which a plant-specific difference requires additional review.

B. PRA Adequacy Under § 50.59 Should Not Depend Exclusively on Prior Application-Specific NRC Approval

BTI also supports allowing quantitative risk results to inform § 50.59 evaluations. Proposed § 50.59(e) would allow a licensee to demonstrate that a change does not result in a “more than minimal increase” under § 50.59(c)(2)(i) and (ii) using a PRA of appropriate scope and quality, while continuing to require maintenance of defense-in-depth and safety margins. The existing qualitative and deterministic pathways would remain available.

The proposal becomes unnecessarily restrictive, however, if “appropriate scope and quality” requires that the relevant PRA model already have been found acceptable in a prior

NRC-approved application. The rulemaking discussion states that an appropriate model must both encompass the proposed change and have been found acceptable for use in a previous NRC-approved application.

Prior NRC review is strong evidence that a PRA is suitable for a particular regulatory application. It should not be the exclusive means of establishing technical adequacy.

The purpose of the new § 50.59 pathway is to improve the determination of whether a proposed change is sufficiently significant to warrant prior NRC review. Conditioning use of that pathway on an earlier application-specific approval creates circularity: a licensee may possess a technically adequate PRA capable of evaluating the proposed change, but cannot use the new risk-informed § 50.59 pathway because the relevant model has not previously been used in a licensing action. The availability of the flexibility then depends in part on licensing history rather than solely on the adequacy of the risk information for the decision at hand.

A licensee may add hazards, improve treatment of dependencies, incorporate new operating experience, or otherwise improve a model after the NRC reviewed it for an earlier application. Conversely, the fact that a PRA was previously accepted for one application does not necessarily establish that it is adequate for every later decision. The relevant question should be whether the PRA has the scope, technical quality, configuration control, and treatment of uncertainty necessary to support the proposed § 50.59 evaluation.

The final rule or DG-1466 should provide an alternative means of demonstrating that adequacy where a relevant PRA model has not already been approved for the specific application. The demonstration should be commensurate with the significance of the decision and sufficient to give the licensee and NRC confidence that the quantitative result is fit for its intended use. Prior NRC approval can remain a straightforward means of satisfying that showing, but it should not operate as a categorical prerequisite.

This change would preserve the role of defense-in-depth and safety margins and would not convert § 50.59 into a mechanism for probabilistic relicensing. The NRC itself describes the proposed PRA option as supplemental to the deterministic licensing basis, not a replacement for it. The question is narrower: whether a technically adequate risk model can inform a change evaluation even when the model's regulatory history does not include a prior application-specific approval.

The same principle would improve treatment of digital instrumentation and controls. The NRC's Question 4 recognizes that common-cause failures in digital systems can trigger § 50.59 criteria that require prior approval and asks whether the proposed revisions adequately address facilities that have transitioned to digital systems. Quantitative risk information can provide useful insight into the safety significance of digital changes, but only if the rule permits technically adequate models to be used without making previous application-specific approval the sole gateway to the new process.

C. VVUQ Approval Should Produce Durable Flexibility for Analytical-Method Changes

BTI supports proposed § 50.221 and the corresponding revisions to §§ 50.59(c)(2)(viii) and 53.1550. The proposal shifts analytical-method oversight away from automatic review of each individual methodological departure and toward an approved process for establishing model and simulation credibility. Under the proposed framework, licensees and applicants may voluntarily establish a VVUQ program, define the models and simulations within its scope, and demonstrate credibility through activities and assessments commensurate with modeling and simulation risk.

This is a significant improvement over a change control system in which use of an improved analytical method can itself trigger a license amendment because it departs from the method described in the FSAR. The NRC identifies that problem directly: advances in modeling can produce more realistic and accurate analyses, yet existing method-of-evaluation controls can require NRC approval even where the new method improves analytical capability.

The value of § 50.221 will depend on whether approval of the VVUQ framework actually substitutes for repeated review of individual methods within its approved scope.

The proposed rule uses a graded approach. Low-risk M&S may be confirmed through the licensee's internal QA process, medium-risk M&S through an independent external organization, and high-risk M&S require NRC review and approval, generally through a license amendment or topical report. That structure appropriately gives the NRC a greater role where reliance on an erroneous model could have greater regulatory or safety consequences.

For high-risk applications, the NRC should focus its initial review on whether the VVUQ program provides a technically sound credibility framework for the defined intended use, including appropriate treatment of verification, validation, uncertainty, applicability, limitations, and relevant error sources. Once those boundaries are approved, subsequent method changes that satisfy the approved credibility criteria should receive the flexibility contemplated by §§ 50.59 and 50.221 rather than being subjected to a substantially equivalent method-by-method approval process through implementation guidance.

The distinction is particularly important for advanced reactors. The NRC recognizes that such designs may not have the decades of operating data and analytical experience available for the existing LWR fleet and states that requiring the same form of method maturity could impose significant testing and analysis before the operating experience needed to improve the model exists. The proposed VVUQ framework is intended to allow those methods to mature as additional data become available without requiring repeated full NRC review, provided the changes remain within an approved VVUQ process.

That does not mean reducing the credibility standard. It means allowing credibility to be demonstrated using evidence appropriate to the technology and intended application. Experimental data, benchmarks, uncertainty characterization, sensitivity and scaling analyses, operating data where available, and other supporting techniques are expressly contemplated by proposed § 50.221. The NRC should evaluate whether that body of evidence adequately supports the intended use rather than treating the absence of a legacy operating-fleet validation database as a reason to default to the existing method-by-method approval structure.

The NRC should also make approved VVUQ work reusable across applicants and licensees where technically applicable. The regulatory analysis assumes that vendors may develop VVUQ programs for adoption by licensees and that, once a program is approved and adopted, later changes to the method can occur without additional NRC review. It estimates both the upfront effort required to develop VVUQ programs and the savings from avoiding later license amendment requests and topical-report reviews. That economic logic depends on the approval being durable.

If the same vendor methodology, credibility framework, intended application, and relevant conditions are used by multiple licensees, each licensee should be able to reference the approved VVUQ work and demonstrate its applicability rather than recreate the underlying credibility

determination. Plant-specific differences that affect the model's intended use or credibility would remain subject to additional evaluation. As elsewhere in this rulemaking, the NRC should review the material difference rather than repeat an already completed technical determination.

This approach is also consistent with the conforming Part 53 changes. Section 53.1550 already performs the Part 53 analogue to § 50.59 and uses risk-informed change criteria as part of the licensing framework. The final rule should preserve that alignment by making the same approved VVUQ framework capable of supporting analytical-method changes across licensing pathways where the technical basis remains applicable.

Responses to Specific Requests for Comment

Question 4. In light of these considerations, do the proposed revisions to 10 CFR 50.59 adequately address issues that may arise from facilities that have transitioned to DI&C? If not, please provide specific comments on changes that would be necessary to address these issues.

BTI generally supports the proposed § 50.59 revisions for facilities using digital instrumentation and controls. The NRC should clarify, however, that use of quantitative PRA in a § 50.59 evaluation depends on the model's technical adequacy for the proposed decision and should not be categorically limited to hazard models that have already been accepted in a prior NRC-approved application. This would allow risk information to better inform the significance of digital-system changes while preserving the existing § 50.59 criteria, defense-in-depth, and safety-margin requirements.

Question 41. The NRC is seeking views on increasing the flexibility provided by the proposed rule. The rule currently does not include the allowance for a licensee or applicant to adopt a previously approved alternative without submitting an application to the NRC, provided that the basis for the staff's approval is demonstrated and documented to be applicable to the adoptee's plant or facility. Would adding such flexibility increase or accelerate the use of the proposed rule? What are some viable approaches to resolve any conflicts of such flexibility with the requirements in 10 CFR 50.59?

Yes. BTI recommends allowing applicants and licensees to adopt previously approved alternatives without a new § 50.220 or § 52.220 application where they demonstrate that the prior technical basis, conditions, and limitations remain applicable. This would make the new pathway more

useful by allowing regulatory determinations to be reused rather than re-established for each facility.

Section 50.59 should continue to govern any separate plant-specific change needed to implement the alternative. If that change independently meets a § 50.59 criterion requiring prior NRC approval, a license amendment would still be required; § 50.59 should not, however, reopen the acceptability of the alternative itself when the NRC has already approved it and the original basis remains applicable.

Recommendations

The NRC should finalize §§ 50.220, 52.220, and 50.221 and the proposed risk-informed revisions to § 50.59, while making three targeted changes. These changes would preserve NRC review of new or materially different technical issues while allowing the new frameworks to perform their intended function: concentrating prior approval on questions that have not already been resolved and on changes significant enough to warrant additional NRC review.

1. **Previously approved alternative acceptance criteria should be reusable** where a later applicant demonstrates that the original basis and conditions remain applicable.
2. **PRA adequacy under § 50.59 should be based on fitness for the intended decision** rather than making prior application-specific NRC approval the exclusive route to using quantitative risk results.
3. **Approval of a VVUQ program should create durable flexibility** for subsequent analytical-method changes and permit approved vendor or programmatic work to be referenced by other licensees where technically applicable.

VI. DETERMINATE AND DATA-BACKED THRESHOLDS FOR REACTOR SAFETY ASSESSMENTS

BTI supports the NRC's effort to establish more determinate and data-backed thresholds for reactor safety assessments. The proposed framework would define design basis events (DBEs) and beyond design basis events (BDBEs) using quantitative initiating-event frequency thresholds, create distinct event categories with graded analytical treatment, and permit more realistic and best-estimate methods for BDBEs when uncertainties are appropriately addressed. This is a

meaningful improvement over relying primarily on event categories and analytical conventions developed around the historical LWR fleet. That structure advances the objective of Executive Order 14300, which directs the NRC, where feasible, to use determinate and data-backed thresholds so that reactor safety assessments focus on credible and realistic risks.¹⁴ DG-1454 itself recognizes that the traditional NUREG-0800 Chapter 15 DBE spectrum may not realistically represent newer designs, particularly reactors relying on passive safety features or non-LWR technologies.

DG-1454 would generally classify initiating events with a mean frequency of at least 10^{-4} per reactor-year as DBEs and events below that threshold but with a mean frequency of at least 10^{-6} per reactor-year as BDBEs. It also requires explicit treatment of aleatory and epistemic uncertainty and considers whether the 95th-percentile frequency crosses the applicable threshold. For BDBEs, the guidance permits realistic assumptions, best-estimate methods, credit for non-safety-related SSCs where appropriate, and relief from some redundancy and single-failure assumptions.

A. Initiating-Event Frequency Should Not Obscure the Safety Significance of the Resulting Event Sequence

DG-1454's decision to classify events according to the frequency of the initiating event, rather than the frequency or safety significance of the resulting event sequence is incorrect. DG-1454 expressly distinguishes an initiating event from an event sequence, which includes subsequent successes and failures of systems, functions, and operator actions. The guidance declines to use event-sequence frequency for event binning in part because doing so could create ambiguity and require applicants to develop a PRA or other systematic risk evaluation earlier in the licensing process.

That choice makes the framework simpler, but it obscures an important characteristic of advanced reactor designs. An initiating event can occur at a frequency that places it in the DBE category even where inherent characteristics, passive safety features, or highly reliable mitigating functions make progression to a safety-significant plant condition much less likely. In that

¹⁴ Executive Order 14300, "Ordering the Reform of the Nuclear Regulatory Commission," 90 Fed. Reg. 22587 (May 23, 2025), <https://www.federalregister.gov/d/2025-09798>.

circumstance, classification based only on the beginning of the event can assign the same regulatory treatment to scenarios with materially different safety significance.

DBEs remain subject to the conservative analytical framework associated with NUREG-0800 Chapter 15 and are used to establish the safety-related SSCs needed to perform required mitigation functions. BDBEs, by contrast, may use more realistic assumptions and best-estimate methods and may credit non-safety-related SSCs. An initiator-only classification can place a highly mitigated advanced-reactor scenario into the DBE framework without fully reflecting the inherent or passive characteristics that reduce the likelihood that the initiator progresses to the condition of regulatory concern.

BTI does not recommend abandoning initiating-event frequency as the organizing principle or requiring every applicant to develop a detailed PRA simply to categorize events. That would undermine the predictability and relative simplicity the proposal is intended to achieve. But the NRC should ensure that implementation does not cause the initiating-event threshold to become a proxy for safety significance where the design provides a technically supported basis for distinguishing the two.

B. Quantitative Thresholds Should Remain Presumptive and Qualitative Overrides Should Be Bounded

A related concern is that DG-1454 may reintroduce substantial case-by-case judgment around thresholds that are intended to be determinate. The guidance requires applicants to consider both qualitative and quantitative criteria when identifying DBEs and BDBEs, in part to address limited operating experience and significant uncertainty. It also relies on expert judgment where relevant data are sparse and on defense-in-depth, uncertainty treatment, and other qualitative considerations when determining the appropriate regulatory treatment.

Those safeguards are necessary. New technologies may not have the operating history needed to support precise numerical frequency estimates, and uncertainty near a threshold cannot be ignored. The problem arises if qualitative considerations become an open-ended basis for overriding the quantitative classification. The proposed § 50.2 definitions themselves refer to thresholds that provide an “appropriate level of safety” and are acceptable to the NRC. Without sufficiently clear implementation criteria, applicants could satisfy the numerical threshold yet

remain uncertain whether the NRC will place the event in a more conservative category based on additional qualitative judgment.

The final framework should distinguish between judgment used to determine whether the numerical estimate is technically credible and judgment used to override the resulting classification. The former is unavoidable where data are sparse or uncertainty is material. The latter should be bounded and supported by a clear technical basis. Otherwise, the new numerical thresholds could become only the starting point for the same type of case-by-case event categorization that the reform is intended to reduce.

C. The Credibility Threshold Should Not Be Based on Frequency Alone

The credibility threshold produces a mirror-image error to the concern above. Preceding this section, BTI described that initiating-event frequency can overstate safety significance and produce unnecessarily conservative treatment. Proposed § 50.2 would define a beyond design basis event as “an initiating event with a frequency below the threshold for a design basis event, but greater than or equal to a threshold for credibility that provides an appropriate level of safety and is deemed acceptable by the NRC,”¹⁵ and DG-1454 sets that threshold at a mean initiating-event frequency of 1×10^{-6} per reactor-year.¹⁶ Events below it are non-credible and “do not require a safety assessment.”¹⁷ Because the threshold is defined on frequency alone, it is consequence-blind. It removes from assessment precisely the class of event a risk-informed framework exists to identify: the very low frequency event with very high potential consequence. That is not an argument about where the line sits. A perfectly placed frequency threshold, applied with graded rather than binary treatment, would still be blind to consequence, because the defect is the dimension in which the line is drawn.

Credibility is better understood as a property of the pair formed by a risk and the decision it bears on, rather than as a property of the event alone. An event term earns a place in an analysis when it is realizable, meaning a lawful causal pathway to the scenario exists; estimable, meaning

¹⁵ LOS Proposed Rule, proposed 10 C.F.R. § 50.2 (definitions of “Design basis event” and “Beyond design basis event”).

¹⁶ U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1454, Implementation of Determinate and Data-Backed Thresholds for Reactor Safety Assessments (Proposed Regulatory Guide 1.269), ADAMS Accession No. ML25283A021 (July 2026) [hereinafter DG-1454], § 4.2, at 18.

¹⁷ DG-1454, § 4, at 17.

its frequency and consequence can be bounded by an accepted method; marginal, meaning the licensed activity's incremental contribution is non-negligible against the scenario's own consequence; and decision-relevant, meaning some option's ranking changes across the risk's credible range.¹⁸ Two features of that structure answer objections that a frequency threshold is often used to address. The improbable but catastrophic natural hazard is excluded by marginality rather than by frequency: an asteroid strike or a caldera eruption is set aside because the licensed activity's increment is negligible against the consequence of the event itself, not because the event is rare. And a hazard that has never occurred at a site but is boundable from a hazard curve remains in scope, because the standard is estimable rather than observed. That is consistent with DG-1454's own provision that an initiating event that has actually occurred in a facility or system of similar design "should always be considered credible,"¹⁹ which is a sound sufficient condition and should not be read to make the absence of occurrence evidence of non-credibility.

