Breakthrough Institute Comment on Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles [Docket ID: NRC-2025-0379 | RIN: 3150-AL36]

June 15, 2026

The Nuclear Regulatory Commission (NRC) is developing a new licensing framework for commercial nuclear reactors, as mandated by the Accelerating Deployment of Versatile, Advanced Nuclear for Clean Energy Act and Executive Order 14300, "Ordering the Reform of the Nuclear Regulatory Commission". It published a draft rule in the Federal Register Notice 91 FR 23628 in May 2026 (NRC-2025-0379, RIN 3150-AL36, "Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles").

This letter and its enclosure[1] provide the perspective of the Breakthrough Institute on the ongoing activities by the NRC to "establish a risk-informed and performance-based regulatory framework for rapid licensing of new microreactors and other reactors with comparable risk profiles and for high-volume deployment of these reactors", known as Part 57, under Title 10 of the Code of Federal Regulations. This correspondence is intended to engage with the NRC as a non-profit and independent stakeholder.

The Breakthrough Institute (BTI) appreciates the opportunity to comment on the Nuclear Regulatory Commission's proposed rule, "Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles" (Docket ID NRC-2025-0379, RIN 3150-AL36).[2] BTI is an independent 501(c)(3) research organization 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.

The timely completion of a risk-informed, performance-based framework to enable rapid deployment of reactors is important to the successful innovation and commercialization of advanced nuclear reactors in the United States. The effort by the NRC staff to write this draft

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[1] See Appendix, "Addressing Specific Requests for Comment."

[2] U.S. Nuclear Regulatory Commission, *Licensing Requirements for Microreactors and Other Reactors With Comparable Risk Profiles*, NRC-2025-0379, 91 Fed. Reg. 23628, https://www.federalregister.gov/documents/2026/05/01/2026-08550/Licensing-requirements-for-microreactors-and-other-reactors-with-comparable-risk-profiles.

regulation on the current timeline is to be commended. Adjustments are necessary to the draft rule to provide a licensing framework to meet this goal.

I. Introduction

The NRC developed Part 57 in response to Section 208 of the ADVANCE Act,[3] which directed the Commission to develop risk-informed and performance-based strategies to license and regulate microreactors, and in response to Section 5(e) of EO 14300,[4] which directed the NRC to establish a process for high-volume licensing of microreactors and modular reactors, including through standardized applications and approvals. The NRC considered and declined to pursue amendments to Parts 50 or 52, and considered and declined to develop Part 57's scope within the Part 53 technology-inclusive framework on the grounds that Part 53 was designed to cover large, complex reactors as well as microreactors. The agency instead developed Part 57 as a new framework that combines innovative licensing tools with elements of the Non-Power Production or Utilization Facility (NPUF) licensing approach in Part 50 and elements from Parts 52 and 53 to create a focused, rapid, and high-volume deployment licensing pathway.

BTI has engaged in numerous major NRC rulemakings, including submitting comments on Part 53, participating in NRC rulemaking public meetings, and working directly with NRC staff and nuclear stakeholders. BTI has a longstanding interest in the development of licensing frameworks that are proportionate to risk, administrable at scale, and capable of supporting the deployment of advanced nuclear technologies.

BTI's overall assessment of the proposed rule is mixed. Part 57 asks the right institutional question: whether the NRC can distinguish among reactors that pose fundamentally different levels of risk and regulate them accordingly. The proposed rule's use of consequence-based eligibility criteria, its emphasis on standardized designs and generic finality, and its manufacturing license and general construction authority provisions reflect a serious attempt to

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[3] ADVANCE Act of 2024, Pub. L. No. 118-67, § 208.

[4] 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.

build a licensing architecture suited to high-volume deployment. Those features deserve to be preserved and strengthened in the final rule.

At the same time, several aspects of the proposed rule require clarification, revision, or more explicit justification. The rule's purpose is unclear about whether Part 57 is meant to define a reactor category, screen reactors by consequence, or create a high-volume licensing pathway for standardized deployment. That ambiguity propagates into the eligibility criteria, Part 57's relationship with Part 53, and the rule's licensing tools. The 10 metric ton fuel-mass limit introduces an independent eligibility bar that is not clearly grounded in the same consequence-based logic that justifies the rule's other provisions. The operational provisions for remote, autonomous, and operator-independent facilities require more explicit institutional grounding to be durable. Several environmental review and siting provisions raise questions about how obligations under the National Environmental Policy Act (NEPA) will be met for non-traditional deployments.

BTI's principal recommendations are as follows.

Table 1. Summary of Principal Recommendations

#	Recommendation
1	**Clarify Part 57's purpose as a high-volume licensing framework.** The rule should not rely solely on the phrase "microreactors and other reactors with comparable risk profiles". The NRC should instead make clear that Part 57's purpose is to enable the rapid and high-volume deployment of nuclear reactors. Consequence-based eligibility criteria serve to enable the high-volume licensing purpose. The proposed rule title suggests the inverse.

#	Recommendation
2	**Incorporate broadly useful Part 57 tools into Part 53.** Most of the novel flexibilities in Part 57—graded quality assurance, flexible codes and standards, manufacturing-license concepts, generic finality, remote and autonomous operation provisions, and standardized operational programs—are risk-informed regulatory tools that Part 53 applicants should also be able to access where justified. The NRC should revise Part 53 to make these provisions available and establish an explicit pathway allowing a design licensed under Part 53 to access Part 57's nth-of-a-kind (NOAK) deployment tools once the design and operational programs are sufficiently standardized, including by revising § 57.18(b) to allow incorporation by reference of prior Part 53 approvals.
3	**Preserve the 1 rem TEDE accident consequence criterion as the primary eligibility gate, while treating maximum hypothetical accident (MHA) and maximum credible accident (MCA) methodologies as consequence-based methods rather than frequency-based risk screens.** Applicants should not be required to add probabilistic risk analysis (PRA) merely because the eligibility framework evaluates bounded consequences rather than event frequency. The NRC should also clarify how the selected methodology affects site boundaries, source-term development, safety-related structures, systems, and components (SSC) classification, emergency planning, security, and environmental review.
4	**Replace or supplement the proposed 10 metric ton fuel-mass limit with performance-based entry criteria tied directly to safety.** Fuel mass is only a proxy for decay heat, source term, accident progression, fission product retention, and emergency planning needs. The NRC should justify why heavy-metal mass is an appropriate screen for simplified, repeatable review. The NRC should also clarify whether the threshold is intended as a safety criterion, an administrative boundary, or both, and specify whether it applies on a per-reactor, per-module, per-plant, or per-site basis.

#	Recommendation
5	**Add a physical-security entry criterion that bounds the complexity of the security case.** Part 57 caps the complexity of the safety case through the § 57.25(a) consequence criterion but imposes no equivalent bound on the security case, allowing an applicant who satisfies § 57.25 to rely on an open-ended, site-specific operational security program (including a credited armed interdiction-and-neutralization response) that cannot be standardized or reviewed on a compressed timeline. The NRC should add an entry criterion, parallel to the § 57.25(a) dose criterion, requiring that the applicable physical-security reference values be met through inherent design characteristics, security-by-design features, and standardized programmatic delay-and-denial measures, without crediting a site-specific armed response whose adequacy depends on case-by-case adversary-pathway and response-timeline analysis. Those reference values should be drawn from the technology-inclusive requirements of § 73.100 rather than § 50.34(a)(1)(ii)(D), aligning Part 57 with Part 53.
6	**Make manufacturing licenses a true deployment tool for factory-built reactors.** For manufactured or transportable reactors, the manufacturing license should be capable of resolving the reactor design, production controls, quality assurance, configuration management, factory testing, standardized interfaces, and appropriate operational-program elements with meaningful finality. The NRC should also explain how changes approved at the manufacturing-license level can be propagated to referencing operating licenses without duplicative amendments.
7	**Ensure consistent emergency planning treatment for all nuclear technologies.** Part 57 forgoes the requirement for an emergency planning zone (EPZ) but requires an emergency plan for covered facilities because of the 1-rem site-boundary consequence criterion. This is intended to support right-sized emergency planning. However, other frameworks, such as the proposed fusion machine licensing framework, allow conditional emergency planning under the same 1-rem threshold. Differing licensing pathways should not justify differing standards. That difference appears to reflect a policy distinction rather than a consequence-based safety distinction. The NRC needs to resolve the inconsistency in emergency plan requirements across rulemakings.

#	Recommendation
8	**Ensure environmental review is consequence-calibrated and scalable to support high-volume deployment.** The proposed categorical exclusion framework contains an internal legal conflict that makes it practically inaccessible, and several eligibility criteria impose deterministic bars untethered to actual environmental impacts. The NRC should resolve the conflict on categorical exclusion eligibility, replace categorical bars with impact-based screening, and allow applicants to use existing and representative site characterization data. Environmental review requirements should be calibrated to the same consequence-based showing that justifies Part 57's streamlined safety review.
9	**Establish implementation guidance and commit to revisiting Part 57 after early applications and deployments.** The NRC should publish guidance explaining how to select among Part 57, Part 53, and existing pathways; how prior work under Parts 50, 52, and 53 can be carried into a Part 57 application without duplication; and how pre-application engagement can be used to build confidence before committing to a full application. After initial implementation, the NRC should evaluate whether the rule is achieving its central purpose: predictable, risk-informed, high-volume licensing of reactors that satisfy consequence-based eligibility criteria.
10	**Coordinate Part 57 with adjacent rulemakings and rule parts.** Part 57 will interact with Part 2 adjudication, Part 51 environmental review, Part 26 fitness-for-duty requirements, Part 70 special nuclear material licensing, Part 71 transportation package certification, Part 72 storage, Parts 73, 74, and 75 security and safeguards requirements, Part 140 financial protection, and the implementation of Part 53. The NRC should identify these interfaces in the final rule or implementation guidance and explain how applicants should sequence related approvals.

With changes, Part 57 can become a durable and effective licensing framework because it is more focused, not because it is less rigorous. The NRC should finalize a rule that preserves adequate protection, gives applicants predictable pathways for standardized deployment, and keeps regulatory effort proportional to the hazards, uncertainties, and deployment configurations presented by the facility under review.

II. Part 57's Purpose

As stated, the NRC is proposing Part 57 "to establish a risk-informed and performance-based regulatory framework for rapid licensing of new microreactors and other reactors with comparable risk profiles and for high-volume deployment of these reactors."[5] The rule would provide flexible licensing pathways, reduce regulatory burden, and keep safety and security requirements commensurate with the hazards posed by covered facilities. BTI supports the agency's effort to create a high-volume deployment and right-sized licensing pathway. The proposed rule's direction is sound, and the entry criteria, the emphasis on standardized designs, and the mechanisms for generic finality and manufacturing licenses are appropriate tools for the problem the agency is trying to solve.

The proposed rule's purpose statement, however, creates a structural ambiguity that could generate avoidable boundary disputes, complicate implementation, and undermine the rule's long-term durability. The rule's title and summary invoke two overlapping concepts, "microreactors" and "other reactors with comparable risk profiles", without defining the first, explaining the comparison baseline for the second, or clearly stating whether Part 57 is primarily a technology-class rule, a consequence-based eligibility rule, or a high-volume deployment architecture. This ambiguity has already created confusion for the Advisory Committee on Reactor Safeguards (ACRS).[6] NRC staff clarified at the May 20, 2026, public meeting that the agency did not intend to define "microreactor" and instead intended Part 57 to be available to any reactor that satisfies the entry criteria.[7] That clarification is useful context, but it should be reflected more clearly in the rule's preamble and implementation guidance that will bind future staff or withstand administrative review.

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[5] U.S. Nuclear Regulatory Commission, *Licensing Requirements for Microreactors and Other Reactors With Comparable Risk Profiles*, NRC-2025-0379, 91 Fed. Reg. 23628, Paragraph 3, https://www.federalregister.gov/d/2026-08550/p-3.

[6] Advisory Committee on Reactor Safeguards, *Proposed Rulemaking on Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles (Part 57)*, May 19, 2026, https://www.nrc.gov/docs/ML2613/ML26133A238.pdf.

[7] Nuclear Regulatory Commission, Public Meeting Schedule: Meeting Details—Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles (Meeting Code 20260503), May 20, 2026, 1:00—3:00 p.m. ET, https:"/www.nrc.gov/public-involve/public-meetings/pmns/20260503.

BTI's view is that Part 57 is best understood, and best justified, as a high-volume licensing pathway. Taken any other direction, Part 57 is not justified as a separate rule; its broadly useful provisions should instead be incorporated into Part 53 or as guidance in Part 53.

1. Principal Analysis and Recommendation: Part 57 Should be a Rule for High-Volume Licensing.

The consequence-based entry criteria in §§ 57.25 and 57.30 do the work of defining the eligible class of reactors to enable the high-volume licensing purpose. The 1 rem TEDE dose-based criterion and the design criteria attributes in § 57.30 together establish, in operational terms, what reactors the rule is designed to serve without narrowing the technology class. By contrast, the phrase "microreactors and other reactors with comparable risk profiles" in the purpose statement adds a technology-class overlay that the rule does not enforce and that could create unnecessary ambiguity about eligibility, finality, and the rule's relationship to Part 53.

That ambiguity is not merely semantic. If future staff, applicants, or reviewing courts treat the "microreactor" language as a substantive eligibility boundary, designs that satisfy the consequence-based criteria could face additional scrutiny or exclusion that the rule's own provisions do not support. The preamble's explanation that Part 57 combines elements of the NPUF licensing approach in Part 50 with elements from Parts 52 and 53 to create a framework "focused on rapid and high-volume licensing" reflects the correct underlying rationale. That rationale should be the rule's stated purpose, not a secondary description.

Framing Part 57 as a high-volume licensing pathway also better advances the objectives reflected in ADVANCE Act Section 208 and EO 14300, both of which call for risk-informed and performance-based licensing strategies. The consequence-based eligibility structure in §§ 57.25 and 57.30 is precisely the kind of risk-informed and performance-based approach those directives call for. Although the NRC had not completed Part 53 when EO 14300 was issued, that timing does not mean Part 57 should be the only licensing pathway for microreactors; Part 53's technology-inclusive pathway for advanced reactors is just as viable for microreactors. What is novel in Part 57 is the high-volume licensing architecture: it covers microreactors, but more importantly, it creates a specialized framework for standardized, repeatable, high-volume deployment of reactors that satisfy consequence-based eligibility criteria.

The NRC should revise the purpose statement and preamble to frame Part 57 as a high-volume licensing pathway for reactors that satisfy consequence-based eligibility criteria. The preamble should state explicitly that §§ 57.25 and 57.30 are the operative eligibility provisions and that a reactor satisfying those criteria is eligible for Part 57 irrespective of its conventional technology classification or thermal power level.

2. Contingent Analysis and Recommendations

BTI's primary recommendation is that Part 57 should be framed as a high-volume licensing rule. If the NRC instead retains a different theory of the rule—either as a low-consequence reactor rule or as a microreactor rule—the NRC should make additional revisions to avoid ambiguity and preserve administrability.

If the NRC determines that the phrase "microreactors and other reactors with comparable risk profiles" should be retained as substantive framing rather than descriptive context, the rule as proposed should not remain a standalone rule and should be enveloped into Part 53 as a high-volume pathway in guidance. Such guidance should include a definition of "microreactor". The definition need not be prescriptive in terms of power level or technology type, but it should be sufficiently clear to give applicants and staff a workable reference. The NRC should also consider whether the § 50.2 definition of "small modular reactor"[8] should be incorporated into the scope, given that EO 14300 § 5(e) directs the NRC to establish a high-volume licensing process for both microreactors and modular reactors. Applicants are already familiar with the Part 50 definition, and using or cross-referencing it would reduce boundary uncertainty while preserving the separate consequence-based eligibility criteria. The NRC should also define "other reactors with comparable risk profiles." It should identify what risk attributes determine comparability—whether the comparison is to microreactors generally, to the entry criteria in §§ 57.25 and 57.30, or to some other reference. Without that explanation, "comparable risk profile" functions as an undefined qualifier that does not provide the regulatory certainty the rule is designed to deliver.

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[8] 10 C.F.R. § 50.2 defines "small modular reactor" as a power reactor licensed to produce heat energy up to 1,000 megawatts thermal per module.

If Part 57 is intended to be a microreactor rule only, as the ACRS suggests,[9] again, the rule should not remain standalone and should be enveloped into Part 53 as guidance as a high-volume pathway. The NRC should define "microreactor" and justify excluding reactors that may be larger but present comparable or lower consequences and equally reviewable deployment cases. A microreactor-only rule would be more deterministic and less technology-inclusive, and it could exclude designs that satisfy the same safety and high-volume licensing objectives. The NRC should still consider whether the § 50.2 definition of "small modular reactor" should be incorporated or referenced. Any alternative size or power threshold would require independent justification.

As we have emphasized, our view is that these regulatory theories are incorrect and do not create regulatory value for the proposed rule. Part 57 should only exist as a separate rule as a high-volume licensing framework. If it is not, Part 57 should not be a separate rule; it should be a pathway through the technology-inclusive risk-informed Part 53 rule.

Table 2. Recommendations – Part 57's Purpose

Affected Provision	Recommendation
Preamble §§ II.A—II.B; Summary	Revise the purpose statement to frame Part 57 as a high-volume licensing pathway for reactors satisfying consequence-based eligibility criteria; identify §§ 57.25 and 57.30 as the operative eligibility screen.
Preamble §§ II.A—II.B	Explain in the preamble why Part 57 is structured as a separate regulatory part, grounding the explanation in the high-volume deployment tools Part 57 provides.

[9] Advisory Committee on Reactor Safeguards, "Proposed Rulemaking on Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles (Part 57)," May 19, 2026, ML26133A238.

Affected Provision	Recommendation
§ 57.0 Purpose.	Add an explicit purpose section in Part 57 text *(analog to § 53.000)* to state high-volume licensing as its purpose for reactors licensed under section 103 of the Atomic Energy Act that satisfy Part 57's eligibility criteria. It should also state that Part 57 is intended to support efficient and rapid deployment to maximize the benefits of standardization while maintaining reasonable assurance of adequate protection of public health, safety, and security.
Contingent Recommendation	
Preamble §§ II.A—II.B; Summary; relationship to Part 53	If the NRC retains a microreactor or "comparable risk profile" theory as substantive framing rather than as a high-volume licensing pathway, Part 57 should not remain a standalone rule; its broadly useful provisions should instead be incorporated into Part 53 as a high-volume licensing pathway and associated guidance.
§ 57.3; Preamble §§ II.A—II.B	If the "microreactors and other reactors with comparable risk profiles" framing is retained, define "microreactor" with a workable reference that need not be power- or technology-prescriptive, define "other reactors with comparable risk profiles" by identifying the comparability baseline (whether to microreactors generally or to the §§ 57.25 and 57.30 criteria), and consider cross-referencing the § 50.2 small modular reactor definition consistent with EO 14300 § 5(e).
§ 57.3; Preamble §§ II.A—II.B	If Part 57 is retained as a microreactor-only rule, define "microreactor," justify excluding larger reactors that present comparable or lower consequences and equally reviewable deployment cases, and independently justify any alternative size or power threshold; consider whether the § 50.2 small modular reactor definition should be incorporated or referenced.

III. Part 57 and Part 53

The NRC should clarify the relationship between Part 57 and Part 53. Many of the flexibilities proposed in Part 57 are not inherently unique to reactors that satisfy the entry criteria in Part 57, but *are* risk-informed and performance-based regulatory approaches that inherently belong in

Part 53. This includes right-sized operational programs, flexible quality assurance, consequence-based security, flexible codes and standards, remote and autonomous operation provisions, among other provisions. These tools also reflect recommendations stakeholders have made for Part 53 to be a truly risk-informed, performance-based advanced reactor framework.[10,11,12]

The only justification for a separate Part 57 licensing pathway is that it is for high-volume deployment. Part 57 is most defensible as a specialized framework for standardized, repeatable licensing of reactors whose safety, security, environmental, operational, and lifecycle issues are sufficiently bounded to support accelerated review, generic finality, manufacturing licenses, general construction authority, standardized operational programs, and NOAK deployment. The proposed Part 57 rule contains many novel licensing tools that enable high-volume deployment, such as: standardized design approvals with generic finality under § 57.142(e), manufacturing licenses under Subpart D, general construction authority under § 57.45(d), and the ability to combine in a single joint application a construction permit, operating licenses for multiple units, a manufacturing license, and approvals for special nuclear material and fuel storage. Together, those tools create a licensing architecture for repeat deployment that is materially different from what Part 53 currently provides.

The final rule should therefore make clear that Part 57 is separate in regulatory value as a high-volume licensing pathway while complementary to Part 53. The NRC should also clarify how applicants may navigate between Part 53 and Part 57, including how a design first licensed or approved under Part 53 may later use Part 57 tools for NOAK deployment once a design and its operational programs are sufficiently standardized.

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[10] 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.

[11] The Breakthrough Institute et al., *Joint NGO Comments on NRC's Rulemaking on Part 53, Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors*, NRC-2019-0062, RIN 3150-AL31, February 28, 2025, ML25069A162.

[12] ClearPath, *Comment on Proposed 10 CFR Part 53: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors,* NRC-2019-0062, February 28, 2025, ML25066A027.

1. Transferability from Part 53 to Part 57

Part 53 should serve as the primary risk-informed framework for first-of-a-kind (FOAK) reactors that may require iteration during licensing, construction, testing, or initial operation. Part 57 should serve as the deployment framework for reactors that are sufficiently complete, standardized, and low-consequence to support high-volume licensing.

The NRC should establish a clear transferability mechanism between Part 53 and Part 57. That mechanism must work on two levels: first, through alignment in licensing tools, analytical approaches, and review expectations between the two rules so that a developer does not have to rebuild its licensing basis when moving from one framework to the other; and second, through explicit guidance on how applicants may move from Part 53 to Part 57 once the design and operational programs are sufficiently mature. Without both, a developer may license a FOAK reactor under Part 53, resolve major design and safety issues through NRC review, and then have to re-review the same issues when seeking Part 57 approval for NOAK deployment. That outcome would undermine the purpose of both frameworks.

For that pathway to work, Parts 53 and 57 must be aligned in both regulatory construct and guidance. A core example of compatible analytical approaches that can resolve such a dynamic is allowing Part 53 to use maximum hypothetical accident (MHA) and maximum credible accident (MCA) methodologies. Part 57 uses these methodologies as performance-based pathways to demonstrate compliance with the 1 rem TEDE dose-based entry criterion at the site boundary. Part 53 should recognize MHA and MCA as acceptable consequence-based methods for demonstrating compliance with its own dose-based safety criteria, so that a developer who characterizes its design's accident consequences under Part 53 using those methodologies does not have to reconstruct its accident analysis from different analytical foundations when seeking Part 57 eligibility. The methodology, the source term, the site boundary assumptions, and the SSC classification logic that flow from an MHA or MCA analysis should be portable from one framework to the other. In a similar fashion, if a design is expected to move from FOAK review to NOAK deployment, the applicant should also be able to establish compatible design, manufacturing, operational, and licensing-basis approaches during Part 53 proceedings as a function of risk-informed, performance-based, and analytically-aligned regulation.