The Executive Order's paired phrase, "credible, realistic," describes two different jobs.²⁰ Credibility governs admission to the analysis. Realism governs how an admitted quantity is estimated. Separating them yields an implementable rule: where an analysis only has to bound a quantity, conservatism is appropriate and additional precision carries little decision value; where quantities are weighed against one another, realistic best-estimate treatment is required, because bias in a weighed quantity silently re-weights the result. That asymmetry is not a matter of analytical care. Where a systematic bias enters a quantity that is weighed, it sets a floor on what the analysis can resolve, and that floor does not fall as the underlying data improve; a structure demonstrated in the low-dose radiation epidemiology context, where the binding constraint on study viability is a bias wall that sample size does not enter at all.²¹ Conservatism applied to a bounding calculation is recoverable, because the direction of the error is known and the bound still bounds; bias applied to a weighted quantity is not. DG-1454 instead grades the estimation standard by event category, retaining conservative Chapter 15 treatment for DBEs and permitting

¹⁸ Adam Stein, "What Is Safe Enough?," presentation at the 18th International Conference on Probabilistic Safety Assessment and Management, panel on Changes in Nuclear Risk Regulation (July 21, 2026).

¹⁹ DG-1454, § 4.1, at 17.

²⁰ DG-1454, at 9 (quoting Exec. Order No. 14300 § 5(h), "Ordering the Reform of the Nuclear Regulatory Commission" (May 23, 2025)).

²¹ Adam Stein, *Sample Size Is Not the Constraint: A Sample-Size-Invariant Bias Wall in Routine Low-Dose Radiation Epidemiology*, (Preprint), <https://doi.org/10.5281/zenodo.22217293>.

realistic, best-estimate methods for BDBEs.²² The standard should follow the function the number performs. A bounding calculation is a bounding calculation whether the event is a DBE or a BDBE, and a quantity that is weighed should be best-estimated in either category. Grading by frequency bin rather than by analytical function is the same error as defining credibility on frequency, expressed in the estimation dimension rather than the admission dimension.

The guide should also state expressly what a credibility determination is. It scopes analytical effort. It does not declare an event harmless, and it relaxes no protective requirement. DG-1454 already implies this as § 8 provides that use of these thresholds “does not obviate the need to demonstrate a facility’s design consists of adequate defense-in-depth,” § 9 says the same of safety margins commensurate with risk significance, and the guide requires that justifications for determining events to be non-credible be documented and retained.²³ But the guide does not say it directly, and the term “non-credible” invites a stronger reading on its face. A sentence making the distinction explicit costs the staff nothing and forecloses a predictable misreading in both directions: by applicants treating the screen as a safety conclusion, and by others reading it as one.

The NRC should preserve the proposed quantitative, graded framework while ensuring that the thresholds remain meaningful in practice. DG-1454 should provide sufficient clarity on how initiating-event frequency, uncertainty, qualitative criteria, and the safety significance of highly mitigated scenarios interact so that applicants can predict the analytical category and regulatory treatment that will result. The objective should be to retain appropriate engineering judgment without allowing that judgment to displace the determinate, data-backed structure that is the principal benefit of the proposal.

Recommendations

1. **Make the quantitative thresholds presumptive and bound qualitative overrides.**

The NRC should preserve the proposed determinate DBE/BDBE thresholds, but DG-1454 should identify the specific circumstances in which qualitative criteria, uncertainty, defense-in-depth, or engineering judgment can change the classification indicated by the numerical threshold. Any departure from the numerical classification should have a

²² DG-1454, at 17 (BDBEs “may use more realistic, best estimate methods for their assessment and are not necessarily subject to redundancy and single failure assumptions”).

²³ DG-1454, §§ 8, 9, at 22.

documented technical basis. This directly addresses the concern that the 10^{-4} and 10^{-6} thresholds could become merely starting points for case-by-case staff judgment. DG-1454 expressly requires both quantitative and qualitative criteria and uses the 95th-percentile frequency as an additional consideration, so implementation clarity matters. Because events just above the credibility threshold remain within the safety-assessment framework while events just below it may be screened from further assessment, small differences in estimated frequency can produce disproportionate differences in regulatory treatment. DG-1454 should provide graded treatment near the boundary where frequency uncertainty could affect the classification, rather than allowing the numerical cutoff alone to determine whether further assessment is required.

2. **Do not let initiating-event frequency alone force unnecessarily conservative treatment of highly mitigated advanced-reactor scenarios.** The NRC should clarify in DG-1454 how inherent and passive safety characteristics, and where appropriate technically supported information about the resulting event sequence, can inform the analytical treatment when initiating-event frequency alone materially overstates the safety significance of the scenario. This should not require every applicant to develop a full PRA merely to categorize events. The NRC deliberately chose initiating-event frequency in part to avoid that burden, but DG-1454 also recognizes that the historical DBE spectrum may not realistically represent passive or non-LWR designs.
3. **Define credibility on the risk-decision pair rather than on frequency alone.** The NRC should revise proposed § 50.2 so that the threshold for credibility is not defined solely by initiating-event frequency, and DG-1454 should provide that an event is not screened from assessment on frequency where its potential consequence is severe. The guide should add a decision-relevance criterion, so that analytical effort tracks what could change a regulatory outcome, and a marginality criterion addressing the licensed activity's incremental contribution, which is the appropriate basis for setting aside improbable high-consequence natural events. It should restate the occurrence criterion in § 4.1 as estimable rather than observed. It should separate the admission standard from the estimation standard, providing that conservatism is appropriate where a quantity is bounded and best-estimate treatment is required where quantities are weighed, graded by analytical function rather than by event category. And it should state expressly that a credibility determination bounds analysis rather than risk and declares nothing safe.

VII. HIGHER-ENRICHMENT AND ACCIDENT-TOLERANT FUEL

BTI supports the NRC's proposed modernization of requirements for higher-enrichment and accident-tolerant fuels. Removing legacy regulatory barriers tied to the traditional 5 weight-percent enrichment limit will allow fuel designs to be evaluated against their actual safety performance rather than an enrichment threshold developed around the existing LWR fuel cycle. BTI also supports the proposed performance-based approach to emergency core cooling system requirements, including the use of fuel-coolability criteria that can account for fuel fragmentation, relocation, and dispersal and accommodate new cladding and fuel technologies. These changes appropriately allow higher-enrichment, higher-burnup, and accident-tolerant fuels to be licensed where applicants demonstrate that the relevant safety functions and acceptance criteria are satisfied, rather than requiring exemptions or forcing new fuels into assumptions developed for conventional zirconium-clad fuel.

However, the final rule should include an explicit discussion of how fuel qualified for reactor operation under the new fuel-dispersal and coolability provisions in proposed § 50.46a(e)(1) and (f) will be classified and handled under the NRC's separate storage and transportation regulations in 10 CFR Parts 71 and 72. Proposed § 50.46a would, for the first time, treat fuel dispersal and partial loss of cladding ductility as compatible with reactor-side coolability, extending to brittle failure of individual rods where coolability can otherwise be demonstrated.²⁴ DG-1263 correspondingly accepts a total loss of post-quench ductility within the burst region of high-burnup rods.²⁵ Both provisions operate on the reactor side of the fuel cycle only; the preamble does not address how fuel that has degraded, dispersed, or lost cladding integrity under this new performance basis will be classified once it leaves the core.

Storage under 10 CFR § 72.122(h)(1) requires that spent fuel cladding be protected against degradation leading to gross rupture, or that the fuel otherwise be confined to preserve retrievability under § 72.122(l), while Part 71 transportation packages rely on the geometric integrity of the fuel to support the criticality and containment demonstrations required under §§ 71.43(d) and 71.55.²⁶ NRC staff has already identified the underlying compatibility problem: a

²⁴ U.S. Nuclear Regulatory Commission, *Modernizing Reactor Licensing, Safety Oversight, and Siting Practices*, Proposed Rule, Docket No. NRC-2025-0975; RIN 3150-AL44, ML26176A438 (July 2026), § XXXVI.F.

²⁵ U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1263, Revision 1, *Establishing Analytical Limits for Zirconium-Based Alloy Cladding* (Proposed New Regulatory Guide 1.224), at 16 (July 2026) (§ C.4.A).

²⁶ 10 C.F.R. § 72.122(h)(1), (l); 10 C.F.R. §§ 71.43(d), 71.55.

2012 staff paper on high-burnup fuel storage and transportation found that Part 71 contains no explicit cladding-integrity or retrievability requirement, that Part 72's gross-rupture and retrievability criteria were not developed with degraded high-burnup cladding in mind, and that licensees unable to demonstrate cladding integrity would need to can or otherwise reconfigure fuel before shipment.²⁷ The NRC's own regulatory analysis for this rulemaking quantifies the industry cost of developing technical solutions to fuel dispersal but does not identify any corresponding cost or risk on the storage and transportation side, even though the same fuel is the subject of both analyses.²⁸ Proposed § 50.46a would extend the range of cladding conditions the agency accepts at the reactor without resolving, or acknowledging, that open compatibility question at the back end. The final rule should direct staff to state, in the preamble or in DG-1263, whether fuel qualified as coolable under proposed § 50.46a(f) is presumptively "damaged fuel" for storage and transportation purposes, and should commit to coordinating any resulting revisions to Part 71 and Part 72 guidance before licensees begin removing fuel qualified under the new criteria from the core.

Recommendations

1. **Preserve the higher-enrichment and Accident-Tolerant Fuel reforms.** Finalize the removal of legacy enrichment barriers and the performance-based § 50.46a pathway where applicants can demonstrate compliance with the applicable safety criteria.
2. **Coordinate reactor fuel approvals with Parts 71 and 72.** The final rule or DG-1263 should explain how fuel qualified under § 50.46a(e)(1) and (f) will be evaluated under Parts 71 and 72, including when changes in cladding condition, fragmentation, or dispersal require additional analysis, confinement, or damaged-fuel treatment. NRC should coordinate any necessary storage and transportation guidance before fuel qualified under the new criteria reaches those stages.

VIII. LICENSE, PERMIT, AND APPROVAL DURATION

BTI supports the NRC's general direction in extending license, permit, and approval durations where the underlying regulatory determination remains valid. BTI supports extending renewed

²⁷ U.S. Nuclear Regulatory Commission, *Compatibility of Requirements for Storage and Transportation of Spent Nuclear Fuel (Retrievability, Cladding Integrity, and Safe Handling)*, ML12220A552, at 1–3 (Aug. 2012).

²⁸ U.S. Nuclear Regulatory Commission, *Regulatory Analysis for the Proposed Rule—Modernizing Reactor Licensing, Safety Oversight, and Siting Practices*, NRC-2025-0975; RIN 3150-AL44, ML26181A099, Table 16 & n.15 (July 2026).

operating licenses to 40 years, removing the existing restriction on early submission of license-renewal applications, allowing voluntary risk-informed and performance-based alternatives for aging management, extending initial and renewed manufacturing licenses to 40 years, and eliminating the fixed expiration and renewal requirement for early site permits (ESPs).

BTI also supports streamlining subsequent license renewal so that NRC review can focus on departures from generic guidance, plant-specific conditions, and aging issues with greater safety significance rather than revisiting aging-management activities whose adequacy has already been established. These reforms can reduce repetitive licensing activity and direct NRC resources toward information that has materially changed or requires plant-specific review. The final rule should preserve these benefits while addressing several technical and implementation issues created by substantially longer periods of operation and approval.

A. The NRC Should Develop a Generic Technical Basis for Operation Beyond 80 Years

The proposed 40-year renewal term creates a pathway for substantially longer operation than the NRC's existing subsequent-license-renewal guidance was developed to address. A plant operating in its initial renewal period could obtain a subsequent renewal extending operation from approximately 60 to 100 years. A plant already authorized to operate through 80 years could seek another 40-year renewal and reach 120 years of total operation. The NRC expressly recognizes both possibilities.

The regulatory pathway is moving ahead of the generic technical guidance for those operating periods. LR-ISG-2026-01 explains that the GALL-SLR Report and SRP-SLR represent the NRC's current state-of-the-art aging-management guidance but were developed based on evaluation of operation from 60 to 80 years. The NRC examined technical issues associated with 100-year operation beginning in 2021, but discontinued that effort because of limited industry interest at the time. More recent NRC work has identified research needs for structural materials beyond 80 years, but the NRC has not yet completed a comprehensive assessment of whether the existing generic guidance requires supplemental activities for those longer periods.

LR-ISG-2026-01 addresses the immediate implementation problem by allowing applicants to continue using GALL-SLR and SRP-SLR while making it incumbent on applicants seeking

operation beyond 80 years to identify any additional aging issues. Those evaluations must consider, among other matters, aging effects with long incubation periods, mechanisms that may progress faster than previously expected, and potential reductions in safety margins. Applicants must then demonstrate either that the existing guidance remains adequate or that supplemental aging-management activities or time-limited aging analyses address the longer operating period.

That approach is reasonable as an interim measure, but it should not become the permanent framework for operation for 100 or 120 years. Aging phenomena common to the existing reactor fleet are better evaluated programmatically than repeatedly reconstructed in plant-specific renewal proceedings. Leaving each applicant to determine independently which portions of the existing 80-year technical basis remain valid can produce duplicative technical work and different review approaches to the same underlying materials and aging questions.

The NRC should resume a programmatic evaluation of aging management beyond 80 years and update GALL-SLR, SRP-SLR, or successor guidance as generic conclusions become supportable. Plant-specific applicants should remain responsible for demonstrating applicability and addressing genuinely plant-specific conditions, but they should not have to independently establish technical conclusions that can reasonably be resolved for the fleet as a whole. This would pair the rule's generic authorization of longer renewal periods with a correspondingly mature generic technical basis for their implementation.

B. Eliminating ESP Expiration Should Also Provide Durable Issue Finality

BTI supports eliminating the fixed expiration and periodic renewal requirement for ESPs. Where site conditions and the information supporting prior NRC findings remain valid, requiring a comprehensive renewal merely because a fixed period has elapsed adds little regulatory value.

The proposed rule, however, would retain a 20-year limitation on finality for certain environmental matters following ESP issuance or approval of an update amendment. The NRC itself recognizes the potential redundancy: applicants referencing an ESP would continue to be required to review for new and significant information, and the NRC specifically asks whether a separate 20-year finality limit remains necessary.

An ESP that no longer expires should not automatically lose the principal regulatory benefit of early issue resolution because 20 years have passed. The relevant question is whether the factual and analytical basis for the earlier determination remains applicable. New and significant information, materially changed site conditions, or another established reopening criterion can provide a substantive reason to revisit a prior determination. Elapsed time alone does not establish that an issue previously resolved by the NRC has become unresolved.

The NRC should align ESP duration and issue finality by allowing resolved matters to remain final unless new information, changed conditions, or an applicable reopening standard provides a basis for renewed review. ESP holders could update particular matters through permit amendments without reopening unrelated determinations that remain supported. That approach would preserve the NRC's ability to address stale information while making the elimination of periodic ESP renewal a meaningful finality reform rather than simply an open-ended permit duration.

Responses to Specific Requests for Comment

Question 20. With the proposed change to remove the requirement for renewal, all changes to update the early site permit (ESP) must be made by amendment. What other options could the staff consider to increase the useability of an ESP? Could the provisions in the, "Commission Policy Statement on Deferred Plants," (52 FR 38077; October 14, 1987) be applied to ESPs? Please provide a basis for your response.

BTI supports allowing ESP holders to update individual matters through targeted amendments without requiring comprehensive reconsideration of the permit. An amendment should reopen only the information or determination the holder elects to update, together with any other matter whose continued validity is materially affected by that change. Matters that remain supported by the original record should retain finality. This would make an ESP more usable over a long period without recreating the periodic renewal process the proposed rule appropriately removes.

The NRC could consider adapting elements of the Deferred Plants Policy Statement to ESPs where they provide an administrative mechanism for maintaining an approval during an extended period of inactivity. Any such approach should complement, rather than replace, the amendment and finality framework. It should not require periodic comprehensive review or reopen matters solely because time has passed. The relevant question should remain whether information or conditions material to the NRC's prior determination have changed.

Question 21. For a COL application that references an ESP, the changes proposed in 10 CFR 52.39 and 53.1188 would limit finality on safety and environmental issues to 20 years following ESP issuance or approval of an update amendment. Because the proposed rule would also retain the requirement for applicants to review for new and significant information when referencing an ESP, the NRC requests input on whether limiting finality to 20 years is necessary to ensure that all relevant environmental information is considered in licensing decisions.

BTI does not believe a fixed 20-year limitation on ESP finality is necessary. The referencing applicant would remain responsible for identifying new and significant environmental information, and the NRC retains authority to address materially changed site conditions or information that undermines the basis for an earlier determination. Those mechanisms address whether the information supporting the ESP remains valid; elapsed time alone does not.

If the information and performance-relevant conditions underlying the NRC's earlier determination have not materially changed, the passage of 20 years does not itself provide a substantive basis for reopening the issue. In that circumstance, the time limit becomes an arbitrary proxy for changed conditions rather than a risk- or information-based trigger for additional review. Finality should instead be preserved unless new and significant information, materially changed site conditions, or another applicable reopening standard provides a substantive basis for renewed review.

Recommendations

1. **The NRC should develop a generic technical basis and updated guidance for reactor operation beyond 80 years.** As longer renewal terms make operation to 100 years and beyond increasingly foreseeable, the NRC should resolve common aging-management and research issues generically rather than requiring each applicant to recreate the technical basis independently. This must not avoid the technical challenges of very long-term operation of plants and incomplete license review during renewal.
2. **The NRC should pair elimination of ESP expiration with durable, information-based issue finality.** Previously resolved ESP issues should remain final unless new and significant information, materially changed site conditions, or another applicable reopening standard provides a substantive basis for renewed review; elapsed time alone should not trigger reconsideration.

IX. EMERGENCY PREPAREDNESS SHOULD REMAIN RISK-INFORMED AND PERFORMANCE-BASED

BTI supports the NRC's effort to make emergency preparedness more performance-based and to remove prescriptive requirements that do not materially improve emergency response. The final rule should preserve that flexibility while ensuring that reductions in emergency-planning requirements remain based on a comprehensive assessment of risk and response needs, rather than treating event frequency, reactor power, or emergency planning zone size as substitutes for the capabilities necessary to respond to an accident. The final rule should remove the technology-class limitation on § 50.160 and condition use of the framework on a demonstration of the facility's consequences and response needs rather than on election alone. It should revise EPZ requirements, eliminate several prescriptive planning provisions, and further risk-inform emergency-plan change control.

A. Emergency Preparedness Should Be Risk-Informed, Not Risk-Based

Emergency preparedness (EP) implements a distinct public-health-and-safety function. The AEA authorizes the NRC to establish standards necessary or desirable to protect health and minimize danger to life or property, and its licensing provisions use reasonable assurance of adequate protection of public health and safety as a governing safety standard.²⁹ The NRC has translated that responsibility into a specific emergency-planning finding in § 50.47(a)(1): before issuing an initial operating or combined license, the NRC must find "reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency."

That distinction should govern how risk information is used in emergency preparedness. PRA and other risk information can appropriately inform the scale and extent of advance planning. Accident frequency, potential consequences, release timing, available mitigation, and the time available for protective action can all affect what planning is warranted. Facilities presenting materially different hazards need not maintain identical emergency-planning arrangements.

But EP asks a conditional question that is different from the question posed in plant design: if an event occurs and conditions deteriorate, what capabilities are needed to respond? A low

²⁹ Atomic Energy Act of 1954 as amended 53(e)(4).

estimated event frequency may justify limited additional design or mitigation requirements, but it does not by itself establish that the event has no relevance to emergency preparedness. An event that contributes little to aggregate risk because it is very unlikely can still require the ability to assess plant conditions and releases, communicate with responsible authorities, coordinate a response, and support appropriate protective actions if it occurs. Screening such an event from design requirements and screening it from emergency preparedness are therefore not the same regulatory judgment.

That structure identifies the core function of emergency preparedness. EP is not primarily an additional mechanism for determining whether an accident should be prevented through plant design. It ensures that the licensee and relevant response organizations possess the capabilities needed if an emergency occurs. It requires capabilities to classify and assess an event, implement and recommend protective actions, communicate with response organizations, maintain command and control and adequate staffing, assess radiological conditions and releases, coordinate emergency response, and demonstrate those functions through drills and exercises.³⁰ An EPZ is one planning tool within that framework; it is not the underlying safety finding. Section 50.47(a)(1) asks whether adequate protective measures can and will be taken, not whether a particular geographic EPZ has been established.

EP should be risk-informed rather than risk-based. Risk information can appropriately inform the scale of preparedness, but it should not become a screening mechanism that removes accident scenarios from emergency planning solely because their estimated frequency is very low. Historical experience illustrates the problem with judging emergency-response needs solely from ex ante risk estimates or eventual outcomes. During an unfolding accident, decisionmakers do not know the ultimate release or consequence. The Fukushima accident is an example. It involved conditions beyond the plant's design basis, yet those conditions occurred and required sustained emergency response. The lesson for EP is not that every very-low-frequency event must drive additional design requirements; it is that once an accident occurs, its prior probability no longer determines what responders must do. EP is about what capabilities are needed to assess conditions, communicate with public authorities, recommend protective actions, and support an informed response as a beyond design basis event develops. EP should preserve the capabilities necessary to respond to such events without treating their low probability as a reason to assume that no response will ever be required.

³⁰ 10 C.F.R. § 50.160(b)(1)(iii)–(iv), (c).

Emergency preparedness is evaluated conditionally; an emergency plan is not applied until an event occurs, and at that moment the prior probability of the event is one. The Commission settled that relationship in the final Part 53 rule by defining emergency preparedness and “emergency response measures” as defense in depth. The Commission also placed emergency preparedness outside the safety determination rather than inside it: “emergency plans and emergency planning zones are not safety features in the design”; § 53.855 “contributes an additional independent layer of defense in depth for commercial nuclear plants”; and “[t]herefore, the emergency plans and emergency planning zones under § 53.855 are not used to demonstrate compliance with subpart B and subpart C of part 53. Rather, compliance with the requirements in § 53.855 provides reasonable assurance that adequate protective measures can and will be taken to protect public health and safety in the event of a radiological emergency.”

Declining to screen a low-frequency event out of the emergency-planning basis does not import frequency into the analysis. A risk calculation weights consequence by frequency to produce an expected value; retaining an event for emergency-planning purposes does the opposite, holding it at a probability of one so that the capabilities needed to achieve adequate protection can be identified. Additionally, retaining an event imposes no dose limit on it. A design-basis event carries an acceptance criterion that must be met or the design must change; emergency preparedness carries no such criterion. Adequate protection under § 50.47(a)(1) is a finding about capability, and emergency preparedness does not promise a low or zero dose for every retained sequence. It requires that the licensee and the responsible offsite authorities can assess conditions, project offsite consequences, notify and communicate, recommend protective actions, and expand the response as conditions warrant.

Importantly, the conditional premise governs how the plan is evaluated; it does not make the planning basis unbounded. The scope of what must be analyzed is set by what bears on the finding § 50.47(a)(1) requires: that adequate protective measures can and will be taken. That standard operates in both directions. It streamlines the analysis where a sequence or an uncertainty cannot change the finding, and it expands the analysis where one can, including where the sequence lies below the frequency at which the design basis is defined.

B. EPZ Requirements Should Be Technology-Inclusive as a Planning Tool

BTI supports the scalable and case-specific EPZ, including allowing an applicant to demonstrate that no fixed EPZ is necessary. The NRC should carry that approach through in a technology-inclusive manner by treating the EPZ as a planning tool, rather than as the emergency-preparedness capability itself.

The proposed rule moves in that direction. The NRC explains that an EPZ bounds the area in which detailed advance planning is needed to implement predetermined, prompt protective actions, and that EPZ size should be commensurate with the planning needs of the facility. The NRC also emphasizes that EPZ size does not determine the full scope of emergency planning: emergency-plan capabilities can support responses beyond the EPZ when actual conditions require it. The proposed amendments to § 50.47(b)(10) similarly remove the EPZ as the outer boundary of protective-action planning, while proposed §§ 50.33(g)(1), 50.47(c)(2), and 53.1109(g)(1) would require appropriate actions within and beyond the EPZ or site boundary.

The same distinction should govern application of the consequence criteria used to establish EPZ size. Existing § 50.33(g)(2)(i), which the proposed rule would revise but largely preserve, uses a 1-rem-over-96-hours consequence criterion together with the need for predetermined, prompt protective measures to determine EPZ size. That criterion determines the geographic scope of advance emergency planning; it is not a requirement that every accident sequence remain below 1 rem. An analyzed event that exceeds the criterion may justify a larger planning area or different protective-action capabilities, but the EP provisions do not by themselves require an applicant to redesign the reactor to reduce the consequence below 1 rem. Treating the criterion as a design acceptance limit would convert a performance-based planning provision into a de facto design requirement.

The proposed change is increasingly important for advanced deployment models. A predefined geographic EPZ can be valuable for a stationary reactor because it permits advance coordination of evacuation routes, jurisdictions, communications, and predetermined protective actions. The same planning construct may provide much less value for a mobile or transportable microreactor whose surrounding population, jurisdiction, transportation network, and emergency-response organizations can change with deployment. Requiring a geographic EPZ to move with the reactor would not necessarily improve preparedness. It could instead turn a planning tool into an administrative requirement that must continually be redefined as the reactor changes location.

For those deployments, the more important regulatory question is whether the licensee retains the capabilities needed to respond wherever the reactor is located: to identify and characterize a release, determine where radiological consequences may occur, identify the responsible public authorities, communicate timely information, recommend appropriate protective actions, and support expansion of the response as actual conditions warrant. The absence of a fixed EPZ should not mean the absence of emergency planning. It should mean that the applicant has demonstrated that predesignating a geographic response zone is unnecessary because the required protective functions can be performed effectively without one.

Transportation of nuclear fuel provides a useful functional comparison. Fresh and spent fuel moves through changing locations and jurisdictions, and the transportation framework does not depend on an EPZ that continuously moves with the shipment. Transportation safety instead relies on requirements applicable to the package, shipment, route, coordination, and response to an accident where it occurs. BTI has similarly emphasized for transportable microreactors that transportation requirements should address route, accident, security, and emergency-response considerations as part of the deployment lifecycle.³¹ The transportation framework is not a substitute for reactor EP requirements, but it demonstrates the underlying point: effective emergency response does not always require a permanently predefined geographic zone.

Microreactors make that principle concrete. The NRC cites technical studies indicating that microreactors and other advanced reactor designs may not require extensive plume-exposure EPZs, and BTI has previously supported right-sized emergency planning for low-consequence reactors, including frameworks in which an emergency plan remains even though a conventional EPZ is not required.³² A technology-inclusive rule should allow the same logic to apply whenever the applicant can demonstrate that predetermined geographic planning does not materially improve the protective response.

The NRC should finalize the proposed scalable, case-by-case EPZ framework and preserve the ability to determine that no fixed EPZ is necessary. The final rule and DG-1430 should clarify that an applicant seeking a site-boundary EPZ or no fixed EPZ must demonstrate the capabilities

³¹ Breakthrough Institute, *Comment on Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles*, Docket NRC-2025-0379 (June 15, 2026).

³² U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1430, Performance-Based Emergency Preparedness (Proposed Revision 1 to Regulatory Guide 1.242), ADAMS Accession No. ML25289A302 (July 2026) [hereinafter DG-1430], app. A § A-3.6, at A-5.

necessary to assess radiological conditions and projected releases, identify affected areas and responsible authorities, communicate information needed for protective-action decisions, and support protective measures as conditions warrant. The regulatory objective should be the capability to protect people during an emergency, not maintenance of a predefined geographic zone where that zone does not improve the response. DG-1430 already supplies the consequence-based route to that outcome. It provides that “[i]f the analysis demonstrates that acute doses or early health effects are unlikely to occur beyond the site boundary and there is a low likelihood of exceeding doses associated with stochastic risks beyond approximately 1600 m (1 mile), the EPZ boundary may be set to the site boundary.”³³ The final rule should rely on that consequence-based criterion rather than on a parallel classification in the regulatory text.

One conforming correction is needed in the guidance that implements these criteria. DG-1430 Appendix A § A-3.8 provides that “if all events considered within the spectrum of accidents result in consequences of less than 1 rem TEDE at the site boundary, then no EPZ is needed.”³⁴ That trigger carries no exposure period. The screening value stated earlier in the same section is expressly bounded—a consequence “less than 25 rem TEDE at the site boundary for the duration of the event”—and the regulatory criterion the guide implements, proposed § 50.33(g)(2)(i)(A), is 1 rem TEDE over 96 hours. The guide therefore omits the temporal qualifier on the one trigger that eliminates the emergency planning zone altogether, leaving the most consequential determination in the appendix less specified than the screening step that precedes it. BTI recommends that § A-3.8 be conformed to the rule by stating the no-EPZ trigger as 1 rem TEDE over 96 hours at the site boundary.

C. The 300 MWt Benchmark Should Remain in the Statement of Considerations instead of the Rule

BTI does not support the 300 MWt benchmark as proposed in the regulatory text. The proposed rule would provide regulatory certainty for a site-boundary EPZ for facilities with an authorized power level below 300 MWt, while retaining case-by-case EPZ determinations for all facilities. BTI supports the NRC’s effort to establish a generic technical basis that can streamline EPZ determinations for lower-power reactors, but the 300 MWt benchmark should remain in the

³³ DG-1430, app. A § A-3.8, at A-8.

³⁴ DG-1430, app. A § A-3.6, at A-5.

rulemaking's Statement of Considerations rather than be codified in the regulatory text. The NRC explains that its proposed 300 MWt provision is based on staff review of technical analyses indicating that reactors below that power level present a very low risk of acute offsite doses requiring predetermined, prompt protective measures.³⁵ That analysis is valuable and should remain part of the agency's technical basis for the rule.

The limitation of thermal power as a generic cutoff is clearest across reactor technologies. Acute offsite dose depends not only on the radionuclide inventory associated with reactor power, but also on whether, when, and in what quantity that inventory can escape the barriers between the fuel and the environment. The technical literature underlying modern EP approaches recognizes that consequence estimates depend materially on source-term characteristics and containment response, including the potential for containment bypass or failure. Those relationships are design- and technology-specific. For example, fuel forms such as TRISO particles, and designs that rely on the performance of a functional containment, produce release behavior different from that assumed for conventional light-water fuel and containment systems, and many designs specifically mitigate the pathways that produce a rapid bypass release. A single thermal-power cutoff therefore will not correspond to the same offsite consequence potential across technologies. The NRC's current regulation already recognizes this. Existing § 50.33(g)(1) permits the size of the EPZ to be determined case by case "for gas-cooled reactors and for reactors with an authorized power level less than 250 MW thermal," pairing a technology criterion with a power criterion; the proposed rule would delete that sentence and leave a power line standing alone. A 300 MWt benchmark may be a useful screening value where the technical basis is applicable, but codifying it as a generally applicable regulatory dividing line risks unintended implementation consequences.

The preamble states that the 300 MWt provision rests on analyses "based, in part, on conservative source terms for designs with an authorized power level less than 300 MWt, which suggest that there is a very low risk of exceeding acute doses offsite requiring a predetermined, prompt response."³⁶ That conclusion holds for one of the two accident sequences in the cited analysis and not for the other. NSIR-2017-002 evaluated a contained long-term station blackout and a containment-bypass interfacing-systems loss-of-coolant accident at core inventory scaling

³⁵ Nuclear Regulatory Commission, *Executive Order 14300: Modernizing Reactor Licensing, Safety Oversight, and Siting Practices: Part 3 of 3*, Public Meeting, August 6, 2026, ML26208A059.