Guidance should additionally explain the pathway of how an applicant that licenses a reactor under Part 53 can later seek Part 57 approval for standardized deployment. Proposed § 57.18(b) allows a Part 57 applicant to incorporate by reference information from previous applications and approvals issued under Parts 50 and 52, but not under Part 53. That asymmetry has no apparent technical justification. An applicant that has received a standard design approval or operating license under Part 53 should be able to reference that approval in a Part 57 application to the same extent as an applicant referencing a Part 50 or Part 52 approval. The final rule should revise § 57.18(b) to include Part 53 approvals and licenses within the scope of referenceable prior actions, or should explain why Part 53 is excluded and whether that exclusion is intended to be permanent.

Importantly, the proposed rule does not address how an applicant that begins a FOAK licensing process under Part 53, Part 50, or Part 52 may subsequently access Part 57's NOAK tools once the design, operational programs, and deployment assumptions are sufficiently standardized. If those tools are available only to applicants who initiate licensing under Part 57 from the outset, their practical value is limited to new entrants with no prior NRC engagement. If they are accessible to applicants transitioning from other parts—which the rule's high-volume deployment rationale would support—the rule should identify the mechanism and the conditions under which a prior Part 50, Part 52, or Part 53 licensing action can serve as the basis for NOAK finality under Part 57.

The transfer pathway should also preserve the value of prior NRC review. If the NRC has resolved a design feature, safety analysis, SSC classification, manufacturing method, operational program, or other licensing-basis matter in a Part 53 proceeding, the same matter should not be re-reviewed from the beginning in a later Part 57 proceeding. The NRC should focus its Part 57 review on what is new, changed, unresolved, or site-specific.

Beyond transferability, such risk-informed tools should naturally reside in Part 53, *a risk-informed and performance-based rule*. Many Part 57 provisions are not inherently high-volume-only provisions; they are tools for scaling regulatory treatment to the safety significance and consequence profile of the reactor. Those tools should be available in Part 53 where justified by the applicant and approved by the NRC. They should not be automatic entitlements, but they should be available options within a risk-informed framework.

This approach would allow Part 53 to function as the first-of-a-kind development framework and Part 57 to function as the high-volume deployment framework. It would also reduce regulatory uncertainty by giving developers a predictable path from prototype licensing to standardized deployment.

2. What Part 53 Can Adopt from Part 57

In short, almost all generally useful Part 57 tools should be available in Part 53. These tools should be scaled to the specific application, justified by the applicant, and reviewed in the context of the reactor's design, safety case, and deployment model. What should remain Part 57-specific are the provisions uniquely enabled by high-volume licensing and Part 57's entry criteria, such as the six-month review model, the 1-rem eligibility gate, and emergency-planning simplifications tied to that gate. Such features in Part 57 are only possible because applicants must first satisfy Part 57's entry criteria. Those provisions should remain in Part 57 and not be adopted in Part 53, while essentially everything else should be in Part 53.

Aligning the two frameworks in this manner would improve transferability between Parts 53 and 57 by allowing applicants to carry forward regulatory approaches, operational programs, manufacturing methods, and resolved licensing issues without unnecessary re-review. This includes flexible use of codes and standards, graded quality assurance, manufacturing-license concepts, standardized operational program review, generic finality, construction general license or at-risk construction concepts, remote and autonomous operation provisions, and change-control tools for standardized designs. For example, applicants should be able to propose alternative codes and standards in Part 53, but they should still have to justify why those standards are appropriate, adequate, and consistent with the safety case.

Part 53 should also incorporate a general license concept similar to that proposed in Part 57. Under Part 57, a general license would allow an applicant, after issuance of a construction permit, to undertake specified construction activities at its own risk before receiving final authorization to operate. Such an approach can support more efficient project execution, provide greater regulatory predictability, and allow applicants to manage project schedules while assuming the associated business risk.

Part 53 should also incorporate a broader concept of generic finality. If the NRC has already resolved a design feature, SSC, manufacturing process, operational program, or safety analysis, that resolution should carry forward when the same issue appears in a later application. Later reviews should then only focus on what is new, changed, unresolved, or site-specific. In practice, the NRC has already demonstrated elements of this approach through the Hermes and Hermes 2 reviews, where information, analyses, and conclusions from the earlier proceeding informed review of the subsequent application, rather than requiring every issue to be reconsidered from the beginning.[13] The NRC's Office of Nuclear Reactor Regulation Director stated that the Hermes 2 review was completed "using about 60% fewer resources than expected, using insights from our previous Kairos review."[14] Formalizing a concept of generic finality would provide a clearer and more predictable framework for this type of regulatory efficiency between Parts 53 and 57.

One example of what should *not* be pulled into Part 53 is emergency planning. In Part 57, the emergency-planning simplification is justified because the 1-rem entry criterion already functions as the risk screen. Part 53 should instead rely on the existing risk-informed emergency-planning framework—applicants should perform the appropriate consequence analyses and emergency planning evaluations, with any resulting emergency planning requirements determined through the existing risk-informed framework, including the upcoming applicable provisions of Part 51.

3. Contingent Analysis and Recommendation

As stated in the above section, if Part 57 is intended to be a microreactor and/or low-consequence reactor rule, the rule should not remain standalone and should be enveloped into Part 53 as guidance as a high-volume pathway. A rule scoped around the label "microreactor" risks creating an arbitrary boundary that is not tied to actual safety significance, consequence, or deployment model. Some microreactors may need first-of-a-kind review, design iteration, and site-specific analysis. Some non-microreactors may have comparable or lower safety consequences. Part 53 is already intended to provide a technology-inclusive, risk-informed, and performance-based

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[13] Nuclear Regulatory Commission, *Safety Evaluation Related to the Kairos Power LLC Construction Permit Application for the Hermes 2 Test Reactor Facility*, July 2024, ML24200A115.

[14] Nuclear Engineering International. "US NRC Completes Safety Review for Hermes 2 Test Reactor Facility." July 2024. https://www.neimagazine.com/news/us-nrc-completes-safety-review-for-hermes-2-test-reactor-facility/.

framework for advanced reactors. A separate rule is not necessary merely to license reactors with lower consequence profiles if the same regulatory outcome can be achieved by applying Part 53 in a risk-informed manner.

Part 57 is justified as a separate rule only if it solves a different problem: high-volume licensing. Its independent value is the ability to support rapid, repeatable deployment of standardized reactors after key design, safety, operational, manufacturing, and environmental issues have already been resolved.

Table 3. Recommendations – Part 57 and Part 53

Affected Section	Recommendation
§ 57.18(b); § 57.142(e); § 57.45(d); implementation guidance	Establish an explicit FOAK-to-NOAK transferability mechanism allowing a developer that has licensed a first-of-a-kind reactor under Part 53 to access Part 57's deployment tools once the design and operational programs are sufficiently standardized; NRC review in the Part 57 proceeding should focus on what is new, changed, unresolved, or site-specific rather than re-reviewing matters already resolved in the prior Part 53 proceeding. Identify when Part 53 is the more appropriate pathway for earlier-stage designs and how work under Part 53 can inform a subsequent Part 57 application.
§ 57.18(b); implementation guidance	Revise § 57.18(b) to include Part 53 approvals and licenses within the scope of referenceable prior actions, and to name early site permits expressly among the referenceable Commission approvals; explain how prior design work, safety analyses, operational program approvals, and banked site approvals from Parts 50, 52, and 53 may be carried into a Part 57 application without duplication.
Implementation guidance	Provide implementation guidance explaining the design and program maturity expected before a Part 57 joint CP/OL application is submitted; explain how pre-application engagement may be used to build eligibility confidence.

Affected Section	Recommendation
10 CFR Part 53	Incorporate into Part 53 the Part 57 provisions that do not depend on the six-month review model or the 1-rem entry criterion, including flexible codes and standards, graded quality assurance, manufacturing-license concepts, generic finality, general license for at-risk construction, remote and autonomous operation provisions, and change-control tools for standardized designs; these tools are necessary both on their own merits as risk-informed regulatory approaches and to enable clean transferability from Part 53 FOAK licensing to Part 57 NOAK deployment.
Contingent Recommendation	
Preamble; 10 CFR Part 53	If Part 57 is retained as a microreactor or low-consequence reactor rule rather than a high-volume licensing pathway, it should not remain a standalone rule; its provisions should be incorporated into Part 53 as a high-volume licensing pathway and associated guidance, because a rule scoped to the "microreactor" label risks an arbitrary boundary untethered from safety significance, consequence, or deployment model that Part 53's technology-inclusive, risk-informed framework can already address.

IV. Entry Criteria and Eligibility

Part 57's eligibility criteria are the core legal and technical boundary of the proposed framework. The NRC is proposing to reduce regulatory burden and accelerate deployment only for reactors that first demonstrate a consequence-based safety case. That structure is sound in principle. A pathway that is available only after a reactor satisfies defined consequence-based eligibility criteria is different from a general shortcut. It can support risk-informed regulation because the reduced procedural and operational requirements are justified by the reactor's demonstrated inability to create large offsite consequences. Consequence-based eligibility criteria should therefore be understood as tools that enable Part 57's high-volume licensing purpose, not as the purpose of the rule itself.

The eligibility criteria will determine which designs may use Part 57, how much analytical conservatism applicants must accept, how site boundaries are established, and how much regulatory burden is later imposed through emergency planning, security, staffing, inspections, and environmental review. If the criteria are too vague, Part 57 will create uncertainty at the front end. If they are too rigid, the rule may exclude designs that can demonstrate comparable or lower consequences through better performance-based evidence. The final rule should make the entry criteria as clear, technically defensible, and performance-based as practicable.

1. Principal Analysis and Recommendations

A. Preserve and Clarify the 1 rem TEDE Criterion as the Primary Eligibility Screen

BTI supports using the § 57.25(a) accident consequence criterion as the central eligibility screen. The 1 rem TEDE criterion gives Part 57 an objective consequence-based anchor and helps distinguish eligible reactors from larger or more complex reactors that should remain under Part 50, Part 52, or Part 53. Since Part 57 is premised on regulatory proportionality, the rule should make clear that the accident consequence criterion is not merely one technical input among many. It is the principal basis for right-sizing requirements.

The NRC should clarify how applicants must demonstrate compliance with § 57.25(a). The proposed framework allows the use of MHA or MCA methodologies. That flexibility is useful, but the final rule and guidance should explain the practical consequences of each approach. Applicants should understand that the use of a highly conservative MHA will tend to produce larger site boundaries and broader safety classifications, while an MCA may require greater analytical support but allow a more realistic licensing basis. The NRC should also clarify how the selected methodology affects site parameter envelopes, SSC classification, emergency planning, and security analyses.

The NRC should also clarify how the 1 rem TEDE criterion affects SSC classification. NUREG-2271 explains that the MHA or MCA source term is used not only to establish the site boundary, but also to determine the level of design, qualification, testing, and maintenance of SSCs needed to keep consequences below the § 57.25(a) criterion. If applied too conservatively, the same design could classify more SSCs as safety-related under Part 57 than it would under Parts 50 or 52, potentially increasing rather than reducing burden. The NRC should ensure that the Part 57 classification approach remains consequence-based and proportionate, and should avoid converting the 1 rem eligibility screen into an overly conservative driver of safety-related SSC scope.

The final rule should avoid allowing the 1 rem TEDE criterion to become detached from the rest of the framework. If an applicant demonstrates that a reactor remains below the criterion under the required assumptions, subsequent requirements should be calibrated to that showing. The NRC should not apply Part 57 eligibility at the front end and then later reintroduce large-reactor assumptions through guidance, staff review practices, or adjacent rule parts without explaining why those requirements remain necessary for a reactor that satisfies the Part 57 consequence-based eligibility criteria.