³⁶ LOS Proposed Rule, preamble discussion, "Emergency Planning Zone Certainty," at 124–125.

factors of 100, 10, and 1 percent of a 2,546 MWt reference core. The 10 percent case corresponds to roughly 255 MWt and is the only case in the study near the proposed threshold. At that scaling factor, the 200 rad acute red-marrow criterion produced a nonzero exceedance distance within 16 km in 30 of 200 parameter samples for the station blackout sequence and in all 200 samples for the bypass sequence; the 50 rad criterion was met in all 200 bypass samples as well.³⁷

SAND2025-08913 reproduces the same split at the same reactor size factor: no calculated acute exceedance at any quantile for the station blackout source term, against 95th-percentile exceedance distances of 1.3 km at 50 rem and 0.27 km at 200 rem for the bypass source term.³⁸

Holding radiological inventory constant, whether the release bypasses containment changes the acute-dose result entirely. That is consistent with how NSIR-2017-002 treats the acute criteria, which it applies precisely because “these high dose levels are expected to warrant prompt protective actions.”³⁹

The same table bears on how far the benchmark can be carried. At the 10 percent scaling factor, the bypass sequence yields a recovery fraction of zero for both the 5 rem and the 1 rem effective-dose criteria. That value is ambiguous in the report’s own terms—exceedance distances either greater than 16 km or not projected to exceed the criterion at any nonzero distance—but here it can only be the former, because the more demanding acute criterion was exceeded in all 200 samples at the same scaling factor, and because SAND2025-08913 independently reports a mean 1 rem exceedance distance of 19 km for the bypass source term at the one percent factor.⁴⁰ On the analyses the NRC cites, a bypass release at approximately the proposed threshold projects 1 rem over 96 hours beyond 16 km, which is the outer bound of the EPZ range this rulemaking proposes. The record therefore does not support reading the 300 MWt provision as generic support for a site-boundary EPZ irrespective of release pathway, and it does not support treating

³⁷ Analyses Informing Emergency Planning Zone Size Determinations: Identification of Parameter Sensitivities, Task 1.5-1.6 Report for User Need NSIR-2017-002, NRC, ADAMS Accession No. ML19338E460, Table B.1, at B-5 (Dec. 2019) [hereinafter NSIR-2017-002 Task 1].

³⁸ Dose Exceedance Distance Sensitivity Based on Parametric Uncertainty, SAND2025-08913, Sandia National Laboratories, Tables 3-2 and 3-4, at 22, 25 (July 2025) [hereinafter SAND2025-08913].

³⁹ NSIR-2017-002 Task 1, SAND2025-08913, § 3.5, at 40; see also id. § 2.2, at 5 (selecting 0.5 and 2.0 Gy-equivalent to the red bone marrow “to inform the potential for early severe health effects”); id. § 3.1, at 26 (identifying 200 rad as the reference dose used in NUREG-0396 for evaluating atmospheric releases).

⁴⁰ NSIR-2017-002 Task 1, SAND2025-08913, Table B.1, at B-5 (recovery fraction of 0 for the ISLOCA sequence at the 10 percent scaling factor for both the 0.05 Sv and 0.01 Sv effective-dose criteria); id. at B-2 (defining the recovery fraction and describing a value of 0 as exceedance distances “either greater than 16 km or that were not projected to exceed the specified dose at any non-zero distance”); SAND2025-08913, supra note 45, Table 3-1, at 21 (mean 1 rem exceedance distance of 19 km for the ISLOCA source term at the 1 percent reactor size factor).

that provision as a basis for determining that no EPZ is necessary. Proposed § 50.33(g)(1) makes the need for an EPZ a case-by-case determination under § 50.33(g)(2), and the final rule and its Statement of Considerations should state that the 300 MWt benchmark does not substitute for that determination.

Whether such a pathway exists is a function of the design, not of its rated power. The analysis underlying the 300 MWt benchmark scales the inventory of a large light-water reactor while holding its relative release fractions constant,⁴¹ so it does not represent designs that rely on high-retention fuel forms or on the performance of a functional containment. The analysis is therefore neither conservative nor non-conservative for those designs; it is silent about them. A power-level criterion cannot see that difference. It would extend automatic relief to a light-water design below the threshold that does have a credible fast-bypass pathway, while withholding it from a functional-containment design above the threshold that does not—the designs for which the acute-dose conclusion is strongest. A cutoff metric that does not transfer across technologies is a poor fit for a rule the NRC intends to be technology-inclusive, and that is a further reason to state the 300 MWt finding as a technical observation in the Statement of Considerations rather than as an operative classification in the regulatory text.

The proposed threshold is neither risk-informed nor performance-based. Thermal power is, at most, a proxy for the consequences and emergency-response needs that should govern an EPZ determination. Codifying the proxy creates a regulatory target: developers have an incentive to design around the threshold, and applicants immediately above it face a different regulatory posture even when the difference in thermal power may have little relationship to the capabilities needed for emergency response. The fact that proposed § 50.33(g)(1) states that a site-boundary EPZ *may* be used below 300 MWt rather than establishing an absolute requirement does not eliminate this concern. Numerical criteria can become de facto dividing lines even when the regulatory text preserves discretion.

Applicants can and have treated a numerical criterion intended to determine the scope of emergency planning as a design target even when the regulation does not require that result. The existing 1,000 MWt definition of a small modular reactor illustrates this problem.⁴² The BWRX-300, rated at approximately 870 MWt, falls within the existing power-based SMR definition, while

⁴¹ NSIR-2017-002 Task 1, SAND2025-08913, § 2.1, at 4 (assuming that “[t]he reduced core inventory has no effect on the accident progression (e.g., release timing and relative isotopic composition of released material)”).

⁴² 10 CFR §§ 50.2, 50.160.

Holtec’s similarly named SMR-300 exceeds the 1,000 MWt limit and has pursued an exemption so that it can use the performance-based emergency-preparedness framework in § 50.160.⁴³ The comparison does not establish that the 1,000 MWt criterion was technically unsound when adopted. It shows how a fixed numerical proxy can produce materially different regulatory treatment for designs that raise the same underlying policy question. The NRC should avoid creating a similar boundary at 300 MWt. That history also informs how the 300 MWt figure should or could be justified. The 1,000 MWt figure in § 50.2 is a scoping definition—a small modular reactor is one licensed “to produce heat energy up to 1,000 megawatts thermal per module”—adopted to establish which applicants could use the performance-based framework in § 50.160.⁴⁴ There was no finding that facilities below 1,000 MWt present consequences suitable for a site-boundary EPZ, and because the criterion is stated per module, the radiological inventory at a site is not bounded. It was also not the only route into § 50.160, which is available to non-light-water reactors and to non-power production or utilization facilities without regard to that figure. Nor did eligibility for § 50.160 itself establish an EPZ; EPZ size continued to be determined under § 50.33(g)(2). A ratio between 300 MWt and 1,000 MWt therefore compares two power proxies rather than establishing margin against a consequence criterion. The provision the proposed rule actually replaces is the sentence in existing § 50.33(g)(1) permitting the size of the EPZ to “be determined on a case-by-case basis for gas-cooled reactors and for reactors with an authorized power level less than 250 MW thermal.”⁴⁵

Moving 300 MWt to the Statement of Considerations would preserve the value of the NRC’s technical work without converting its conclusion into an operative regulatory classification. The NRC should explain that its existing analyses support 300 MWt as a useful generic screening benchmark for identifying facilities for which a site-boundary EPZ is likely to be sufficient without additional applicant-specific analysis, where the assumptions underlying that technical basis remain applicable. Applicants should be able to rely on that generic determination rather than reproduce the NRC’s analysis. The benchmark should not operate as a categorical boundary: facilities above 300 MWt should remain able to demonstrate that their source term, radionuclide-retention characteristics, release pathways, and resulting radiological consequences support a site-boundary EPZ, while additional analysis could remain appropriate below 300 MWt

⁴³ Nuclear Regulatory Commission, *Request for Exemption to Permit the SMR-300 to Utilize 10 CFR 50.160*, public meeting, March 19, 2025.

⁴⁴ 10 CFR § 50.2 (definition of “Small modular reactor”).

⁴⁵ 10 CFR § 50.33(g)(1); LOS Proposed Rule, proposed 10 C.F.R. § 50.33(g)(1).

where materially different design characteristics undermine the generic basis. The regulatory determination would remain tied to radiological consequences and emergency-planning needs, with 300 MWt serving as a screening tool for streamlined treatment rather than a deterministic licensing threshold.

Responses to Specific Requests for Comment

Question 17. Should design features, other than power level, or site attributes be considered for determining when additional analysis by the applicant is necessary to justify site-boundary EPZ for facilities less than 300 MWt? Please provide the basis for your response.

Design features should be considered where they materially affect the source term, release pathway, timing, or magnitude of potential offsite consequences. Those characteristics can be more directly relevant to emergency-planning needs than thermal power itself. In particular, fuel retention performance, containment or confinement performance, and the potential for bypass or other direct release pathways can differ substantially among reactor technologies, such that the same thermal power does not imply the same acute offsite dose potential. Site attributes should likewise be considered where they materially affect the resulting consequences or the protective actions that may be necessary.

The NRC should not convert these considerations into a new list of categorical design-based thresholds. Instead, they should inform whether the generic technical basis supporting streamlined treatment remains applicable to the facility. Where a design's characteristics provide equal or greater assurance that acute offsite doses requiring predetermined, prompt protective measures are unlikely, additional analysis should not be required merely because the reactor exceeds a generic thermal-power benchmark. Conversely, where a facility below 300 MWt has characteristics that materially undermine the assumptions supporting the generic basis, the NRC should retain authority to require additional analysis.

If the NRC nonetheless elects to retain a codified threshold, BTI recommends that it be conditioned on both power level and the absence of a credible pathway for a rapid release that bypasses the containment or functional containment, rather than on power level alone. Conditioning the threshold on the characteristic that actually controls acute dose would keep the codified provision aligned with the technical basis the NRC has developed, and would avoid

extending automatic relief to a design whose release pathway defeats the assumption on which the benchmark rests.

D. A Site-Boundary EPZ Should Not Automatically Eliminate Offsite Reasonable Assurance

The NRC should not allow a determination about EPZ size to become a determination about which accidents remain relevant to emergency planning. DG-1430 draws this distinction clearly. Its generalized EPZ methodology is risk-informed rather than risk-based: frequency may inform the size of the EPZ, but the determination does not rely on a frequency cutoff. DG-1430 further explains that lower-frequency events may be excluded from the quantitative EPZ-sizing calculation when they would not materially affect the result, while those events are still retained for planning.

That distinction is important as methodologies originally developed for non-LWR and LMP applications are used more broadly. An event may be sufficiently unlikely that it does not justify expanding a predetermined geographic planning zone, yet still have consequences that require an emergency response if it occurs. Screening such an event from design-basis requirements or the quantitative EPZ calculation does not answer the emergency-planning question. Emergency preparedness addresses the conditional case: if the event occurs, who may need to act and what response capabilities must be available? A risk-informed framework can use frequency to scale advance planning without treating very-low-frequency, high-consequence events as irrelevant to emergency response. DG-1430's approach—screening events from EPZ sizing where justified while retaining them for planning—provides the appropriate model.

That premise is under pressure from the opposite direction. Industry guidance would screen event sequence families out of the EPZ determination below a frequency of “5E-7 per plant year or greater, when assessed at the 95th percentile,” expressly “[i]n keeping with the implementation of the LMP approach.”⁴⁶ The frequency-consequence target is a risk-based construct built to allocate allowable consequence against frequency for design and licensing-basis-event selection. Whatever its merits there, it is the wrong instrument for a conditional planning determination, because it uses frequency-based risk instead of conditional risk, and because Part 53 (primarily

⁴⁶ Nuclear Energy Institute, NEI 24-05, Rev. 0, An Approach for Risk-Informed Performance-Based Emergency Planning, ML24184C122, § 4.2.1, at 19 (June 30, 2024).

derived from LMP) has already established that emergency planning is not used to demonstrate compliance with the safety criteria it serves. The NRC's draft guidance is correct on this point, and BTI supports it. DG-1430 provides that "[f]requency information may be used to temper the size of the EPZ, but the EPZ determination is not risk-based and does not rely on a frequency cutoff," directs applicants following RG 1.233 not to "consider demarcation of lowest-event frequencies on the F-C target and category definitions a hard-and-fast cutoff to apply to EPZ determinations," and provides that the events considered "should also include those events that have been identified as non-credible or have been retained in the PRA analysis."⁴⁷ BTI recommends that the NRC state this principle in the final rule preamble rather than leaving it to draft guidance that can be revised without notice and comment. A frequency floor adopted for design purposes should not migrate into the emergency-planning basis by reference.

One provision of the draft guidance does not hold that line. DG-1430 Appendix A § A-3.7 directs that "dose assessment results are to be characterized by a distribution of exceedance distances reflecting the combined effect of accident frequency and variability in meteorological conditions," and permits dose-distance results to be "aggregated using frequency information to evaluate the likelihood of exceeding various dose quantities as a function of distance."⁴⁸ BTI recommends that the NRC reconsider this aggregation step. Accident frequency and meteorological variability are not the same kind of quantity and should not be combined into a single distribution. Weather variability is aleatory variation in the consequences of one event. A set of accident frequencies describes a set of distinct, discrete events with different source terms, release timings, release pathways, and warning times, each of which places different demands on responders. Aggregating across them yields an average of conditions that never occur together, and the averaging works in one direction: the sequences that determine what capabilities are needed—a containment bypass, an early release, a release with little warning time—are exactly the ones a frequency-weighted aggregate suppresses, because they are rare relative to the contained sequences that dominate the distribution. The planning value of a consequence analysis lies in the discrete outliers, not in the central tendency. BTI recommends that dose-distance results be presented per source term, conditional on the occurrence of that source term, as § A-3.7's own first example already contemplates,⁴⁹ and that any frequency-weighted

⁴⁷ DG-1430, app. A § A-3.1, at A-3.

⁴⁸ DG-1430, app. A § A-3.7, at A-5 to A-6.

⁴⁹ DG-1430, app. A § A-3.7, at A-5 (providing that analyses "may simply present dose distance curves conditional upon the occurrence of the source term without consideration of frequency").

aggregation should be presented as supplementary information rather than as the basis for the EPZ determination. A clustering approach to aggregation, where events are aggregated based on consequence-weighting, may be a better approach to determine what emergency response capabilities are required.

A performance-based emergency-planning basis also needs a principled stopping point, or the obligation to consider additional sequences becomes unbounded. The criterion is the decision-relevance screen described in Section VI above, applied here to the emergency-planning determination. An uncertainty is decision-irrelevant when its entire plausible range lies within a single decision region; when no admissible resolution of it, however favorable or unfavorable, would cross any boundary that the regulatory determination actually turns on. Effort spent characterizing decision-irrelevant uncertainty yields no safety benefit, and an acceptance criterion that tests it imposes burden without improving the determination. The symmetrical failure is over-truncation, in which uncertainty that is decision-relevant is excluded, yielding a determination that appears better supported than it is. For emergency preparedness the operative boundaries are not dose bands but the discontinuities in the required response: whether predetermined, prompt protective measures are necessary under proposed § 50.33(g)(2)(i)(B); the distance at which the § 50.33(g)(2)(i)(A) criterion is met, which bounds the area of predetermined planning;⁵⁰ and the points at which a further capability under § 50.160(b)(1)(iii) or (iv) becomes necessary, such as offsite dose projection, notification of a responsible offsite authority, or a protective-action recommendation within a required time.⁵¹ A sequence may be set aside when its entire plausible range of consequence, release pathway, and timing lies within one such region, that is, when every capability it would demand is already demanded by a sequence that has been retained. That test is bounded, because the set of required capabilities is finite while the set of conceivable sequences is not, and it terminates without reintroducing a frequency threshold. It also resolves the case that motivates frequency screening: a sequence at 1×10^{-10} per year is set aside not because it is improbable, but because no admissible resolution of it changes what the licensee and the responsible offsite authorities must be able to do.