B. Treat MHA and MCA as Consequence-Based Methods, not Frequency-Based Screening Tools

The NRC should make clear that the MHA and MCA approaches are consequence-based methods. They are used to test whether a bounding or credible accident remains within the dose criterion. They are not frequency-based risk screens and should not be converted into them.

That distinction matters for applicability. The preamble explains that the MHA approach uses conservative bounding assumptions without relying on probability, while the MCA approach excludes physically unrealistic scenarios and may require additional analytical support to justify a more realistic source term. That flexibility is appropriate. The final rule and guidance should explain the practical consequences of each approach clearly enough for applicants to evaluate the tradeoff early. An applicant using the MHA approach should understand that conservative bounding assumptions will tend to produce larger site boundaries and broader SSC classification scope. An applicant using the MCA approach should understand what additional analytical support the NRC will expect to justify the more realistic source term.

The absence of a full frequency analysis should not be treated as a defect in Part 57, nor should it trigger PRA-style burdens that are unnecessary for a low-consequence eligibility determination. Where a reactor can demonstrate that consequences remain below the 1 rem TEDE criterion under either MHA or MCA methodology, the safety question for eligibility has been answered in the terms the rule itself establishes. The NRC should not require applicants to add PRA merely because the consequence-based screen does not itself quantify event frequency.

This position is consistent with BTI's Part 53 comments: risk information should be useful where it improves decision-making, but it should not become an all-purpose procedural burden.[15] Part 57 should not import PRA-style complexity through staff expectations or guidance when the rule's own eligibility structure is based on bounded consequences. Requiring frequency analysis as a routine supplement to MHA or MCA would risk undermining the central point of Part 57: if the consequences are bounded, the regulatory framework should be right-sized accordingly.

The NRC should also avoid creating unnecessary cliff-edge effects through the MHA and MCA methodology choices. The final rule should not make eligibility turn on unclear staff preferences between methodologies or on whether an applicant uses a conservative bounding case rather than a more realistic credible accident case. The agency should provide clear guidance on acceptable methodologies, assumptions, and margins so applicants can evaluate eligibility early without being forced into excessive analysis merely to avoid methodological uncertainty.

C. Replace the 10 Metric Ton Fuel-Mass Limit with Performance-Based Criteria

The proposed § 57.25(b) fuel-mass limit creates an internal tension in the framework. The fixed 10 metric ton heavy-metal limit operates as an independent eligibility bar regardless of whether an applicant has already demonstrated compliance with the § 57.25(a) dose criterion. Fuel mass is a proxy for safety consequence, but it is not a direct measure of accident consequence, decay heat removal capability, fission product retention, passive safety performance, or security risk.

That proxy relationship is imprecise in ways that matter for the rule's stated purpose. A modular reactor could have a larger fuel inventory than a microreactor and still demonstrate low site-boundary consequences because of lower decay heat, passive or inherent safety characteristics, robust fission product retention, low-pressure operation, or simpler accident progression. A fixed fuel-mass limit would exclude that design even if its safety case is more thoroughly demonstrated than a smaller reactor that barely satisfies the dose criterion. That outcome is inconsistent with consequence-based eligibility.

The NRC's own preamble reflects this uncertainty. The proposed rule suggests that the 10 metric ton limit is intended to keep accident progressions simpler and reduce possible radioactive

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[15] 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.

releases. However, the 1 rem TEDE criterion already limits the modeled accident dose to the public; the design criteria attributes already require the applicant to show heat removal, fission product retention, and reactivity control; and fuel design requirements already require limits that keep normal-operation and anticipated-event doses within Part 20 without relying on active safety systems. The agency acknowledges that the fuel-mass limit is a deterministic screen and specifically solicits comments on whether a performance-based alternative would be preferable. BTI's view is that it would. The NRC should replace § 57.25(b) 10 metric ton fuel-mass limit with a performance-based criterion tied to accident progression, such as the adiabatic heat rate threshold the NRC has identified as an alternative.

The fixed fuel-mass limit could also distort reactor design decisions in ways that do not necessarily improve safety. Designers may be pushed to reduce heavy-metal inventory, alter fuel form, shorten core life, lower power density, split a design into smaller modules, or accept more frequent refueling or replacement cycles simply to remain below an eligibility threshold. Those choices may increase operational complexity, transportation frequency, fuel-handling activity, or lifecycle cost without reducing accident consequences. Conversely, a design with a somewhat larger fuel inventory but stronger passive heat removal, more robust fission product retention, lower-pressure operation, slower accident progression, or greater safety margin could be excluded even if it presents lower actual risk. A performance-based criterion would better allow applicants to optimize reactor architecture around safety-relevant outcomes rather than around an inventory proxy.

Performance-based criteria tied directly to the safety functions that justify Part 57 treatment—site-boundary dose, decay heat removal capability, source term, radionuclide retention, accident progression, and the ability to maintain fuel and fission product barriers without active systems or prompt operator action—would better align the eligibility screen with the rule's stated purpose and with the safety features that distinguish eligible reactors from large light-water reactors.

D. Add an Entry Criterion for Physical Security

Part 57's security provisions bound consequence but not application complexity. Section 57.25 limits the safety case to what can be demonstrated below 1 rem TEDE without crediting active systems or operator action, which bounds design and review complexity and makes the safety review standardizable and repeatable. The security provisions contain no similar limitation. An applicant that satisfies § 57.25 may nonetheless rely on a full operational physical security program under § 57.325, including a credited armed response to interdict and neutralize the design-basis threat under § 57.325(b)(4)(iv) whose scope, site-specificity, and analytical burden are unbounded. A security case resting on credited timely neutralization is the least standardizable element of a reactor application: it is site-specific, cannot be resolved generically at the design or manufacturing-license level, and must be re-established at each deployment. Left unbounded, it is incompatible with both the streamlined review timeline and the generic-finality and manufacturing-license tools on which high-volume deployment depends.

The NRC should add a security eligibility criterion that bounds the form of the protective strategy rather than imposing a new reduced reference dose limit. To qualify for Part 57, an applicant should demonstrate that the § 73.100(b)(3) reference values can be met against the design-basis threat, at a minimum, through the design's inherent characteristics, engineered security-by-design features under § 57.30(f), and standardized programmatic controls—relying on delay and denial rather than on a credited, site-specific armed response whose adequacy would require case-by-case adversary-pathway and response-timeline analysis. This criterion is materially less restrictive than the unmitigated screen in § 57.60(a)(8)(v)(A)(3), because it credits engineered and standardized operational measures rather than none; it is more bounded than an open-ended § 57.325 program, because it excludes reliance on the one element that cannot be standardized or reviewed on a compressed schedule. It is also internally coherent with the rule's own determination that force-on-force exercises are unnecessary for this reactor class: a design that needs not be validated through force-on-force should not be permitted to depend on credited timely neutralization to satisfy its security objective.

Adopting the fully unmitigated as the entry threshold in § 57.60(a)(8)(v)(A)(3) was considered, but would be too strict: it credits nothing operational and requires the design alone to hold consequences below the reference values, a bar so demanding that few designs could meet it. This would leave the streamlined pathway practically empty and exclude standardized designs whose security cases are nonetheless fully bounded and repeatable.

The existing unmitigated assessment in § 57.60(a)(8)(v)(A)(3) is then best understood not as the entry test but as a further-streamlining determination: a design admitted under § 57.25(c) that can also meet the reference values with no credited mitigation or operator action receives the additional relief of operating without an operational physical security program, while a design that meets the entry criterion through standardized programmatic delay-and-denial measures remains eligible and implements those measures. Designs that cannot meet the entry criterion at all remain free to license under Parts 50, 52, or 53, but they are not, by their nature, candidates for standardized high-volume deployment.

E. Clarify how Section 57.30 Design Criteria Attributes Function

The design criteria attributes in § 57.30 connect the numerical entry criteria to the design features that make low-consequence operation plausible. They are important to the coherence of the eligibility framework, but the proposed rule does not clearly explain their legal and operational status.

The NRC should clarify whether the § 57.30 attributes are independent eligibility requirements, design-basis requirements, review criteria, or guidance-level expectations that must be reflected in the applicant's principal design criteria. If the attributes are enforceable eligibility requirements, applicants need clear acceptance criteria and clear methods for demonstrating compliance. If they are intended to guide the formulation of principal design criteria, the rule should explain how much design-specific tailoring is permitted and how the NRC will evaluate compliance during review.

The final rule should also ensure that the attributes are not converted into prescriptive design requirements. Part 57 should remain open to different ways of demonstrating heat removal, fission product retention, reactivity control, and security by design, where the applicant can support its approach. The appropriate question is whether the design performs the relevant safety functions with sufficient reliability and margin to support the consequence-based safety case—not whether the design uses a particular architecture or safety strategy.

F. Resolve Eligibility Questions

The NRC should clarify how eligibility is evaluated when multiple reactors are deployed at a single site, within a nuclear plant with shared SSCs, within a large designated area, or as manufactured units deployed at different locations over time. As discussed above, Part 57's finality and standardization provisions raise distinct questions about how licensing conclusions scale across multi-unit deployments. The eligibility question is different but related: the rule should explain at what level of aggregation (reactor, plant, site, or fleet) the § 57.25(a) and § 57.25(b) criteria are applied.

The proposed rule does not state whether the 1 rem TEDE criterion is evaluated for a single reactor, the limiting reactor, the nuclear plant as a whole, concurrent operation of multiple reactors, shared SSC failures, or credible multi-unit interactions. How the fuel-mass limit applies when several reactors are co-located or when reactors are replaced, refurbished, decommissioned, or staged over time is also unaddressed. Without that clarification, applicants may face uncertainty about whether Part 57 eligibility is determined at the reactor, plant, site, or fleet level.

This creates a potential inconsistency between the dose criterion and the fuel-mass criterion for multi-unit plants. If the 10 metric ton fuel-mass limit applies per reactor, but the 1 rem TEDE criterion is evaluated cumulatively across a multi-unit plant or shared-SSC configuration, then integrated multi-unit designs may effectively have less dose margin per reactor than individually licensed or geographically separated reactors. That result may be appropriate in some configurations, but the rule should state the principle clearly rather than leave it to case-by-case interpretation.

Shared SSCs raise the related concern. Integrated designs may use shared heat removal, shielding, control, support, security, or balance-of-plant systems in ways that improve safety, reduce complexity, or lower cost. If Part 57 treats shared SSCs primarily as aggregation risks without recognizing their safety benefits, the rule could unintentionally penalize integrated multi-unit designs. Developers may respond by separating reactors geographically, duplicating systems, or licensing units individually to avoid cumulative-dose constraints, driving design and deployment choices for licensing convenience rather than safety.

The final rule should therefore explain how cumulative dose, per-reactor fuel mass, shared SSCs, and multi-unit interactions will be evaluated together. The NRC should require applicants to address credible shared-system failures and common-cause interactions, but it should not create a default assumption that integrated multi-unit designs are less eligible for Part 57 than separated single-unit deployments. This issue should be handled without undermining the rule's consequence-based safety screen. The NRC should not assume that the existence of multiple small reactors automatically creates a large-reactor risk profile. But it should require applicants to address credible interactions, shared systems, common-cause vulnerabilities, and deployment configurations that could affect the consequence analysis.

2. Contingent Analysis and Recommendation

If preserved as a microreactor and comparable low-consequence rule, the final rule should still explain the technical basis for the threshold, clarify how the limit relates to the § 57.25(a) dose criterion, and clarify the unit of application for multi-reactor configurations in the preamble and regulatory text.