Measured against that test, a frequency screen at 5×10^{-7} per plant year is over-truncation. It removes sequences without asking whether including them would cross any capability boundary, and therefore yields an EPZ determination that appears better supported than it is. DG-1430 already

⁵⁰ LOS Proposed Rule, proposed 10 C.F.R. § 50.33(g)(2)(i)(A) and (B).

⁵¹ 10 CFR § 50.160(b)(1)(iii), (iv).

applies the correct form of test, but to the wrong variable. It provides that “if the mean frequency of a scenario is below a screening threshold, but the upper end of the frequency uncertainty range lies above that threshold, then the scenario should be considered for inclusion in the analysis”—a decision-region test applied to a frequency boundary.⁵² BTI recommends that the final rule and the guidance apply that same reasoning to the boundary the emergency-planning determination actually turns on: whether any admissible resolution of the uncertainty would change the required response capability or the area within which predetermined, prompt protective measures must be planned. That formulation bounds the analysis, gives applicants a defensible basis for stopping, and does so without importing a frequency cutoff into a determination that assumes the event has occurred.

Section 50.47(f) risks collapsing the same distinction at the institutional level. Section 50.47(a)(1) requires the NRC to find reasonable assurance that adequate protective measures can and will be taken in a radiological emergency. That requirement does not depend on EPZ size. Section 50.47(a)(2) supports the offsite portion of that finding by directing the NRC to consider FEMA determinations regarding the adequacy and implementability of State and local emergency plans. Yet § 50.47(f) makes §§ 50.47(a)(2), (b), and (c)(2) inapplicable to offsite radiological emergency response plans whenever the plume-exposure EPZ does not extend beyond the site boundary.

That automatic linkage becomes more consequential as the NRC makes site-boundary and no-EPZ outcomes available to a broader range of reactors. A finding that predetermined protective-action planning is unnecessary beyond the site boundary does not establish that offsite authorities will never need to act. An actual event may still require notification and coordination with State, local, or Tribal authorities, assessment of offsite consequences, or protective actions outside the area for which advance geographic planning was warranted. The absence of an offsite EPZ should not itself eliminate the mechanism the NRC uses to determine that necessary offsite protective measures can and will be taken.

Scaling an EPZ is legitimate; eliminating offsite-facing emergency-response capability is not. An EPZ may properly be reduced to the site boundary, or dispensed with, where significant offsite preplanning and coordinated, predetermined protective actions are not among the capabilities necessary to achieve adequate protection. It does not follow that offsite-facing capability

⁵² DG-1430, *supra* note 39, app. A § A-3.1, at A-3.

becomes unnecessary. BTI agrees with the NRC's guidance, which is explicit that applicants "should not apply the results of the EPZ determination to conclude that no such response actions are required beyond the EPZ boundary or outside of the facility boundary if there is no defined EPZ,"⁵³ and the preamble is to the same effect: "the EPZ size does not change the requirements for emergency planning; it only sets bounds on the planning for a predetermined, prompt response," and "[t]he capabilities within the emergency plan can support expanding the response beyond the EPZ during an actual emergency, should that prove necessary."⁵⁴ However, Section 50.160 carries that through only in part. The capabilities to recommend protective actions to offsite authorities as conditions warrant, to make notifications to organizations that may have responsibilities for responding during emergencies, and to assess and report the extent and magnitude of radiological releases apply without regard to EPZ size.⁵⁵ But all of § 50.160(b)(1)(iv)(B) remains conditioned on "a plume exposure pathway EPZ that extends beyond the site boundary," and several items in that list should not be switched off by a site-boundary determination. The most important is § 50.160(b)(1)(iv)(B)(6), "[t]he means of making offsite dose projections and the means of communicating the offsite dose projections to the offsite response coordinating agencies." Assessing a release onsite is not the same as projecting where offsite consequences will occur, and a projection capability is what an offsite authority needs from the licensee if a release exceeds what the site-boundary determination assumed. The same is true of paragraph (B)(1), the documented contacts and arrangements with the Federal, State, local, and Tribal agencies with responsibilities for coping with emergencies, and paragraph (B)(2), the means of notifying those organizations. These three are low-burden. They require no evacuation time estimate, no offsite response facility, no public alert notification system, no reentry plan, and no offsite exercise program (the items in paragraph (B) that genuinely depend on a predetermined, prompt offsite response area). BTI recommends that § 50.160(b)(1)(iv)(B)(1), (2), and (6) be relocated to § 50.160(b)(1)(iv)(A), so that they apply to every licensee under the performance-based framework with a site-boundary EPZ, and that the remainder of paragraph (B) stay conditioned on an EPZ that extends beyond the site boundary.

The same gap appears, in broader form, in Appendix E. Proposed paragraph I.5 provides that "[f]or power reactors with site-boundary EPZs or no EPZ, the degree to which compliance with the requirements of this section and sections III, IV, and V of this appendix is necessary will be

⁵³ DG-1430, app. A § A-3.9, at A-9.

⁵⁴ LOS Proposed Rule, preamble discussion, "Emergency Planning Zones," at 119.

⁵⁵ 10 CFR § 50.160(b)(1)(iii)(B), (C), (F)(4); compare id. § 50.160(b)(1)(iv)(B).

determined on a case-by-case basis.”⁵⁶ That places the emergency-plan information required in the safety analysis report, the content of the emergency plans, and the implementing procedures at case-by-case discretion for the same class of facilities, and it establishes no floor. A licensee proceeding under appendix E and § 50.47(b) with a site-boundary EPZ would therefore have less assured offsite-facing capability than one proceeding under § 50.160, where paragraphs (b)(1)(iii) and (b)(1)(iv)(A) apply regardless of EPZ size. BTI does not object to grading these requirements to the facility. It objects to grading them without a stated minimum to maintain EP as defense-in-depth. The final rule should identify, in paragraph I.5 itself, the capabilities that remain necessary irrespective of EPZ size—at a minimum the ability to characterize a release and project offsite consequences, to identify and communicate with the responsible offsite authorities, and to recommend protective actions as conditions warrant—and should provide that the case-by-case determination operates above that floor rather than upon it.

The NRC should revise § 50.47(f) so that a site-boundary EPZ does not automatically misapply § 50.47(a)(2). The final rule should preserve a clear distinction between EPZ sizing, accident screening, and emergency-response capability: frequency may inform the geographic scope of predetermined planning, but EPZ size should not operate as a categorical switch for the offsite reasonable-assurance framework.

Responses to Specific Requests for Comment

Question 12. What are the potential benefits and challenges of a change to the conduct of exercises to applicants and licensees and offsite response organizations? Please provide a basis for your response.

Changing exercise frequency could reduce unnecessary burden where emergency response organizations can demonstrate continuing proficiency through other training, drills, and performance measures. Longer intervals can also increase the risk that deficiencies, staff turnover, or weaknesses in coordination with offsite organizations are not identified promptly. The NRC should permit graded exercise frequencies based on demonstrated readiness rather than establish a longer interval as a uniform default.

Question 13. What specific criteria related to maintaining onsite ERO proficiency should be considered in evaluating whether a biennial, triennial, or quadrennial exercise is appropriate? Please provide a basis for your response.

⁵⁶ LOS Proposed Rule, proposed appendix E to 10 CFR part 50, ¶ I.5

The NRC should consider the stability and experience of the emergency response organization, turnover in key positions, performance in prior exercises and drills, the frequency and quality of intervening training, significant changes to the facility or emergency plan, and demonstrated coordination with offsite response organizations. The relevant criterion should be whether the available evidence provides reasonable assurance that required response capabilities remain proficient.

Question 14. How would a potential change impact the conduct of exercises of the offsite plans required under paragraph IV.E.2.c of appendix E to 10 CFR part 50? Please provide a basis for your response.

Changes to onsite exercise frequency should not weaken the ability to test interfaces with State, local, Tribal, and other offsite response organizations. Where offsite organizations depend on integrated exercises to demonstrate coordination, communication, protective-action decision-making, or other joint functions, the exercise schedule should continue to provide adequate opportunities to test those interfaces.

Question 15. If there was a change to the exercise frequency, should the exercise cycle be changed or otherwise be revised to ensure there is adequate opportunity for the ERO to demonstrate adequate performance under paragraph IV.E.2.j of appendix E to 10 CFR part 50? Please provide a basis for your response.

Yes. If the NRC permits longer exercise intervals, the exercise cycle should be structured so that the emergency response organization still demonstrates the range of required capabilities within a reasonable period. A longer calendar interval should not result in important response functions going untested for substantially longer periods merely because the overall exercise frequency has changed.

Question 16. Are there additional design factors or analyses that should be considered to provide certainty on EPZ determinations? Please address within the comment the technical basis for consideration and how they relate to EPZ determinations and, specifically, the criteria of 10 CFR 50.33(g)(2).

Yes. EPZ determinations should consider design and consequence characteristics that bear directly on whether predetermined, prompt protective measures would be necessary, including the potential source term, release characteristics and timing, accident progression, radionuclide retention, mitigation capability, and relevant site conditions. Thermal power is relevant but should not replace the comprehensive assessment contemplated by § 50.33(g)(2).

The most effective certainty measure available to the NRC is not an additional criterion but the specification of the existing one.

SAND2025-08913 documents that analysts making individually defensible modeling choices arrive at materially different dose exceedance distances for the same release,⁵⁷ and concludes that an analyst “should be careful to avoid overinterpreting the precision of consequence analysis metrics or emergency planning zone determinations when very simple, defensible modeling selections can substantially reduce or extend dose exceedance distances.”⁵⁸

NSIR-2017-002 adds that the choice of statistic used to reduce the weather distribution to a distance moves the result by a further factor of two to four.⁵⁹

BTI reads that finding as identifying a fixable property of the method rather than an inherent defect in a consequence-based criterion. Each of these parameters is a modeling convention, not a property of the facility. A criterion whose result depends on which defensible convention the analyst selects does not provide the certainty the NRC is seeking, and the remedy is to specify the convention rather than to substitute a different metric.

BTI recommends that DG-1430, Appendix A prescribe the screening calculation: acceptable meteorological data or synthetic default for sites not yet characterized; a default plume meander model and surface roughness treatment; default breathing rates and dose coefficient source; and the statistic used to reduce the weather distribution to a single distance. Appendix A already addresses each of these parameters, but as considerations to be documented rather than as specified defaults.⁶⁰ Converting them into a prescribed screening basis, with departures permitted on justification, would give applicants and the staff a reproducible result without foreclosing site-specific analysis.

Recommendations

1. **The NRC should remove the technology-class limitation on § 50.160, but should not finalize eligibility by election alone as proposed.** It should make the performance-based pathway available where reactor-specific or generically applicable evidence demonstrates

⁵⁷ SAND2025-08913, Tables 3-1 and 3-3, at 21, 24.

⁵⁸ SAND2025-08913, § 4, at 49.

⁵⁹ NSIR-2017-002 Task 1, § 4, at 42.

⁶⁰ DG-1430, app. A §§ A-3.3, A-3.4, A-3.5, at A-4.

that the alternative framework will preserve the protective capabilities needed for the facility's accident spectrum. Frequency should remain an input to that determination, not a categorical screening criterion. Accident scenarios may appropriately be excluded from quantitative EPZ sizing when technically justified, but that exclusion should not automatically remove them from emergency planning. The governing question should remain whether, if a radiological emergency occurs, the licensee and relevant response organizations have the capabilities necessary to assess conditions, communicate, recommend or implement protective actions, and provide reasonable assurance that adequate protective measures can and will be taken.

2. **The NRC should also clearly state that consequence criteria used to determine EPZ size are planning criteria, not independent design limits, and should not be treated as requiring design changes solely to remain below the EPZ screening value.** The NRC should remove the 300 MWt threshold from proposed §§ 50.33(g)(1), 50.47(c)(2), and 53.1109(g), and retain the supporting 300 MWt analysis in the Statement of Considerations. The regulatory text should continue to allow EPZ determinations based on the consequences of the facility and the need for predetermined, prompt protective measures, without making thermal power itself the regulatory dividing line.
3. **The NRC should make corresponding changes to the implementing guidance.** DG-1430 Appendix A should present dose-distance results per source term, conditional on the occurrence of that source term, and treat any frequency-weighted aggregation as supplementary rather than as the basis for the EPZ determination; should prescribe the screening calculation in §§ A-3.3 through A-3.5 rather than leaving the meteorological data source, plume meander model, surface roughness treatment, breathing rate, dose coefficient source, and reduction statistic to the analyst; should conform the no-EPZ trigger in § A-3.8 to the 96-hour exposure period in the rule; and should provide that a sequence may be set aside when no admissible resolution of the remaining uncertainty would change the required response capability or the area within which predetermined, prompt protective measures must be planned, rather than when its frequency falls below a threshold. The NRC should also carry the premise stated in DG-1457 into DG-1430: that emergency preparedness “is carried out as a matter of prudence rather than in response to a quantitative analysis of accident probabilities,” and that “[t]he effectiveness of an

emergency plan is independent of probability.”⁶¹ Finally, the final rule should state a capability floor in Appendix E, paragraph I.5 above which the case-by-case determination operates.

X. PART 100 SITING MODERNIZATION

BTI supports the NRC’s proposed modernization of Part 100 and its shift toward a graded, consequence-informed approach to site characterization. The existing siting framework can require substantially similar site investigations for reactor designs with materially different radiological consequences. The proposed Tier 1 and Tier 2 framework would allow lower-consequence reactors to use the less prescriptive requirements in Subpart A while retaining the more detailed Subpart B requirements for higher-consequence reactors. That basic structure better aligns the scope of site characterization with the potential consequences of the facility and should be preserved.

A. The NRC Should Clarify Treatment of Uncertainty in the Unmitigated Consequence Analysis Used to Determine Tier and NSC

BTI supports the NRC’s proposed consequence-based approach to siting and the use of an unmitigated consequence analysis to determine reactor Tier and Nuclear Siting Category (NSC). DG-4036 already provides substantial methodological direction. It identifies ANSI/ANS-2.26-2004 (R2021) § 6.2 as the baseline methodology, limits credit to inherent physical or chemical characteristics of the hazardous material and available dispersal energy, prohibits credit for mitigating and robust features in determining NSC, and provides a source-term framework based on material at risk, airborne release fraction, and respirable fraction.

The remaining ambiguity concerns how uncertainty in those inputs affects classification. DG-4036 directs applicants to use nominal or best-estimate ARF and RF values based on the material characteristics and credible dispersal mechanisms. It also recognizes that the available data may contain substantial uncertainty and states that additional margin may be prudent

⁶¹ U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1457, Guidance on Making Changes to Emergency Plans for Nuclear Power Reactors (Proposed Revision 2 to Regulatory Guide 1.219), ML25289A351 (July 2026), § 1.3, at 12.

where uncertainty is high. The guidance does not, however, establish how that uncertainty should be treated when it could change the resulting Tier or NSC.

The proposed framework assigns discrete regulatory categories based on consequence. Tier 1 applies below 25 rem TEDE at the exclusion area boundary, while reactors above that value or with undetermined consequences remain in Tier 2. DG-4036 further divides the consequence range into NSC-1 through NSC-5, including boundaries at 5, 25, and 100 rem. If reasonable uncertainty in an input cannot move the calculated consequence across one of those boundaries, further refinement of that uncertainty cannot change the classification. Where the uncertainty could cross a boundary, by contrast, the treatment of that uncertainty can determine the applicable Tier or NSC and the resulting scope of site characterization.