A contingent approach that could reduce the tension between the two entry criteria, if maintained as a microreactor and low-consequence rule, would be to define the pathway around microreactors and NRC-defined small modular reactors while retaining the § 57.25(a) dose criterion as the operative safety screen. The existing § 50.2 definition of small modular reactor provides a technology-neutral boundary that is already familiar to applicants. Cross-referencing that definition, while preserving § 57.25(a) as the primary safety gate, would avoid using a 10 metric ton proxy to solve a definitional problem that can be addressed more directly through the microreactor and modular reactor framing. Under that approach, Part 57 would not become available to every reactor below 1,000 megawatts thermal per module—eligibility would still require a low-consequence showing under § 57.25(a) supported by the § 57.30 design criteria attributes—but the framework would avoid excluding modular reactors solely because they exceed a fixed fuel-mass threshold despite having lower decay heat, simpler safety cases, and demonstrably low accident consequences.

The preamble provides a review of SNM quantities across reactor types and identifies that, with the exception of a large molten salt reactor, the non-large-LWR designs evaluated contained no more than approximately 9.3 MTHM at the beginning of an operating cycle. The NRC should explain more directly how that empirical distribution informs the 10 metric ton threshold—whether it is calibrated to exclude designs whose decay heat or source term approaches large-LWR characteristics, whether it provides a conservative margin above the observed non-large-LWR range, or whether it reflects a policy choice to limit Part 57 to a defined subset of reactor designs. Without that explanation, the threshold appears as an administrative line rather than a technically grounded consequence proxy.

If both entry criteria are maintained, as proposed or revised, the final rule should state whether both criteria must be satisfied independently, whether the fuel-mass limit provides an additional safety function beyond what the dose criterion captures, and whether a reactor that satisfies the dose criterion but exceeds the fuel-mass limit has any mechanism to seek an alternative eligibility showing. The proposed rule does not address that scenario, and the preamble does not explain the relationship between the two criteria in terms that would allow applicants or reviewing courts to evaluate whether the fuel-mass limit is independently justified.

The final rule should also state whether the 10 metric ton limit, if kept, applies per reactor, per manufactured reactor module, per nuclear plant, or per site. For configurations involving multiple reactors with shared SSCs, staged deployment, large designated areas, or replacement and refurbishment over time, the applicable unit of measurement may significantly affect eligibility. That question should be answered in the regulatory text rather than left to case-by-case staff interpretation.

Our view is that the entry criteria, as presently drafted, do not accurately reflect the valued purpose of Part 57 as a high-volume licensing framework. Neither entry criterion accurately gates Part 57's proposed purpose as a microreactor, low-consequence, or combined pathway. Nonetheless, if the NRC staff retains, in whole or in part, low-consequence or microreactor regulatory theories, the NRC should make the above revisions.

Table 4. Recommendations – Eligibility Criteria

Affected Section	Recommendation
§ 57.25(a); Preamble § II.C; implementation guidance	Clarify how the 1 rem TEDE criterion in § 57.25(a) informs SSC classification, site-boundary determination, emergency planning, and security analyses; state that compliance with § 57.25(a) is the primary safety basis for Part 57's streamlined provisions.

Affected Section	Recommendation
§ 57.25(a); implementation guidance	Clarify that MHA and MCA are consequence-based methods and that the absence of a probabilistic frequency analysis is not an analytical deficiency; provide guidance on methodology, assumptions, and acceptable margins sufficient for applicants to assess eligibility before committing to a full application.
§ 57.25(b); Preamble § II.C	Replace the § 57.25(b) fuel-mass limit with performance-based criteria tied to decay heat removal, source term, radionuclide retention, and accident progression without active systems or prompt operator action.
Add § 57.25(c) Security review eligibility.	The applicant must demonstrate that the radiological consequences of a design-basis-threat-initiated event can be maintained below the reference values in § 73.100(b)(3) of this chapter through the inherent characteristics of the design, engineered security-by-design features under § 57.30(f), and standardized programmatic protective measures, without crediting a site-specific armed response to interdict or neutralize the design-basis threat before a release exceeding those reference values could occur. A reactor that cannot meet this criterion is not eligible for licensing under this part.
§ 57.30; Preamble § II.C; implementation guidance	Clarify whether the § 57.30 attributes are independent eligibility requirements or design-basis criteria; explain acceptance criteria and demonstration methods; avoid converting attributes into prescriptive design requirements.
§ 57.25(a)—(b); § 57.60(a)(4); Preamble § II.C	Clarify how the § 57.25(a) dose criterion and § 57.25(b) fuel-mass or performance-based criterion apply in multi-unit configurations, including co-located reactors, shared SSCs, and staged or replacement deployments; state whether the dose criterion is evaluated per reactor or cumulatively.
Contingent Recommendations	
§ 57.25(b); § 57.3; Preamble § II.B	*Consider whether cross-referencing the § 50.2 small modular reactor definition would more directly address the eligibility boundary than a fuel-mass proxy, while preserving § 57.25(a) as the operative safety screen.*

Affected Section	Recommendation
§ 57.25(b); Preamble § II.C	*If the fuel-mass limit is retained, explain in the preamble the technical basis for the 10 metric ton threshold and how it was derived from the NRC's SNM quantity evaluation.*
§ 57.25(b); Preamble § II.C	*If the fuel-mass limit is retained, clarify in the regulatory text how § 57.25(b) relates to § 57.25(a) and whether any mechanism exists for a reactor satisfying the dose criterion but exceeding the fuel-mass limit to seek an alternative eligibility showing.*
§ 57.25(b); § 57.60(a)(4)	*If the fuel-mass limit is retained, clarify in the regulatory text whether the limit applies per reactor, per manufactured module, per nuclear plant, or per site, including for staged, replacement, and shared-SSC configurations.*

V. Licensing Pathways Under Part 57

Part 57's licensing pathways are designed to translate consequence-based mechanisms into real-world rapid deployment. BTI commends the NRC for proposing several promising licensing processes, including joint construction permit and operating license applications, manufacturing licenses for standardized factory-built reactors, standard design approvals, generic finality, and a limited general license for certain construction activities. NRC review should focus on the issues that materially affect safety, security, and environmental protection, while avoiding repeated review of identical information across deployments. Thus, the value of the Part 57 licensing pathways will depend on whether they work together as an integrated deployment framework.

The core implementation question is whether Part 57 will allow applicants to resolve generic issues once and carry those resolutions forward into subsequent site-specific applications, or whether it will simply create another set of optional licensing tools parallel to Parts 50, 52, and 53 that applicants must navigate.

This issue should be understood together with the eligibility concerns discussed above. Entry criteria determine whether a reactor may use Part 57. Licensing pathways determine whether that eligibility can be converted into an efficient and predictable deployment sequence. The NRC should clarify how joint CP/OL applications, manufacturing licenses, SDAs, generic finality, and the general construction license interact; what information can be resolved generically; what must remain site-specific; how prior approvals under other frameworks may be referenced; and how departures from approved designs or programs will be reviewed. Without that implementation clarity, these otherwise promising tools could become fragmented pathways rather than a coherent high-volume licensing framework.

1. Principal Analysis and Recommendations

A. FOAK vs NOAK

BTI supports the proposed joint CP/OL pathway as a central feature of Part 57. For reactors with mature designs, standardized operational programs, and well-defined site-parameter envelopes, a joint application can reduce duplicative review by allowing the NRC to evaluate construction and operation together. This structure is especially valuable for NOAK deployment, where the NRC should be able to rely on previously resolved design and programmatic issues and focus later reviews on site-specific, applicant-specific, environmental, and departure-related matters. The preamble states this process is designed to support OL issuance within 6 to 12 months of application acceptance. The pathway intentionally avoids the ITAAC structure of Part 52 and instead achieves licensing efficiency by front-loading the complete design submission rather than closing out requirements progressively during construction.

Some stakeholders may raise this as a distinct FOAK usability issue. The complete-design requirement is not a burden to be relieved; it is what makes the single safety review and compressed timeline possible. The NRC cannot conduct a comprehensive review in 6 to 12 months without final design information and complete operational programs at the outset. Recommending that the NRC modify this requirement to accommodate less mature FOAK designs would undermine the pathway's core function. The pathway is sound as structured and evolves from existing pathways such as Part 52.

The practical question is different: a developer whose design is not yet mature enough to support a complete Part 57 application should not be using Part 57 at that stage. Part 53 provides an alternative pathway for applicants at earlier design-development stages (and can be improved as discussed in Section III), and the NRC's existing pre-application engagement processes are available to help developers build confidence in their approach before committing to a full application. At the May 20, 2026, public meeting, NRC staff confirmed this directly in response to a stakeholder concern that the joint CP/OL pathway precludes first-of-a-kind licensing by eliminating any preliminary design or PSAR pathway.[16] Staff stated that a FOAK applicant may use Part 57 if the design is mature, that Parts 50 and 53 remain available for less mature designs, and that the standard design approval under Subpart E provides an incremental review option for applicants who are not yet ready for a full application. The issue is not that Part 57 is unavailable to FOAK applicants but that developers may not have a clear sense of what design maturity is required before Part 57 becomes the right choice. FOAK deployments are inherently prototypes due to the simple nature of FOAK, and that is not what rapid, high-volume deployment is.

Two things would address the pathway selection and pre-application engagement problem. First, the NRC should provide clear guidance explaining the design and program maturity expected before a Part 57 joint application is submitted, and should identify how pre-application engagement can be used to build that confidence without requiring a premature full application commitment; this includes pre-application meetings, requests for additional information, and staff feedback on eligibility. The guidance should also explain when Part 53 is likely the more appropriate pathway for earlier-stage designs, and how work developed under Part 53 can inform a subsequent Part 57 application once the design is sufficiently mature.

Second, and relatedly, the pathway from an earlier FOAK proceeding under Part 53, Part 50, or Part 52 into a Part 57 NOAK deployment is not addressed in the proposed rule. As discussed in Section III, proposed § 57.18(b) allows a Part 57 applicant to incorporate by reference prior approvals from Parts 50 and 52 but not Part 53. A developer who matures a design through Part 53 and then seeks Part 57 NOAK deployment cannot reference that prior work. The NRC should revise § 57.18(b) to include Part 53 approvals, and should explain in implementation guidance how prior design work, safety analyses, and operational program approvals from any of the existing frameworks may be carried into a Part 57 application without duplication.

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[16] Nuclear Regulatory Commission, Public Meeting Schedule: Meeting Details—Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles (Meeting Code 20260503), May 20, 2026, 1:00—3:00 p.m. ET, https://www.nrc.gov/public-involve/public-meetings/pmns/20260503.

B. High Volume Licensing Architecture

High-volume deployment under Part 57 requires tools that allow the NRC to resolve common design, safety, environmental, and operational issues once and to rely on those resolutions across repeat applications without reopening settled questions. Part 57 provides those tools, but does not explain how they are intended to work together or how applicants should sequence them to realize the rule's deployment potential.

The final rule and accompanying implementation guidance should explain how these tools interact—how generic finality obtained in a FOAK proceeding reduces the scope of NOAK reviews, how manufacturing license finality on the reactor design relates to the site-specific elements of a subsequent construction permit application, and how standardized operational programs approved in one proceeding can be incorporated by reference in subsequent applications without duplicative review.

C. Multiple-Unit and Shared-Facility Deployments

Many of the deployment scenarios that motivate Part 57—remote industrial facilities, defense installations, distributed energy applications—will involve multiple identical or substantially similar units operating at the same location or within a designated large geographical area. The rule's high-volume licensing tools are most valuable precisely in those scenarios. The proposed rule addresses certain aspects of multi-unit deployments in § 57.60(a)(4), requiring applicants to analyze configuration limits, evaluate hazards to safety-related SSCs arising from co-located construction and operation, describe shared infrastructure, and include technical specifications for shared portions of the nuclear plant. Those requirements address physical interactions among units, but they do not resolve the broader question of how the rule's finality and standardization provisions apply as the number of deployed units grows.

A licensing basis adequate for a small number of co-located units may not automatically extend to deployments of ten, twenty, or more identical reactors at the same site. Shared infrastructure and resources may create interaction effects that single-unit or small-fleet safety analyses do not capture. Cumulative environmental impacts, common-cause failure modes, and security perimeter sizing all have the potential to scale in ways that the per-unit analysis supporting a FOAK or early NOAK application does not address.

Multi-unit deployments are among the most commercially significant applications Part 57 is designed to support. But, clarity is needed on if and how the rule expects applicants to analyze as units are added (e.g., a licensed facility of 10 units that later chooses to add two additional units), and what the NRC will treat as generically resolved versus requiring site- or fleet-specific evaluation. Providing that framework now, rather than resolving it through case-by-case staff judgment, would improve licensing predictability and reduce the risk of inconsistent treatment across multi-unit applications. The existing requirements of § 57.60(a)(4) should be clarified to distinguish between the physical interaction analysis already required and the fleet-level cumulative review question the rule does not yet address.

D. Make Manufacturing Licenses a True Deployment Tool

For factory-built low-consequence reactors, the manufacturing license (ML) may be one of the most important licensing innovations in the proposed rule. If implemented correctly, it can allow the NRC to review and approve standardized reactor designs rather than repeating the same review for every deployment. This approach is well-suited to high-volume licensing. Many microreactor or manufactured reactors business models depend on repeat production, factory fabrication, transport to deployment sites, and standardized operation. While the NRC will still authorize construction and review site-specific considerations, a manufacturing license can help move NRC review from a project-by-project model toward a product-and-fleet model. It can also provide a practical bridge from FOAK deployment to NOAK deployment by allowing design, manufacturing, and generic programmatic issues to be resolved before repeated site-specific applications.

The transportation provisions, however, could limit the manufacturing license's practical flexibility. NUREG-2271 recognizes that, where a reactor cannot meet all Part 71 requirements, applicants may pursue alternatives, including the use of a risk-informed methodology in lieu of deterministic compliance with § 71.41(a). The guidance also notes that NRC endorsement of available risk-assessment methodologies applies to the methodology for performing calculations, not necessarily to the assumptions or estimated results. Applicants must therefore demonstrate that the selected approach results in an acceptably low level of risk. To make this framework predictable for high-volume deployment, the NRC should define more clearly what "acceptably low" means in this context.

The NRC should also clarify criticality and security requirements for fuel-manufactured reactors. The manufacturing license should identify the features relied on to prevent criticality during storage and transport, such as fixed neutron absorbers, locked reactivity control mechanisms, configuration controls, or other design-specific measures. The NRC should also clarify how physical security and cybersecurity requirements apply during transport, including the division of responsibilities among the manufacturer, transporter, and site licensee.

Post-shipment inspection requirements should also be standardized. Proposed Part 57 appropriately requires inspection upon arrival to verify that the manufactured reactor remains in compliance with the manufacturing license, has not been damaged in transit, and satisfies interface requirements with the rest of the nuclear plant. For high-volume deployment, the NRC should provide consistent inspection criteria, documentation expectations, acceptance standards, and corrective-action processes to enable the pathway to function as intended

Transportation flexibility must also be implemented in a way that supports public confidence. The NRC should explain how alternative analyses provide an equivalent or superior safety basis, how shipment integrity will be verified, and how emergency response, safeguards, and security considerations will be addressed. Clear, performance-based criteria will better support both innovation and transparency than open-ended case-by-case determinations. Without reasonable bounds, transportation could become a recurring bottleneck for every manufactured reactor shipment and undermine the high-volume licensing purpose of Part 57.

E. Preserve SDA Flexibility While Clarifying Reopening and Audit Expectations

BTI supports the standard design approval (SDA) as a "stepping stone" towards high-volume standardization under proposed Part 57. The ability to obtain approval for an entire reactor design or a major portion of a design can provide an important early regulatory milestone for low-consequence reactors, especially FOAK designs that are not yet ready for a manufacturing license or joint CP/OL application. BTI also supports the NRC's proposal to make Part 57 SDAs indefinite in duration. Indefinite duration can reduce duplicative review in the long run, improve regulatory stability, and allow an approved design to remain available for later reference in manufacturing license or deployment applications.

These features are especially valuable because SDAs may function less as a NOAK deployment tool and more as a FOAK design-maturation tool. A developer may use an SDA to resolve key design issues, mature interfaces, support financing and customer commitments, and then proceed to a manufacturing license or joint CP/OL application once the design and deployment model are more fully developed. However, the NRC should clarify two implementation issues to ensure the SDA pathway provides practical value.

First, the NRC should define the circumstances under which an SDA may be reopened based on "significant new information that substantially affects the earlier determination or other good cause." BTI recognizes the need for a safety valve to address materially relevant new information, design obsolescence, or changes in the state of knowledge. But if this standard is applied too broadly, indefinite SDA duration may provide limited commercial certainty. The final rule or guidance should make clear that reopening should be limited to information that is material to the underlying safety determination and not used to relitigate issues previously resolved in the SDA review.

Second, the NRC should ensure that audit-readiness expectations do not undermine SDA's value as a flexible design-maturation tool. If a later ML or CP/OL application references an SDA, NRC review may appropriately confirm that the later application conforms to the approved design and satisfies relevant interface conditions. But engineering-document audits should not become a second full design review or require applicants to have complete deployment-level engineering maturity at the SDA stage. The SDA should remain a tool for resolving design issues earlier in the commercialization process, not a de facto complete application.

For SDAs covering major portions of a design, NRC should also clarify how interface requirements will be documented and carried forward. Applicants should know which findings are final, which assumptions must be verified later, and which interface conditions remain open for ML or CP/OL review. Clear treatment of these issues would make SDAs more useful as a bridge from FOAK design development to standardized manufacturing and deployment under Part 57.

F. Strengthen Generic Finality

Generic finality is essential to Part 57. High-volume deployment cannot occur if every subsequent application can reopen settled design and programmatic issues.

First, the initial application requesting generic finality will require substantially more upfront information than a standard site-specific application through Parts 50, 52, or 53. Applicants must define site-parameter envelopes, evaluate the design against relevant external hazards, and identify which operational-program elements are generic and which must remain site-specific. This burden may be appropriate where an applicant seeks fleet-wide licensing value, but NRC guidance should make the expected scope and level of detail predictable.

Second, because generic finality can bind future proceedings, the proposed rule appropriately broadens the opportunity for intervention on generic matters during the initial licensing case. That approach protects procedural rights, but it also means the first generic-finality proceeding may be more complex than a standard site-specific licensing case. NRC should manage this complexity by clearly distinguishing generic issues from site-specific issues and by establishing efficient hearing procedures for issues that, once resolved, are intended to govern later NOAK applications.

The NRC should also explain how departures from the approved design affect finality. Minor site-specific adaptation should not automatically destroy the value of a standardized approval, but material departures that affect the safety case should be reviewed under defined standards. The NRC should define departure categories, specify the level of analysis required for each, and clarify when a departure is safety-significant enough to require expanded staff review, ACRS review, or litigation.

The final rule should distinguish among at least three categories of issues: issues resolved generically and not subject to ordinary relitigation; issues resolved generically but reopenable under defined standards for new information or material departures; and issues that remain site-specific from the outset. That structure would make finality more transparent for applicants, intervenors, and NRC staff. It would also reduce the risk that nominally standardized approvals are eroded by case-by-case uncertainty.

G. Clarify the Scope and Details of the Proposed General License for Construction

The proposed general license for certain construction activities in § 57.45(d) is an important and innovative feature of Part 57. For low-consequence reactors intended for high-volume deployment, the ability to begin limited construction activities before issuance of a site-specific construction permit can help translate generic regulatory finality into actual deployment efficiency. If implemented clearly, the general license could reduce avoidable schedule delay for nth-of-a-kind (NOAK) facilities without sacrificing safety or security.

However, the final rule and implementation guidance should provide greater clarity in several areas.

First, the NRC should clarify how the proposed definition of "construction" in § 57.3 applies to actual implementation. Because the definition of "construction" focuses on safety- and security-related SSCs and their foundations, applicants will need clear guidance on which activities may proceed under the general license and which require a construction permit. The NRC should provide examples covering grading, excavation, foundation work, embedded components, reactor-building structures, security structures, electrical and heat-transfer interfaces, utility connections, shared SSCs, and balance-of-plant work. Without this clarity, applicants may either underuse the general license flexibility or undertake work that later requires rework or creates licensing uncertainty.

Second, the NRC should ensure that environmental prerequisites do not eliminate the benefits of shorter timelines associated with the general license. Proposed § 57.45(d) requires applicable permits and federal environmental consultations to be completed before use of the general license, and requires a redress plan for adverse environmental impacts. These are important safeguards, but the NRC should clarify how they interact with the environmental review for the CP/OL. Otherwise, environmental review could become the bottleneck to general license issuance

Finally, the NRC should consider how a design first licensed or demonstrated outside Part 57 can become eligible for NOAK use of the general construction license. The proposed general license is available only after the NRC has docketed a joint CP/OL application that references a manufacturing license, a prior Part 57 CP/OL afforded generic finality for the same manufactured reactor design, and that has qualified for a categorical exclusion from NEPA requirements as established in Subpart K.[17] That may be too narrow for any true FOAK deployments under Part 57, but not for early deployments under 57 that *have* been licensed under Part 50, Part 52, or Part 53 previously. The final rule should allow equivalent prior approvals to support NOAK use of the

[17] § 57.45(d)(1)

general construction license where the applicant demonstrates that the design meets Part 57 eligibility criteria, that the relevant generic issues were previously resolved with adequate finality, and that subsequent Part 57 review is limited to applicant-specific, site-specific, and departure-related matters. Implemented this way, the general license for construction would remain appropriately limited to NOAK reactors, but would become a practical bridge between generic finality and rapid deployment.

H. Provide an applicant-facing pathway map

Finally, the NRC should provide a pathway map for Part 57 applicants. That map should explain how joint CP/OL applications, manufacturing licenses, SDAs, generic finality, and the general construction license may be combined. It should also identify the expected sequencing of related approvals under Parts 70, 71, 72, 73, and 140, where those approvals are relevant to a deployment model.

The pathway map should be practical rather than merely descriptive. It should show, for example, how a FOAK applicant could mature a design, obtain feedback on eligibility, pursue an SDA or manufacturing license, secure finality for standardized issues, and then deploy NOAK units through site-specific applications that are limited to genuinely site-specific matters. It should also explain how an applicant can transition from Part 57 to Part 53, Part 50, or Part 52 if early eligibility work shows that Part 57 is not the right pathway.

The final rule should not leave applicants to assemble the licensing architecture from scattered subparts. The intended value of Part 57 is not simply that several pathways exist; it is that applicants can choose and combine them predictably. Clear guidance on pathway selection, sequencing, finality, and site-specific residual issues would make Part 57 more useful and reduce the risk that optionality becomes another source of regulatory uncertainty.

In sum, BTI supports the NRC's use of multiple licensing pathways in Part 57, but the final rule should be more explicit about what each pathway is for and what regulatory value it provides. The NRC should clarify design and program completeness requirements for joint CP/OL applications, make manufacturing licenses a true product-level standardization tool, define the practical effect of SDAs, give generic finality real force, make the limited general construction license usable, and publish an applicant-facing pathway map. These changes would better align Part 57 with the goal of repeatable, high-volume deployments.