The guidance should preserve its nominal or best-estimate approach rather than allow uncertainty to drive applicants toward uniformly bounding assumptions. DG-4036 itself recognizes that traditional accident-analysis references often provide bounding ARF and RF values and specifically directs applicants toward nominal or best-estimate values appropriate to the material and credible dispersal mechanism. Requiring bounding inputs whenever uncertainty exists would undermine that approach and could move an applicant into a higher category even where additional conservatism is not necessary to resolve the regulatory decision. At the same time, an applicant whose reasonable uncertainty spans a Tier or NSC boundary should provide enough additional information to support the lower classification or conservatively use the higher category.

This clarification is important because Tier and NSC affect the subsequent scope of siting review. DG-4036 grades meteorological, hydrologic, seismic, and other site-characterization activities according to the reactor's category, including reduced meteorological requirements for lower-consequence reactors and alternative seismic treatment for qualifying Tier 1, NSC-3 applicants. A consistent treatment of classification-sensitive uncertainty would help ensure that those graded requirements track meaningful differences in consequence rather than differences in how applicants choose to add conservatism to uncertain inputs.

Response to Specific Request for Comment

Question 42. The proposed rule introduces a new graded approach to siting that provides entry criteria for the less restrictive requirements in subpart A to 10 CFR part 100 (i.e., demonstrating an unmitigated consequence of

less than 25 rem (0.25 Sv) TEDE at the site exclusion area boundary) as an alternative to the prescriptive siting requirements in subpart B. The associated draft guidance provides detailed information on how to evaluate this entry criteria. Given the significance of this new graded approach to siting, are there implications in terms of implementation of this new approach that should be further explored in guidance or addressed at the final rule stage?

Yes. The NRC should further address implementation of the graded siting framework in the final rule and associated guidance. DG-4036 should clarify how uncertainty in the unmitigated consequence analysis should be treated when it could affect a Tier or NSC determination, while avoiding decision-irrelevant analytical refinement where the classification is unaffected. The NRC should also ensure that graded site characterization extends to the evidentiary burden itself by allowing qualified existing information and bounding analyses where they are sufficient to support the relevant licensing finding.

B. Siting Near Population Centers Should Not Apply a Separate Societal-Benefits Test

BTI supports removing categorical barriers to siting reactors in areas of higher population density. The proposed rule replaces the existing population-distance restriction with a new condition that is not well aligned with the AEA. Under proposed § 100.21(b), a site that does not satisfy the prescribed population-center distance may be found acceptable only through an assessment comparing the site's societal risks with its societal benefits.⁶² However, proposed § 100.21(h) would retain the NRC's preference for sites away from very densely populated centers and for areas of low population density unless a different site is justified through consideration of safety, environmental, economic, or other factors.⁶³

The NRC should remove the underlying preference in § 100.21(h). If population density affects accident consequences, emergency planning, evacuation feasibility, or another safety-relevant consideration, those effects should be addressed directly through the applicable consequence and emergency-planning requirements. A general preference for lower-population sites adds

⁶² Nuclear Regulatory Commission, *Modernizing Reactor Licensing, Safety Oversight, and Siting Practices*, 91 Fed. Reg. 44,560, 44,716 (July 16, 2026) (proposed 10 CFR § 100.21(b)(1)).

⁶³ Nuclear Regulatory Commission, *Modernizing Reactor Licensing, Safety Oversight, and Siting Practices*, 91 Fed. Reg. 44,560, 44,716 (July 16, 2026) (proposed 10 CFR § 100.21(h)).

little once the NRC has established requirements capable of determining whether a specific design can be safely located at a specific site, and separately requires EP as defense-in-depth.

The NRC should remove that additional benefits test under 100.21(b), 100.10(e), 100.11(a)(3)(i)(B), and other referencing sections. The central safety question under the AEA is whether a facility can be licensed consistently with reasonable assurance of adequate protection of public health and safety and the common defense and security. AEA Section 103 authorizes commercial reactor licenses subject to safety standards established by the NRC and provides that a license may not be issued where issuance would be inimical to the common defense and security or the health and safety of the public.⁶⁴ That balance is also reflected in the NRC's mission, as revised pursuant to § 501 of the ADVANCE Act, which directs the agency to regulate civilian nuclear energy efficiently and in a manner that does not unnecessarily limit its benefits to society.⁶⁵

The NRC can and must consider benefits to society in decision-making.⁶⁶ However, the AEA assigns protection and benefit to different operations, and the proposed provision performs neither. Adequate protection is a condition upon the grant of a license. Section 182(a) requires an application to contain the information enabling the Commission to find that the proposed activity “will provide adequate protection to the health and safety of the public,”⁶⁷ and the companion licensing provision bars issuance that would be inimical to the health and safety of the public. That is a threshold to be satisfied, not a quantity to be traded. Adequate protection is cost-blind: if a measure is necessary to provide adequate protection of public health and safety, the NRC may not decline to require it because its costs exceed its benefits, although cost may inform the choice among alternative measures that each provide adequate protection. Societal benefit enters through two further distinct obligations.⁶⁸ Section 2011(a) directs that the development, use, and control of atomic energy “be directed so as to make the maximum contribution to the general welfare, subject at all times to the paramount objective of making the maximum contribution to the common defense and security,”⁶⁹ which is a direction to pursue the

⁶⁴ Atomic Energy Act of 1954 § 103(a), (b), (d), 42 U.S.C. § 2133(a), (b), (d).

⁶⁵ Nuclear Regulatory Commission, SECY-24-0083, Mission Statement Update Options Pursuant to Subsection 501(A) of the Advance Act of 2024, ML25024A040.

⁶⁶ Adam Stein, The Breakthrough Institute, Considering Nuclear Energy's Benefit to Society, November 2024, <https://thebreakthrough.org/issues/energy/considering-nuclear-energys-benefit-to-society>

⁶⁷ 42 U.S.C. § 2232(a) (Atomic Energy Act § 182(a)).

⁶⁸ 10 CFR § 50.109(a)(4)(ii), (a)(5), (a)(7); see also 10 C.F.R. § 53.1590(a)(4)(ii), (a)(5), (a)(7).

⁶⁹ 42 U.S.C. § 2011(a) (Atomic Energy Act § 1(a)).

welfare gains available among protection-adequate options. Section 501 of the ADVANCE Act adds a different and exclusionary requirement: that licensing and regulation be conducted “in a manner that is efficient and does not unnecessarily limit” the civilian use of radioactive materials and deployment of nuclear energy, or “the benefits of civilian use of radioactive materials and nuclear energy technology to society.”⁷⁰ The first obligation is violated by omission, where the Commission declines a welfare gain it could have taken without compromising protection. The second is violated by commission, where the Commission adopts a design that limits a covered benefit more than the safety objective requires and a less-limiting but equally protection-adequate alternative was available. The two fail independently and at different points in the decision.

Proposed § 100.21(b) implements neither obligation. It retains the population-based restriction as the default and requires the applicant to request relief from it by demonstrating compensating societal benefit. That converts societal benefit from an objective the Commission is directed to pursue, and from a value the Commission is forbidden to limit unnecessarily, into a burden of proof assigned to the applicant. It also exempts the restriction itself from the only test § 501 supplies. The question § 501 poses is whether the Commission’s chosen regulatory design unnecessarily limits a covered benefit relative to the protection-adequate alternatives before it, and the Commission should be able to articulate why the design it selected does not. Proposed § 100.21(b) asks a different question, of the wrong party, at the wrong stage.

An applicant that satisfies the applicable health-and-safety requirements does not need to, and should not be required to, establish an additional quantum of societal benefit merely because the reactor is located closer to population. Adequate protection is a safety standard, not a requirement to demonstrate that the chosen site is the safest possible location or that any incremental difference in societal risk is affirmatively offset by unrelated economic, environmental, or infrastructure benefits. Population and site characteristics should affect the licensing decision when impacting the NRC’s safety findings. They should not create a separate public-interest balancing test once those findings can be made.

The question should be whether a population-based restriction is necessary to achieve the NRC’s safety and security objectives. Under the proposed framework, an applicant may instead have to justify relief from that restriction by showing enough compensating benefit. A reactor that can

⁷⁰ Accelerating Deployment of Versatile, Advanced Nuclear for Clean Energy Act of 2024, Pub. L. No. 118-67, § 501(a).

protect public health and safety at a site near population centers should not face an additional licensing burden merely because an otherwise acceptable site does not conform to a historical siting preference. A rule that restricts an otherwise acceptable site unless an applicant can affirmatively prove additional societal benefits risks doing exactly that. Sites near load centers, industrial users, existing transmission, retiring fossil facilities, or communities seeking new generation may provide substantial advantages. The NRC should not require those benefits to serve as compensation for locating a reactor at a site that already satisfies the applicable safety standards.

For all of these reasons, the NRC should not require an applicant to justify siting near a population center by demonstrating additional societal benefits. That requirement is inconsistent with the direction Congress provided in Section 501 of the ADVANCE Act and should not be promulgated. It would also create a problematic precedent for future rulemakings. Section 501 directs the NRC to conduct its regulatory mission in a manner that does not unnecessarily limit nuclear deployment or the societal benefits of nuclear energy.⁷¹ It does not make those benefits an additional burden that applicants must affirmatively prove before the NRC may approve an otherwise acceptable licensing outcome. Requiring such a showing here would invert that statutory direction: societal benefits would become a condition imposed on the applicant rather than a consideration informing how the NRC exercises its regulatory authority. Once embedded in Part 100, that approach could be carried into other licensing frameworks, allowing the NRC to condition regulatory flexibility on applicant-specific demonstrations of societal benefit even where the applicable safety requirements are satisfied. That would recreate, in a different form, the type of regulatory limitation Section 501 directs the NRC to avoid.

Proposed § 100.21(h) also creates an administrability problem without providing a corresponding safety benefit. The proposed rule identifies several possible considerations such as access to skilled labor, transmission infrastructure, transportation, seismic characteristics, reuse of retired fossil sites, and reduced environmental effects that all impact the NRC's decision. Those considerations are qualitatively different and do not share a common metric. The rule does not establish how much benefit is sufficient, how benefits should be compared with incremental societal risk, or how competing economic, environmental, and safety considerations should be

⁷¹ Accelerating Deployment of Versatile, Advanced Nuclear for Clean Energy Act of 2024 (ADVANCE Act), Pub. L. No. 118-67, § 501(a), 138 Stat. 1447, 1471 (2024).

weighed when comparing sites at varying distances to population centers. The result could be an open-ended discussion over whether an otherwise safe site is sufficiently beneficial.

The proposed revisions to Part 100 are intended to provide greater siting flexibility by scaling site-characterization requirements to reactor consequence and allowing consideration of higher-population-density sites where warranted. Although BTI disagrees with some changes, the intent is justified. The revisions to Part 100 will not provide the intended outcome if Regulatory Guide 4.7 rev. 4⁷² calculation of 500pp/sq. mi. density metric is not also updated based on BTI's past recommendations.⁷³ The staff incorrectly dispositioned BTI's comment on the docket. Instead, the RG 4.7 implements a distance multiplier that is arbitrary, not scaled to societal risks or benefits, and will be the functional limitation for new reactor siting. The current methodology, including its distance-based application of the population-density criterion, can substantially reduce the universe of sites considered suitable before the applicant reaches the more risk-informed, site-specific analysis contemplated by the proposed Part 100 revisions.⁷⁴

Population growth around existing nuclear sites illustrates why the NRC should not treat population density as an independent proxy for site suitability for advanced reactors. BTI's analysis of population data for existing nuclear facilities found that approximately 15 percent exceed the NRC's 500-persons-per-square-mile criterion within 10 miles, and 17 percent exceed the criterion at some distance from the site, reflecting in part population growth since the sites were originally approved.⁷⁵ If population growth has caused a meaningful share of sites that already host operating or formerly operating nuclear facilities to no longer satisfy the NRC's preferred population criteria, applying that preference to a new SMR can exclude an established nuclear site even where the proposed reactor presents substantially lower offsite consequences than the

⁷² Nuclear Regulatory Commission, *General Site Suitability Criteria for Nuclear Power Stations*, Regulatory Guide 4.7, Revision 4, February 2024, [ML23348A082](#).

⁷³ Adam Stein, The Breakthrough Institute, *Comment on Draft Regulatory Guide DG-4034: General Site Suitability Criteria for Nuclear Power Stations*, Docket ID NRC-2023-0153, October 18, 2023, [ML23326A032](#).

⁷⁴ Spencer Toohill and Adam Stein, The Breakthrough Institute, *Population Density Considerations at the United States Nuclear Regulatory Commission*, *Transactions of the American Nuclear Society*, Vol. 133, No. 1, pp. 514–517 (Nov. 2025). <https://www.ans.org/pubs/transactions/volume-133/>; J. Hansen et al., Idaho National Laboratory, *Investigating Benefits and Challenges of Converting Retiring Coal Plants into Nuclear Plants*, INL/RPT-22-67964, Rev. 2, Sept. 13, 2022, <https://sai.inl.gov/content/uploads/29/2024/11/c2n2022report.pdf>; Muhammad R. Abdussami, Kevin Daley, Gabrielle Hoelzle, and Aditi Verma, *Investigation of Potential Sites for Coal-to-Nuclear Energy Transitions in the United States*, 11 *Energy Reports* 5383–5399 (2024), <https://www.sciencedirect.com/science/article/pii/S2352484724002993>.

⁷⁵ Adam Stein, The Breakthrough Institute, *Comment on Draft Regulatory Guide DG-4034: General Site Suitability Criteria for Nuclear Power Stations*, Docket ID NRC-2023-0153, October 18, 2023, [ML23326A032](#).

reactor for which the site was originally approved. Population growth around an existing site does not by itself establish that a lower-consequence reactor cannot be safely deployed there. The relevant regulatory question should be whether the proposed reactor, at the site as it exists today, satisfies the applicable safety and emergency-planning requirements.

The same issue applies to repowering existing fossil-generation sites. DOE's coal-to-nuclear study found that 80 percent of screened retired and operating coal sites were amenable to advanced-reactor siting and identified population density as an important discriminator for backfit feasibility.⁷⁶ These sites may provide existing transmission, transportation, industrial infrastructure, and experienced energy workforces—the same types of site-specific benefits the proposed revisions to § 100.21 recognize. The NRC's siting framework should not foreclose such locations based primarily on population proximity where the applicant can demonstrate acceptable site-specific consequences.

C. Meteorological Site-Characterization Requirements Should Be Graded to the Licensing Decision

BTI supports DG-1463's objective of allowing qualified applicants to rely on offsite meteorological information, but the final guidance should ensure that this flexibility is usable in practice. As discussed in BTI's separate comment on DG-1463, the regulatory finding concerns whether the applicant has provided sufficient, technically justified information to support the applicable dose, licensing, and emergency-preparedness findings.⁷⁷ A requirement for onsite monitoring should follow from a need that cannot otherwise be satisfied, not operate as the default means of demonstrating adequacy.

DG-1463's pairwise comparisons among offsite stations can provide useful information about regional consistency, but they do not by themselves establish the error associated with using either station to characterize an unmeasured reactor site. The numerical agreement criteria should accordingly function as screening indicators rather than stand-alone pass/fail gates. Where the screening values are met, further analysis may be unnecessary. Where they are not, an

⁷⁶ U.S. Department of Energy, *Investigating Benefits and Challenges of Converting Retiring Coal Plants into Nuclear Plants*, INL/RPT-22-67964, September 2022.

⁷⁷ Breakthrough Institute Comment on Draft Regulatory Guide DG-1463, "Meteorological Monitoring Programs for Nuclear Power Plants" (Proposed Revision 2 to Regulatory Guide 1.23).

applicant should remain able to demonstrate adequacy directly on the quantity that controls the regulatory finding, including by comparing the resulting dispersion factors or adopting a demonstrably conservative bounding value. For safety-related analyses, the demonstration should also distinguish conservative from non-conservative differences rather than treating discrepancies in either direction as equivalent.

The most important enabling change is to define the bounding pathway DG-1463 already contemplates. At the construction-permit or ESP stage, an applicant should be able to support the preliminary licensing finding with a demonstrably conservative dispersion factor when greater meteorological precision cannot affect that finding. The absence of a suitable nearby monitoring station should require greater conservatism or other technical justification; it should not automatically require one or two years of onsite monitoring before an application can be submitted. That default is poorly matched to advanced-reactor deployment models and imposes delay before NRC review begins without necessarily improving the preliminary safety determination. Site-specific information can then confirm the preliminary basis at the OL or COL stage when a more mature finding is required.

The NRC should also grade meteorological requirements to the function the information performs and the consequence significance of the determination. Permitting requires information sufficient to support a preliminary dose analysis; final licensing requires confirmation of the as-sited plant; operations may require information for routine releases, effluent monitoring, and receptor-specific determinations; and emergency preparedness may require reliable real-time meteorological information before fuel load. These functions do not require identical instrumentation, record lengths, or timing. For facilities with limited offsite consequences, the meteorological information needed to support a licensing decision may be correspondingly limited, and the guidance should not require detailed dispersion analysis where additional meteorological precision cannot affect the applicable finding.

The same graded principle applies to confirmatory measurements. Where regional or offsite information is sufficient for the preliminary finding but some site-specific uncertainty remains, the applicant should be able to resolve that uncertainty through appropriately scoped later confirmation rather than a full pre-application RG 1.23 monitoring program. Conversely, a facility that does not need an extensive climatological record for its dose analysis may still need relatively simple real-time wind information for emergency response. The guidance should

separate those functions rather than allowing the need for one to trigger the requirements associated with all of them.

Recommendations

1. **The NRC should revise DG-4036 to distinguish between uncertainty that can and cannot affect the Tier or NSC determination.** Where reasonable uncertainty is decision-irrelevant (cannot change the applicable category), additional refinement should not be required solely to reduce that uncertainty. Where uncertainty could change the category, the applicant should either refine the material inputs sufficiently to support the selected classification or use the higher category. The guidance should also require applicants to identify the material uncertainties affecting the unmitigated consequence calculation and explain whether they could alter the resulting Tier or NSC. This approach would preserve DG-4036's nominal, best-estimate methodology while directing additional analytical effort to uncertainty that can actually change the siting decision.
2. **The NRC should revise § 100.21 so that siting near population centers is permitted whenever the applicant demonstrates adequate protection and compliance with the applicable health, safety, security, emergency-planning, and site-specific requirements. The NRC should remove the requirement to demonstrate offsetting societal benefits and remove the general low-population siting preference.** If a particular population characteristic creates a safety-relevant concern, the NRC should require the applicant to address that concern directly rather than requiring a broader showing that the site produces sufficient societal benefits to compensate for its location.
3. **The NRC should revise DG-1463 to (1) define an acceptable bounding demonstration, (2) eliminate the onsite-monitoring program as the default response to inadequate offsite station coverage, (3) grade the required meteorological showing to the licensing stage and regulatory function, and (4) retain the pairwise statistical criteria as screening tools rather than dispositive acceptance gates.** The ultimate determination should focus on whether the meteorological basis is sufficiently representative or conservative to support the applicable regulatory finding. BTI's separate comment on DG-1463 provides detailed recommendations concerning the numerical criteria, station selection and terrain

differences, statistical treatment, model-specific wind-speed behavior, averaging intervals, and implementation of the offsite-data pathway.⁷⁸

XI. CROSS-RULE COORDINATION

The proposed rule is one component of a broader NRC effort to modernize interdependent parts of the licensing and regulatory framework. At the same time, the NRC is separately revising contested-adjudication procedures, environmental-review requirements, materials licensing, security requirements, licensing pathways for microreactors and other reactors with comparable risk profiles, and more. These reforms address different regulatory functions, but they will operate together for the same applicants, designs, and licensing actions. The benefits of any one reform will depend in part on whether the adjacent rules use compatible definitions, preserve the same prior regulatory findings, and place procedural and substantive requirements in a workable sequence.

The NRC should treat these interfaces as part of the design of the final regulatory framework, not as issues to be resolved only after the individual rules are implemented. The concern is not simply that several rulemakings are proceeding at the same time. It is that differences among them can determine when construction may begin, which prior approvals remain usable, what issues remain open in a later licensing or adjudicatory proceeding, and whether a streamlined pathway actually reduces the critical path. Where the same concept or prior NRC finding operates across multiple Parts, the final rules and associated guidance should identify the relationship expressly and preserve pathway-specific requirements only where they reflect a substantive regulatory difference.

The following issues warrant specific coordination: the relationship between construction and environmental-review requirements; portability of generic technical and programmatic approvals across Parts 50, 52, 53, and 57; and alignment of Part 2 adjudicatory procedures with the broader generic-finality and repeat-deployment framework.

A. Construction and Environmental-Review Boundaries Should Operate as a Coordinated Framework

⁷⁸ Breakthrough Institute Comment on Draft Regulatory Guide DG-1463, "Meteorological Monitoring Programs for Nuclear Power Plants" (Proposed Revision 2 to Regulatory Guide 1.23).

Section II.A explains why the NRC should apply a consistent, function-based construction boundary across licensing pathways. The cross-rule issue is that the definition of construction determines not only when NRC construction authorization is required, but also how construction sequencing interacts with Part 51 environmental review and the NRC's expanding set of pathways for early site work. Those requirements should operate as one coordinated framework.

Part 51 is an example of the potential inconsistency. As BTI explained in its recent NEPA comment, proposed § 51.4 would look to the applicable licensing part to determine what constitutes construction, making the environmental-review boundary directly dependent on the construction definitions adopted in Parts 50, 53, 57, and 70.⁷⁹ Different substantive definitions can therefore cause functionally comparable onsite work to require prior NRC authorization under one licensing pathway but remain preconstructed under another. For example, foundation work for a non-safety-related SSC credited with a safety-significant or defense-in-depth function could fall within the proposed Part 50 construction boundary because the work may affect that required function, while Part 57 does not expressly extend its construction definition to the broader category of safety-significant SSCs. The resulting difference would affect not only construction sequencing but also which activities require NRC authorization and associated Part 51 review. The NRC should apply a consistent functional principle across these frameworks, with differences retained only where required by the AEA or justified by a distinct safety or licensing finding.

The need for a consistent definition of construction is particularly important as the NRC creates additional pathways for early site work. Applicants may proceed with activities outside the construction boundary, seek a limited work authorization, or, if eligible, rely on the proposed general construction license. Those pathways involve different licensing and environmental prerequisites. Variations or conflicts across licensing regimes threaten the predictability and implementation of NEPA. DG-4037, "Preparation of Environmental Reports for Nuclear Power Stations," ML25268A122 was part of the rulemaking but withdrawn due to conflicts with the NEPA rulemaking.⁸⁰

⁷⁹ The Breakthrough Institute, *Comment on Implementation of the National Environmental Policy Act*, Docket ID NRC-2025-0478, Comment ID NRC-2025-0478-0893, August 21, 2026, at 40–41, <https://www.regulations.gov/comment/NRC-2025-0478-0893>.

⁸⁰ Nuclear Regulatory Commission, *Implementation of the National Environmental Policy Act*, Docket ID NRC-2025-0478, 91 FR 42086.

Without cross-rule consistency, the differences in construction boundaries affect not only what actions can be performed by the applicant before a construction permit, but the boundaries of environmental reviews. Their schedule value depends on applicants being able to determine before work begins which authorization is required and what environmental review must precede it. Changes to Parts 40, 50, 53, 57, or 70 and the current proposed changes to §§ 50.10 and 53.1130 should be implemented as parts of the same regulatory architecture rather than as independent procedural reforms.

B. Generic Technical and Programmatic Approvals Should Remain Portable Across Licensing Pathways

The NRC's modernization efforts increasingly depend on resolving technical and programmatic issues once and carrying those findings forward to later deployments. Yet the individual licensing frameworks do not consistently specify how an NRC finding obtained under one Part can be used under another. Proposed § 57.18(b), for example, permits a Part 57 applicant to incorporate applicable NRC approvals issued under Parts 50 and 52 but does not include Part 53.⁸¹ More broadly, Parts 50, 52, 53, and 57 are developing parallel mechanisms for generic finality, manufacturing, and repeat deployment without a common rule for recognizing equivalent prior findings.

This gap is most consequential at the FOAK-to-NOAK transition. A first plant may appropriately use Part 50 or Part 53 while a design is still being refined, then transition to Part 57 or another standardized pathway for repeat deployment. The reverse may also occur. BTI's Part 57 comment recommended an explicit mechanism allowing prior work under Parts 50, 52, and 53 to carry into Part 57, with subsequent review focused on what is new, changed, unresolved, or site-specific.⁸² BTI's Part 53 comment likewise emphasized preserving the transferability of prior licensing work

⁸¹ U.S. Nuclear Regulatory Commission, *Modernizing Reactor Licensing, Safety Oversight, and Siting Practices*, Docket NRC-2025-0975, proposed changes concerning generic finality, manufacturing licenses, § 50.55a alternatives, and Appendix T.

⁸² The Breakthrough Institute, *Comment on Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles*, Docket ID NRC-2025-0379, Comment ID NRC-2025-0379-0123, June 15, 2026, <https://www.regulations.gov/comment/NRC-2025-0379-0123>.

and regulatory knowledge across pathways.⁸³ The applicant's initial choice of licensing path should not determine whether an otherwise relevant NRC finding retains regulatory value.

The same issue extends beyond design approvals. Manufacturing-license information, approved codes and standards, § 50.55a alternatives, and QA/QMS findings may all support later deployments under a different framework. If the NRC approves a reactor manufacturing process, associated design requirements, a particular code or standard, or an alternative to a § 50.55a requirement as part of an ML, later site applications should ordinarily rely on that determination. Reconsidering the generic acceptability of the same manufacturing code or design standard at every deployment would erode the finality that the manufacturing license is intended to provide.

The governing requirements are not necessarily interchangeable: a § 50.55a alternative does not automatically satisfy Part 53 or Part 57, and a Part 57 QA program does not automatically satisfy Appendix B or Appendix T. But the technical findings underlying those approvals should remain reusable where the relevant design, functions, manufacturing conditions, and technical basis remain applicable. Similarly, a FOAK reference plant should not be disqualified solely because it was licensed under a different Part or used a different QA/QMS framework if its operating and manufacturing experience provides the technical basis required for the later deployment.

Without a common recognition principle, developers may have to repeat review of the same design methods, code selections, manufacturing controls, suppliers, or quality programs solely because a deployment crosses a regulatory boundary. NRC could also develop parallel bodies of technically equivalent approvals under Parts 50, 53, and 57, reducing the value of manufacturing-license finality and fleet standardization.

The NRC should establish a common cross-pathway recognition framework based on equivalency and gap analysis. A prior NRC finding should remain creditable where the underlying design, analysis, program, or technical basis is materially unchanged, the earlier review resolved the matter at issue, and the receiving framework does not impose a substantively different requirement. Where requirements differ, the applicant should identify and address that gap rather than reproduce the original showing. NRC guidance should explain how this principle

⁸³ The Breakthrough Institute, *Comment on 10 CFR Part 53: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors*, NRC-2019-0062, February 28, 2025, [ML25069A179](#).

applies to design and manufacturing findings, codes and standards, and QA/QMS approvals across Parts 50, 52, 53, and 57.

C. Reactor-Side Fuel Qualification Should Be Coordinated With Storage and Transportation Requirements

The NRC should coordinate reactor-side fuel modernization with the separate requirements governing transportation and storage. Proposed § 50.46a would establish reactor-side acceptance criteria under which fuel performance is evaluated in part through continued core coolability, including treatment of fuel fragmentation, relocation, and dispersal.⁸⁴ As discussed in Section VII, qualification under that reactor-side framework does not by itself determine how the resulting fuel condition will be treated after discharge.

Spent-fuel storage and transportation are governed by separate requirements in Parts 72 and 71. Those frameworks make their own findings concerning confinement, retrievability, criticality, containment, and the condition or configuration of spent fuel. A fuel condition that is acceptable for reactor operation because the core remains coolable may require additional analysis or controls when the same fuel is later placed in storage or transportation. That does not establish that the fuel would be unacceptable under Parts 71 or 72; it means that the § 50.46a finding and the later storage or transportation findings are not interchangeable.

Storage and transportation requirements for this fuel should be scaled to its demonstrated condition and inventory rather than to assumptions developed for intact, large-cask spent-fuel populations. A single dispersed or fragmented high-burnup assembly presents a different confinement problem than the debris-bed and gross-rupture provisions in Part 72 were developed to address, and the final rule should not treat fuel qualified as coolable under § 50.46a as presumptively “damaged fuel” for storage and transportation purposes without a fuel-specific technical basis for that conclusion.

Rather than leave that classification question unresolved, the NRC should identify whether the risk-informed, performance-based demonstration pathway it has proposed elsewhere can supply the basis. BTI’s comment on the proposed package certification rule recommended that the

⁸⁴ U.S. Nuclear Regulatory Commission, *Modernizing Reactor Licensing, Safety Oversight, and Siting Practices*, Docket NRC-2025-0975, discussion of increased-enrichment fuel, fuel-storage criticality, and fissile-material packaging requirements.

Commission clarify whether the alternative-demonstration pathway in proposed § 71.41(e), or a parallel one, should be clarified to the containment and criticality-safety findings implicated by dispersed or degraded fuel would give applicants a defined route to demonstrate Part 71 and Part 72 compliance for fuel qualified under the new § 50.46a criteria.⁸⁵

Two core points come from the proposed Part 57 rule. First, the general license for irradiated-fuel disposition under Part 72, transportation authorization under Part 71, and manufacturing-site storage authorization do not currently connect for a module or assembly that does not remain on the deployment site, leaving a potential authorization gap between operation and return. Second, a discharged fuel assembly or reactor module may pass through several possession states and the proposed rules do not specify who bears Nuclear Materials Management and Safeguards System reporting responsibility, or how possession handoffs are documented, at each transition.⁸⁶ Fuel qualified as coolable under proposed § 50.46a because it disperses or degrades rather than remaining intact is more likely to require exactly this kind of downstream handling, which makes resolving both gaps directly relevant to this rulemaking rather than a matter this rulemaking can leave to Part 57 alone.

Decommissioning funding assurance raises a related question. Where discharged fuel is removed for storage or returned to a manufacturer or centralized facility, BTI has recommended that the Commission specify when the operating-site licensee's funding obligation is reduced or discharged and what funding assurance the receiving facility must hold for the transferred inventory. The same question follows directly from this rulemaking's fuel-dispersal provisions to the extent they increase the likelihood that discharged fuel leaves the reactor site in a condition, or on a schedule, that existing decommissioning-funding mechanics did not anticipate.

The NRC should provide a coordinated pathway identifying when Part 71 and, where applicable, Part 72 approvals should be obtained relative to reactor and manufacturing licenses; which reactor-side design and safety findings may be referenced in those reviews; and which transportation- or storage-specific findings must be made independently. For transportable reactors, the NRC should also clarify how possession, irradiated-fuel management, and related responsibilities transition when an irradiated reactor module leaves the deployment site and is transferred to a manufacturer or other authorized recipient.

⁸⁵ The Breakthrough Institute, *Comment on Modernizing Package Certification Requirements*, Docket ID NRC-2025-1667; RIN 3150-AL62, August 19, 2026, § II (Recommendation 7).

⁸⁶ LOS Proposed Rule, § VIII.4.

D. Procedural Modernization Should Preserve Substantive Licensing Finality

The NRC should align its Part 2 adjudicatory reforms with the generic-finality and repeat-deployment mechanisms being established across Parts 50, 52, and 53. The value of resolving generic matters once depends not only on substantive issue finality, but also on whether later adjudications recognize that finality and focus on the limited matters that remain open. The proposed rule provides a hearing opportunity when generic finality is first established while preventing resolved generic matters from being re-adjudicated in later proceedings that reference the approval.