Table 5. Recommendations – Licensing

Affected Section	Recommendation
Preamble § II.B; §§ 57.142(e), 57.45(d), 57.18(a)(2) and (5), 57.60(a)(8), Subpart D	Explain how Part 57's high-volume deployment tools work together as an integrated licensing architecture, including how generic finality, manufacturing licenses, general construction authority, and standardized operational programs interact across FOAK and NOAK proceedings.
§ 57.60(a)(4); implementation guidance	Clarify how the rule's finality and standardization provisions apply to multi-unit deployments; distinguish between the physical interaction analysis required under § 57.60(a)(4) and the fleet-level cumulative review question that the rule does not yet address.
Implementation guidance; conforming amendments to Part 71	Define more clearly what "acceptably low" risk means for alternative Part 71 transport analyses in NUREG-2271 for fueled manufactured reactors, to make the transportation framework predictable for high-volume deployment
§§ 57.197(d), 57.197(e); Subpart J; conforming amendments to Parts 71, 73	Clarify which design features prevent criticality during manufacturing, storage, and transport; clarify how physical security and cybersecurity responsibilities are divided among the manufacturer, transporter, and site licensee.
§ 57.197; implementation guidance	Standardize post-shipment inspection criteria, documentation expectations, acceptance standards, and corrective-action processes for manufactured reactors arriving at deployment sites.
Implementation guidance	Explain how alternative transport analyses provide an equivalent or superior safety basis, how shipment integrity will be verified, and how emergency response, safeguards, and security will be addressed; provide performance-based criteria rather than open-ended case-by-case determinations.

Affected Section	Recommendation
§ 57.220(a); Preamble § II.H	Define the circumstances under which an SDA may be reopened based on "significant new information that substantially affects the earlier determination or other good cause"; clarify that reopening is limited to information material to the underlying safety determination and may not be used to relitigate previously resolved issues.
§ 57.142; implementation guidance	Clarify audit-readiness expectations so that SDAs function as design-maturation tools; define what engineering-document audit scope is appropriate when a later ML or CP/OL application references an SDA, and confirm that audit does not constitute a second full design review.
§ 57.142; implementation guidance	Clarify interface requirement documentation for SDAs covering major portions of a design; identify which findings are final, which assumptions must be verified later, and which interface conditions remain open for ML or CP/OL review.
§ 57.142(e); implementation guidance	Provide guidance making the expected scope and level of detail for generic finality applications predictable, including how site-parameter envelopes must be defined and which operational-program elements may be treated as generic
§ 57.142(e); conforming amendments to 10 CFR Part 2	Establish efficient hearing procedures for generic finality proceedings that clearly distinguish generic issues from site-specific and applicant-specific issues.
§ 57.142(e); implementation guidance	Define departure categories and specify the level of analysis required for each; clarify when a departure is safety-significant enough to require expanded staff review, ACRS review, or relitigation.
§ 57.142(e); implementation guidance	Distinguish in the rule or guidance among three categories of issues: resolved generically and not subject to ordinary relitigation; resolved generically but reopenable under defined standards for new information or material departures; and site-specific from the outset.

Affected Section	Recommendation
§ 57.3; § 57.45(d); implementation guidance	Clarify how the definition of "construction" in § 57.3 applies to specific activities under the general license; provide examples covering grading, excavation, embedded components, reactor-building structures, security structures, utility connections, shared SSCs, and balance-of-plant work.
§ 57.45(d); Subpart K; Preamble § II.I	Clarify how the environmental prerequisites in § 57.45(d) — including federal environmental consultations and NEPA categorical exclusion qualification under Subpart K — are sequenced with the overall licensing schedule so they do not eliminate the general construction license's schedule benefit.
§ 57.45(d); Preamble § II.I	Allow FOAK reactors initially licensed under Parts 50, 52, or 53 to become eligible for NOAK use of the general construction license where the applicant demonstrates that the design satisfies Part 57 eligibility criteria, that relevant generic issues were previously resolved with adequate finality, and that subsequent Part 57 review is limited to site-specific, applicant-specific, and departure-related matters.
Implementation guidance	Publish guidance explaining how joint CP/OL applications, manufacturing licenses, SDAs, generic finality, and the general construction license may be combined and sequenced; include related approvals under Parts 70, 71, 72, 73, and 140; address developer readiness assessment and transition pathways between Part 57 and Parts 50, 52, and 53.

VI. Quality Assurance and Codes & Standards

Part 57's cost and deployment benefits depend heavily on whether the NRC actually implements quality assurance, codes and standards, and supplier qualification in a way that is commensurate with safety significance. The draft regulatory analysis estimates net averted costs of between $3.76 billion and $11.84 billion over a 40-year period, attributing much of that benefit to the reduction in exemption requests that would otherwise be required under Parts 50 and 52.[18] Those

[18] Proposed rule, 91 Fed. Reg. at 23683. *Regulatory Analysis, § IX.*

savings are achievable only if the final rule's flexibility is implemented in practice through accepted QA programs, approved non-nuclear codes, and a supplier base that is actually broadened. If staff defaults to Appendix B or § 50.55a through guidance or review expectations, the rule's QA and codes provisions will remain theoretical. The final rule and accompanying implementation guidance should close that gap. Because these savings accrue across repeated, standardized units rather than any single build, graded QA and code flexibility is best understood as a precondition for high-volume deployment, not a general administrative convenience.

BTI supports Part 57's movement away from prescriptive Appendix B and § 50.55a mandates. The sections below assess the proposed provisions and identify where the rule or guidance should provide additional clarity.

1. Quality Assurance

Proposed § 57.60(a)(3) requires applicants to describe their QA program applied to the design, fabrication, manufacturing, construction, and testing of safety-related SSCs. Unlike Parts 50 and 52, Part 57 does not mandate compliance with the 18 criteria of Appendix B. Applicants may propose and justify QA programs tailored to the safety significance of their specific SSCs, including industry-approved standards such as ANSI/ANS-15.8-1995, which governs research and test reactors.

The proposed approach is sound in principle. Appendix B was developed for large light-water reactors with complex safety systems and large source terms. Applying all 18 criteria uniformly to reactors that satisfy Part 57's consequence-based eligibility criteria adds control burden without commensurate safety benefit for lower-significance components. The flexibility to use ANSI/ANS-15.8 or comparable standards reflects appropriate recognition that those standards already provide adequate quality assurance in the research and non-power reactor context.

The preamble identifies broadening the nuclear supplier base as an explicit policy objective, noting that the flexibility to select QA programs may encourage participation from qualified commercial suppliers and mitigate risks of shortages, backlogs, and higher deployment costs. For manufacturing licenses, proposed § 57.270 requires reporting of significant QA breakdowns, equivalent to § 50.55(e). The commercial-grade dedication process must be conducted in accordance with the applicant's § 57.60(a)(3) QA program rather than Appendix B.

Two issues need clarification. First, the rule permits applicants to propose tailored QA programs but does not explain the criteria NRC staff will use to evaluate them. The preamble identifies safety significance as the organizing principle, and the preamble acknowledges that QA standards share common elements across programs. But the rule does not identify what level of rigor is expected at each tier of safety significance, what alternative standards are likely to be found acceptable, or what analytical framework staff will apply when reviewing a non-Appendix-B program. Without that guidance, applicants face uncertainty about whether their proposed programs will be accepted, and staff face the same uncertainty about how to evaluate them. The practical result could be informal convergence on Appendix B as the path of least resistance — exactly the outcome the rule is designed to avoid.

Second, for manufacturing licenses, QA is not a peripheral procurement issue; it is central to the deployment model. A manufacturing license that resolves factory fabrication controls, supplier qualification, commercial-grade dedication, configuration management, inspection and testing processes, and corrective action with meaningful finality would allow repeat production of identical reactor modules without duplicative site-by-site review of the same program. The proposed rule does not address whether QA programs approved as part of a manufacturing license proceeding are eligible for generic finality under § 57.142(e), or whether NOAK applicants referencing an approved manufacturing license may rely on its QA determinations without full re-review.

2. Codes and Standards

Proposed § 57.60(a)(9) requires applicants to provide information on the codes and standards used to design their facility. Part 57 does not incorporate by reference the specific codes mandated by § 50.55a, recognizing that many of those codes are technology-specific and may not be appropriate for microreactor designs. Applicants may choose which consensus codes and standards to apply, including international standards not previously used in NRC licensing. For each proposed code or standard, the applicant must evaluate it for applicability, adequacy, and sufficiency, and must provide justification if a code is supplemented or modified. The criteria from selected standards must be clearly stated and shown to provide appropriate levels of reliability, safety, and performance. The preamble acknowledges that the acceptability of any consensus code or standard will ultimately be determined on an application-specific basis during individual licensing review.

There are implementation uncertainties that should be considered and resolved by NRC staff. The rule's statement that any consensus code or standard will "ultimately need to be found acceptable on an application-specific basis during an individual licensing review" creates uncertainty that could undermine the flexibility the provision is designed to provide.[19] If each code selection is treated as a fresh determination without reference to prior accepted uses, applicants cannot predict whether their proposed standards will be accepted, and the review process will not become more efficient over time. The NRC should explain in implementation guidance how codes and standards approved in prior Part 57 proceedings — or in related frameworks, including Part 53 and Part 50 advanced reactor licensing — will be treated in subsequent applications, and whether approved code selections may be referenced without re-evaluation.

The rule also does not address how codes and standards selections interact with manufacturing license finality. For factory-built reactors, fabrication and assembly codes will govern repeated production runs. If the acceptability of those codes must be re-evaluated for each deployment site application rather than resolved once in the manufacturing license proceeding, the flexibility the rule provides will be limited in practice to FOAK licensing rather than the NOAK deployments where its benefit is greatest.

The proposed approach correctly recognizes that § 50.55a's ASME and other mandated codes were developed for large LWR pressure systems and are not appropriate defaults for microreactor designs that may use different materials, configurations, operating conditions, and non-water coolants. The recommended clarifications will help applicants select and justify appropriate codes and standards that reflect the same graded, function-specific logic that motivates the QA provisions.

[19] § 57.60(a)(9).

Table 6. Recommendations – Quality Assurance and Codes & Standards

Affected Section	Recommendation
§ 57.60(a)(3); implementation guidance	Provide implementation guidance identifying the criteria staff will use to evaluate tailored QA programs under § 57.60(a)(3), including expected rigor at each tier of safety significance and types of alternative standards likely to be found acceptable.
§ 57.142(e); Subpart D; implementation guidance	Clarify whether QA programs established in a manufacturing license proceeding are eligible for generic finality under § 57.142(e) and whether NOAK applicants may rely on those determinations without duplicative review.
§ 57.60(a)(9); implementation guidance	Provide guidance explaining how codes and standards approved in prior Part 57 proceedings will be treated in subsequent applications, and whether approved selections may be referenced without re-evaluation.
§ 57.142(e); § 57.60(a)(9); Subpart D	Clarify whether codes and standards selections established in a manufacturing license proceeding are eligible for generic finality under § 57.142(e) and may be relied upon by NOAK applicants.

VII. Operational Programs and Oversight

Part 57 must also establish an operational and oversight model that fits standardized and potentially distributed reactors that satisfy consequence-based eligibility criteria. The proposed rule will not achieve its purpose if it streamlines initial licensing but then applies inflexible operational programs, staffing rules, inspection assumptions, and change-control requirements. The final rule should therefore explain how operational requirements will be right-sized to the demonstrated consequence-based safety case while preserving clear accountability for safety, security, and compliance.

1. Adopt a Graded Framework for Remote and Autonomous Operation

BTI supports the NRC's decision to recognize remote monitoring, remote operation, and autonomous operation in Part 57. However, the final rule and guidance should avoid treating all remote and autonomous functions as equivalent.

The NRC should adopt a graded oversight framework that distinguishes remote monitoring, remote operation of non-safety-related functions, remote operation of safety-related functions, supervised autonomous operation, and fully autonomous operation beyond the proposed operator-dependent facility and operator-independent facility classes. Each tier should have requirements commensurate with the safety significance of the function being performed. A facility that only transmits plant data to an offsite center should not be reviewed the same way as a facility that allows remote safety-related commands, and neither should be reviewed the same way as a facility that relies on autonomous systems to perform safety functions during abnormal or accident conditions.