The NRC's separate proposed rule on *Streamlining Contested Adjudications in Licensing Proceedings* pursues the same objective procedurally by providing expedited treatment where prior NRC approvals have already resolved substantial portions of the licensing basis. BTI previously urged the NRC to coordinate that rulemaking with Part 53 so that independently developed procedural changes do not produce inconsistent treatment across licensing pathways.⁸⁷ For example, proposed § 2.1207 provides an expedited schedule where an application references both a categorical exclusion and an NRC approval providing adjudicatory finality to design information, including a design certification or manufacturing license.

The NRC should treat the broader modernization effort as an integrated process rather than a series of isolated rulemaking dockets and specifically identified Part 53 as an example where independently developed procedural changes could create inconsistent results. At the time, both the Part 53 rule and the contested-adjudications proposal were revising the definition of "contested proceeding" in § 2.4 using different formulations. BTI recommended that the NRC reconcile overlapping Part 2 amendments and clarify how Part 2 procedures incorporated into Part 53 would operate as the two frameworks evolved.

The two rulemakings are not yet fully aligned. The Part 2 proposed rule was structured around existing Parts 50 and 52 licensing pathways and does not provide equivalent treatment for Part 53 proceedings, while the present proposed rule would now create generic-finality mechanisms under Part 53 as well. An application that relies on an equivalent generically resolved design

⁸⁷ The Breakthrough Institute, *Comment on Streamlining Contested Adjudications in Licensing Proceedings*, Docket ID NRC-2025-1501, Comment ID NRC-2025-1501-0031, <https://www.regulations.gov/comment/NRC-2025-1501-0031>.

should not receive materially different adjudicatory treatment solely because the underlying approval or subsequent license proceeds under Part 53 rather than Part 50 or 52. The same principle should apply as new Part 57 high-volume licensing mechanisms are integrated into Part 2.

Recommendations

1. **The NRC should implement construction, preconstruction, and early-work requirements as a coordinated cross-rule framework.** The final rule should apply a common functional construction principle across the relevant licensing pathways, conform dependent rules and guidance, including Part 51, and identify any pathway-specific differences that remain necessary because of statute or a distinct safety or licensing basis.
2. **The NRC should establish a pathway-neutral framework for recognizing prior NRC findings and preserving their substantive and procedural finality across Parts 50, 52, 53, and 57.** Applicants should be able to rely on prior design, manufacturing, codes and standards, QA/QMS, analytical, and programmatic findings when they demonstrate technical applicability and satisfy requirements unique to the receiving framework. A change in licensing pathway should not itself require renewed technical review or reopen a matter in adjudication; additional review should focus on material differences, new information, unresolved matters, and requirements unique to the receiving framework.
3. **The NRC should coordinate reactor licensing with the downstream storage and transportation requirements applicable to the same fuel and reactor designs.** The NRC should identify which reactor-side findings may be relied on under Parts 71 and 72, which findings must be made independently, and how authorization, possession, and responsibility transition when irradiated fuel or transportable reactor modules leave the deployment site.

XII. IMPLEMENTATION AND GUIDANCE

Implementation will be one of the central challenges of the final rule. The proposed rule does not revise a single licensing process or technical standard; it brings together changes to construction, design finality and departures, codes and standards, quality assurance, analytical methods, fuel, licensing duration, emergency preparedness, siting, and other parts of the reactor regulatory framework. The NRC is also developing approximately two dozen guidance documents to support

these changes. The effectiveness of the final rule will depend on whether these provisions and guidance documents operate as a coherent system rather than as separate reforms applied independently by different review teams.

That scale creates a particular need for clear implementation mechanics. Applicants and licensees will need to understand which provisions apply automatically, which are optional, which require an affirmative licensing action, and which flexibilities may be used independently or only as part of a broader framework. NRC staff will likewise need consistent direction on how the new provisions interact with existing licensing bases, prior approvals, exemptions, license conditions, and guidance. Without that coordination, implementation could recreate through guidance or review practice the uncertainty and case-by-case burden that the rule is intended to reduce.

The NRC should accordingly treat the guidance package as an integrated implementation framework. The final rule and accompanying guidance should clearly distinguish regulatory requirements from acceptable methods, identify dependencies among the new options, and provide a common implementation roadmap showing the applicable regulatory vehicle, prerequisites, and point at which each option becomes controlling. The NRC should also plan to update the guidance as experience with the first applications and licensing actions reveals where clarification or adjustment is needed.

A. The NRC Should Provide a Clear Implementation Pathway for the New Regulatory Options

The final rule and implementing guidance should provide an applicant-facing map of the principal licensing and approval pathways created or modified by the rule. For construction and repeat deployment, that map should identify how design approval, generic finality, reference-plant verification, environmental prerequisites, application docketing, general-license eligibility, NRC inspections, and subsequent construction-permit or combined-license review fit together. Similar clarity is warranted wherever the rule creates a voluntary alternative that depends on a prior NRC determination or a particular stage of licensing.

Implementation guidance should also address cases that do not fit neatly within a single-unit licensing sequence. Shared and dual-function SSCs can serve multiple units or modules that are

at different stages of licensing, construction, or operation. The applicable authorization should not depend on ad hoc project-specific interpretations of the same recurring configuration. The NRC should establish or endorse reusable classification methods for shared, dual-function, and multi-module SSCs and explain which authorization governs work affecting more than one module or licensing status.

Inspections and verification milestones should likewise be used according to the uncertainty that remains. The general construction license should not acquire routine hold points that function as additional prior approvals for work the rule otherwise authorizes. Targeted inspection, vendor oversight, audits, or verification milestones may nevertheless be appropriate where they address a defined safety issue, establish an eligibility criterion, or allow an applicant voluntarily to reduce regulatory uncertainty before proceeding at risk. The same principle applies to graded QA. Additional oversight can compensate for unresolved FOAK, supplier, or process uncertainty without becoming a permanent requirement after the relevant basis has been demonstrated.

B. Transition Provisions Should Make the New Flexibilities Usable by Existing and Pending Projects

The NRC should explain how existing licensees and applications already under review may elect to use voluntary pathways established by the final rule. A new option provides limited practical benefit if its availability depends on beginning an entirely new licensing process or maintaining the legacy framework until a later project. Where an applicant or licensee can satisfy the substantive eligibility criteria for a new pathway, the transition mechanism should allow use of that pathway without requiring unnecessary duplication of findings already made in the pending or existing proceeding.

Appendix T illustrates the issue. If an applicant must first establish its QMS, supplier qualifications, procurement controls, and manufacturing processes under Appendix B and can transition only after the initial plant has been completed and operated, much of the supply-chain structure that Appendix T is intended to make more flexible will already be established. The final rule should identify how existing licensees and pending applicants may transition to Appendix T when they can demonstrate the required design, QMS, supplier, and implementation maturity, subject to additional oversight where material uncertainty remains.

License renewal presents a different transition question. A renewed license may be issued before the period of extended operation begins, while inspections, aging-management enhancements, or other obligations supporting that renewal may need to be completed while the existing license remains in effect. The final rule or implementation guidance should identify an enforceable mechanism for those pre-effective-period obligations so that the timing of the renewed license does not create uncertainty about when the requirement applies or how the NRC oversees compliance.

Operation beyond 80 years also requires a programmatic implementation strategy. Applicants should remain responsible for plant-specific operating experience, aging-management demonstrations, and applicability of the generic technical basis to their facilities. But common questions concerning long-duration materials behavior, degradation mechanisms, environmental effects, and aging of passive long-lived SSCs are better addressed through generic NRC research and guidance where practicable. The NRC should identify common research gaps and update its generic license-renewal guidance as the technical record develops rather than requiring each subsequent applicant to independently recreate the same generic technical basis.

C. Implementing Guidance Should Not Recreate the Requirements the Rule Is Intended to Replace

The NRC describes the accompanying regulatory guides as voluntary means of implementing the rule. That distinction should remain meaningful in practice. A performance-based regulation can become effectively prescriptive if the guidance provides one highly developed acceptable method while applicants using another technically sound approach face uncertain acceptance criteria, open-ended review, or substantially greater regulatory risk.

The implementation package should distinguish clearly between the regulatory finding the NRC must make and the particular method described in a regulatory guide for supporting that finding. Where an alternative approach is permitted, the NRC should state the performance, technical-adequacy, or safety-assurance criteria that the alternative must satisfy. A departure from a regulatory guide should trigger review of whether those criteria remain satisfied, not an assumption that departure from the guide is itself a regulatory deficiency.

This principle applies across several provisions in the proposed rule. Alternative QA frameworks require sufficiently clear acceptance criteria to prevent the established Appendix B framework from becoming the only predictable path. PRA guidance should evaluate whether a model is technically adequate for the regulatory decision rather than make prior application-specific approval the only practical gateway to a risk-informed pathway. Guidance implementing quantitative event thresholds should define the circumstances under which qualitative considerations justify a different treatment so that bounded judgment does not become an independent, open-ended classification standard. An approved VVUQ program should likewise provide meaningful flexibility within its approved scope rather than lead to repeated review of the same technical credibility basis.

The NRC should consider publishing a consolidated implementation crosswalk for the guidance package identifying the regulatory provision being implemented, the finding or criterion that must be satisfied, the relevant regulatory guide, the status of that guide as a voluntary acceptable method, and the means by which an applicant may propose an alternative. Such a crosswalk would help applicants and staff distinguish binding requirements from implementation examples and would reduce the risk that differences among the guides produce inconsistent review expectations.

D. The NRC Should Establish a Standard Mechanism for Reusing Prior Determinations

The final rule creates several opportunities for regulatory learning and reuse, but implementation guidance should explain how a later applicant invokes a prior NRC determination. The mechanism should be straightforward: the later applicant identifies the prior approval or finding, demonstrates that its technical basis, scope, conditions, and limitations remain applicable, and identifies any material differences. NRC review should then focus on those differences and on requirements unique to the later application rather than repeat the full underlying review.

This mechanism should apply not only to design certifications or other formal licensing approvals but also, where appropriate, to programmatic and vendor-level technical determinations. An approved QA framework, code or standards alternative, VVUQ program, analytical method, or similar technical basis may be developed for use by more than one facility.

Where the NRC has already evaluated that basis for a defined scope of application, subsequent applicants should be able to reference it and demonstrate applicability rather than recreate the same technical showing in each docket.

Reuse should not eliminate applicant responsibility or NRC authority to address changed circumstances. A new review remains appropriate where the later application falls outside the approved scope, relies on materially different assumptions or service conditions, introduces a significant change, or presents information that undermines the prior technical basis. The implementation process should make that boundary explicit so that applicants can distinguish between a required applicability demonstration and a reopening of the underlying determination.

E. The NRC Should Review Early Implementation and Update the Guidance Package

The NRC should plan to review the rule after sufficient early implementation experience has accumulated. This is particularly important because the rule combines numerous new and revised regulatory pathways with a supporting package of 22 draft regulatory guides, one interim staff guidance document, and one NUREG update. Many of the most important implementation questions will become visible only when NRC staff, applicants, licensees, and vendors begin applying these provisions together in actual licensing and oversight activities.

The review should focus on whether the new framework is producing the regulatory outcomes the NRC intended. Relevant questions include whether voluntary alternatives are practically usable; whether different review teams are applying common eligibility, acceptance, and change control criteria consistently; whether guidance is creating unnecessary prior-approval requirements or other burdens not required by the rule; and whether dependencies among the different regulatory options are creating new critical paths. The NRC should pay particular attention to areas where the final rule relies on new or substantially revised implementation approaches, including Appendix T, risk-informed § 50.59 evaluations, VVUQ programs, generic finality and repeat deployment, and the new siting framework.

BTI recommended a similar approach in its comments on other NRC rulemakings.⁸⁸ The NRC should establish a defined early implementation review after experience has been gained from the first applications, licensing actions, and inspections conducted under the new framework. Where that review identifies inconsistent staff interpretations, unnecessarily prescriptive guidance, duplicative review, or regulatory mechanisms that are not producing the intended flexibility, the NRC should revise the affected guidance or internal procedures promptly. More fundamental problems can be addressed through targeted rule amendments where necessary. The success of this reform should ultimately be measured by how coherently and predictably the framework operates in practice, not solely by whether each individual provision appears workable when issued.

Recommendations

1. **The NRC should provide clear implementation and transition pathways that make the final rule's new regulatory options usable by both new and existing projects.** Guidance should identify the applicable regulatory vehicle, prerequisites, and transition mechanism for each major option and allow existing licensees and pending applicants to use voluntary new pathways when they satisfy the applicable criteria without unnecessarily repeating prior findings. The NRC should also address recurring implementation questions such as shared or multi-module SSCs and obligations that arise before a renewed license becomes effective.
2. **Implementation guidance should preserve the flexibility created by the final rule rather than recreate prescriptive requirements through review practice.** The NRC should distinguish regulatory requirements from voluntary acceptable methods, state clear acceptance criteria for alternative approaches, and use inspections, hold points, and additional oversight in proportion to unresolved uncertainty rather than as routine substitutes for prior approval.
3. **The NRC should establish a standard process for reusing prior technical and programmatic determinations.** A later applicant should be able to identify a prior NRC finding, demonstrate that its technical basis, scope, conditions, and limitations remain applicable, and focus subsequent review on material differences and requirements unique to the later proceeding rather than repeating issues the NRC has already resolved.

⁸⁸ As cited throughout this comment.

4. **The NRC should conduct an early implementation review and update the guidance package based on experience with the first applications, licensing actions, and inspections.** That review should identify inconsistent staff interpretations, unnecessarily prescriptive guidance, duplicative review, or dependencies among new regulatory options that create unintended barriers, with guidance and internal procedures revised as needed.

XIII. CONCLUSION

BTI supports the NRC's effort to modernize reactor licensing, safety oversight, construction, and siting. The proposed rule contains many reforms that should be finalized because they move the regulatory framework toward performance-based, risk-informed, and technology-inclusive decisionmaking. The final rule should preserve that direction by tying regulatory requirements and prior-approval obligations to safety and regulatory significance rather than historical categorization, licensing pathway, or conservatism for its own sake. The NRC can maintain adequate protection of public health and safety and the common defense and security while avoiding requirements that unnecessarily constrain the deployment and societal benefits of civilian nuclear energy.

The NRC should preserve flexibility once it has been created. Where the NRC has already resolved a technical or programmatic issue, later applicants should be able to rely on that determination when its technical basis, assumptions, conditions, and limitations remain applicable. Applicants remain responsible for demonstrating applicability, and the NRC retains authority to review material differences, changed circumstances, and genuinely new issues. But repeated review of an issue already resolved does not provide additional safety simply because the determination appears in a new application, at a different site, or under a different licensing pathway. A modern regulatory system should learn from prior review and make that regulatory learning reusable.

The same systems perspective should guide coordination across NRC rules. Construction, generic finality, quality assurance, analytical methods, fuel qualification, emergency preparedness, siting, adjudicatory procedures, environmental review, transportation, and storage do not operate in isolation. Changes made in one licensing pathway can alter the meaning, availability, or practical value of pathways established elsewhere. The NRC should coordinate related rulemakings and

implement guidance so that differences among Parts reflect genuine statutory, technical, or safety distinctions rather than the timing or sequence in which individual rules were developed.

Implementation will ultimately determine whether this rule achieves its purpose. Regulatory guides should preserve, rather than narrow, the flexibility established in the regulations; voluntary acceptable methods should not become de facto requirements; and applicants should have clear pathways for using new options, transitioning from existing frameworks, and referencing prior NRC determinations. The success of this modernization effort should be measured by whether the resulting framework operates coherently, predictably, and proportionately in practice. BTI encourages the NRC to finalize the reforms that advance those objectives and to revise the remaining provisions and guidance consistent with the recommendations in this comment.

BTI appreciates the opportunity to submit these comments and would welcome further engagement with the staff on the issues raised in this letter.

Sincerely,

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