Several implementation issues warrant clarification to ensure oversight coherence. First, the NRC should define the evidentiary standard for autonomous operation, particularly where applicants seek to demonstrate safety under abnormal and accident conditions without reliance on human intervention or external command. Second, the NRC should reconsider whether a mandatory manual shutdown capability is always necessary for designs that demonstrate operator independence through passive or inherent safety features. A deterministic manual-shutdown floor may be appropriate in some cases, but should not become an unnecessary requirement where it does not add safety value.

Third, remote operation depends on cybersecurity, signal integrity, and communications reliability. The NRC should clarify how command pathways, data links, sensor information, and human-system interfaces will be classified, qualified, protected, and inspected. For remote or autonomous designs, a cyberattack or corrupted data stream should not be capable of defeating the safety case or producing off-site consequences above regulatory limits.

Fourth, the NRC should address connectivity and deployment realities for remote facilities. Many Part 57 facilities may be located in areas where wireless or satellite communications are difficult to maintain. The rule should define required performance for data availability, latency, redundancy, loss-of-link response, and fallback safe-state behavior.

A graded framework would preserve Part 57's flexibility while giving applicants, NRC reviewers, and the public a clearer basis for determining when remote and autonomous operation is acceptable.

2. Implement GLROs as a Clearly Bounded Risk-Informed Operator Licensing Tool

BTI supports the proposed Generally Licensed Reactor Operator (GLRO) framework for operator-independent facilities under Part 57. GLROs are appropriate where the safety case demonstrates that no operator action is required to maintain the facility within the § 57.25(a) dose-based criterion. In those circumstances, the NRC should not impose traditional individual operator licensing requirements that are premised on operators performing credited safety actions. BTI's recommendations focus on the transferability of programs between licensing frameworks and implementation clarity.

Licensed reactor operators are separately defined in §53.725(c) to operate a self-reliant-mitigation facility. The corollary in Part 57 is an Operator-Independent Facilities (OIFs). There are some differences between how the facilities GLROs can operate and how they are treated in each rule. Ideally, Part 57 would reference GLROs in Part 53 instead of redefining a license class with the same name, but different functions. The challenge is that Part 53 defines GLROs specifically for self-reliant-mitigation facilities, which do not exist in Part 57. The result is a stumbling block, where GLROs for a first-of-a-kind design in Part 53 might not be transferable to a Part 57 framework. The recommended adjustment is to amend §53.725(c) to allow GLROs to operate self-reliant-mitigation facilities as defined in §53.800, or an operator-independent facility as defined in § 57.390. Then § 57.390 should reference the definition of GLRO from §53.725(c).

Final guidance should clarify the demonstration required to establish operator-independent status; how maintenance errors, remote operations, autonomous functions, and emergency response actions are treated; when design or operational changes require reevaluation of GLRO eligibility; and when a facility must transition from GLROs to specifically licensed operators. Clear implementation guidance will allow GLROs to reduce unnecessary burden while preserving the safety significance of operator licensing.

3. Develop a Data-Driven Inspection Model for Part 57 Facilities

BTI supports the NRC's proposal to move away from the full-time resident inspector model for Part 57 facilities and toward targeted, performance-based inspections commensurate with facility risk. This is appropriate for reactors that qualify for Part 57, and that may be factory-built, remotely monitored, highly standardized, or deployed with small numbers of onsite staff.

The current framework identifies the right direction, but leaves important implementation details unresolved. NRC should clarify how inspection frequency, inspection scope, and escalation criteria will be determined.

BTI recommends that guidance identify the types of operational data, logs, alarms, surveillance results, maintenance records, cybersecurity records, and corrective-action information that NRC may access remotely before or between onsite inspections. The record *does* establish the regulatory authority for such access and identifies broad categories of information that are expected to be available to the NRC remotely or between inspections. In the final disposition of comments on the proposed rule, if stakeholders identify that a specific list is necessary, the NRC staff should revise guidance to be more comprehensive. The NRC should also define event-triggered inspection criteria, including what off-normal conditions, repeated equipment issues, human-performance concerns, or safety-significant data trends would prompt supplemental inspection.

This approach would preserve the benefits of a right-sized inspection program while maintaining regulatory confidence. For remote or isolated deployments, it would also reduce reliance on inspector proximity and allow NRC to focus on-site inspection resources where performance data indicate elevated risk. The objective should not be fewer inspections for their own sake, but better-targeted inspections based on facility risk, operating experience, and real-time or near-real-time performance information.

Table 7. Recommendations – Operational Program and Oversight

Affected Section	Recommendation
§ 57.391; § 57.60(a)(1)(vi); § 57.60(a)(8)(xi); implementation guidance	Adopt a graded oversight framework that distinguishes remote monitoring, remote operation of non-safety-related functions, remote operation of safety-related functions, supervised autonomous operation, and fully autonomous operation; apply requirements commensurate with the safety significance of each tier.
§ 57.395(a)(7)(iv); implementation guidance	Reconsider whether a mandatory manual shutdown capability is always necessary for operator-independent facilities that demonstrate through passive or inherent safety features that no operator action is required to satisfy § 57.25(a); if the requirement is retained, explain the technical basis for requiring manual shutdown where it does not add demonstrated safety value.
§ 57.60(a)(8)(xi); implementation guidance	Define required performance standards for remote monitoring and control communications infrastructure, including data availability, latency, redundancy, loss-of-link response, and safe-state behavior upon communications failure; the rule requires a description and plan for a remote operation or monitoring program but sets no performance floor, which is a practical gap for deployments at remote or isolated locations.
§ 57.391(a), § 57.400(f); implementation guidance	Clarify the trigger and process for reevaluating operator-independent classification when design or operational changes are made; specify the conditions under which a facility must transition from GLROs to specifically licensed operators under §§ 57.420—57.427, and how maintenance errors, remote operations, and autonomous functions are treated when assessing whether the § 57.25(a) criterion continues to be met without operator action.

Affected Section	Recommendation
§ 57.355; Preamble § V.M (Subpart L—Inspections); implementation guidance	Define the targeted inspection program for Part 57 facilities, including inspection frequency, scope, escalation criteria, and the types of operational data, logs, alarms, surveillance results, and corrective-action information NRC may access remotely between onsite visits; define event-triggered inspection criteria; and distinguish routine performance oversight from inspections required upon initial deployment, module replacement, or operator classification transition. The rule states only that NRC will use "targeted inspections and performance oversight" with no further specificity.
§ 57.400(a); implementation guidance	In addition to reporting the identity of all GLROs annually, new GLROs should be reported within 60 days. This does not significantly increase the burden, but would avoid a new GLRO from working 364 days before reporting.
§ 57.390; § 57.405; 10 C.F.R. Part 53; implementation guidance	Amend §53.725(c) to allow GLROs to operate self-reliant-mitigation facilities as defined in §53.800 , or an operator-independent facility as defined in § 57.390. Then § 57.390 should reference the definition of GLRO from §53.725(c).

VIII. Emergency Planning, Security, and Safeguards

Emergency planning, physical security, cybersecurity, and safeguards requirements under Part 57 should be commensurate with the consequences the reactor can plausibly create, the form and enrichment of the material it possesses, the role of security-by-design in its licensing basis, and the operational model the NRC has approved. Regulatory requirements should be proportionate to demonstrated risk rather than inherited from large light-water reactor precedents. Part 57 moves in the right direction in its emergency planning, security, and safeguards provisions.

The emergency planning provisions eliminate traditional emergency planning zones, scale classification levels to demonstrated consequences, and preserve coordination with offsite response organizations. The physical security provisions establish a consequence-based assessment that credits design features and calibrates operational security requirements accordingly. The cybersecurity provisions give applicants a choice of frameworks. The material control and accounting provisions apply a scaled version of Part 74 requirements appropriate for this reactor class.

Several of those provisions, however, leave gaps or ambiguities that the final rule should address. The sections below evaluate each domain and identify where the proposed approach is sound, where it needs clarification, and where the rule should address issues that will matter in practice.

1. Emergency Planning Relative to the Consequence-Based Safety Screen

Part 57 does not define an emergency planning zone (EPZ) for eligible facilities because if an applicant has demonstrated under § 57.25(a) that a bounding accident would not exceed 1 rem TEDE at the unrestricted area boundary, the safety basis for a traditional plume-exposure-pathway EPZ does not exist. This is consistent with existing 10 CFR 50.160 and § 50.33(g)(2)(i)(A), which allow a site-bounded EPZ if the public dose, as defined in § 20.1003, is not projected to exceed 10 mSv (1 rem) total effective dose equivalent (TEDE) over 96 hours from the release of radioactive materials. Part 57's entry criteria are more conservative by requiring the same dose limit for the duration of the event.

In place of a traditional EPZ, the rule requires applicants to develop site-specific emergency plans under § 57.60(a)(8)(iv) addressing accidental releases, loss of control of radioactive material, protection of emergency workers, and coordination with offsite response organizations. The rule uses three of the four standard emergency classification levels—Notification of Unusual Event, Alert, and Site Area Emergency—and does not include General Emergency. The preamble explains that an EPZ is most useful for implementing precautionary actions in rapidly progressing incidents involving multiple jurisdictions, and that the characteristics of Part 57-eligible facilities make planning for such precautionary actions unnecessary.

BTI supports the proposed approach. The rule correctly preserves the requirement for a site-specific emergency plan while eliminating EPZ requirements that would be disproportionate to demonstrated consequences. The three-tier classification structure is appropriate for the same reason: the General Emergency classification was designed for scenarios involving the potential for acute doses across a wide area, which Part 57-eligible facilities are not postulated to produce.

There is, however, a major inconsistency in policy decisions around emergency planning in recent rulemakings, most notably the Regulatory Framework For Fusion Machines.[20] BTI commented on that docket and noted the contradiction in emergency plan requirements between Part 57.[21] The fusion machine proposed rule does not strictly require an emergency plan unless 1 rem is exceeded during an accident, stating, "the NRC would continue to require applicants to determine if the maximum dose to a person offsite could exceed 1 rem (10 mSv), and if so, to provide an emergency plan for offsite protection of the public." The Part 57 rule, which establishes a firm 1-rem limit at the site boundary, requires an emergency plan regardless of the potential offsite dose. This is an inconsistency between rules and a contradictory application of regulatory requirements for the same consequence value (1 rem). There is no established basis for the NRC to maintain the contradiction between the Part 57 and fusion machine treatments in the final rule.

This concern bleeds into further gaps in emergency planning in the proposed rule. Part 57 and NUREG-2271 do not explain what the NRC expects from site-specific emergency plans across the range of deployment scenarios that Part 57 is designed to support. Emergency planning for a remote off-grid installation, a shared industrial site, a defense facility, or an operator-independent deployment with no permanent on-site staff may look materially different from conventional emergency planning. The rule requires coordination with off-site response organizations but does not identify which elements of the emergency plan may be standardized through a manufacturing license, standard design approval, or referenced operational program, and which require site-specific treatment at each deployment. This will require future guidance or modification of existing guidance. At a minimum, off-site communication and monitoring capabilities should be required.

[20] U.S. Nuclear Regulatory Commission, "Regulatory Framework for Fusion Machines," Federal Register 91, no. 38 (February 26, 2026): 9476—9498, https://www.federalregister.gov/d/2026-03865.

[21] The Breakthrough Institute, Comment on Proposed Regulatory Framework for Fusion Machines" Docket ID NRC-2023-0071, Comment ID NRC-2023-0071-0107, May 27, 2026.

The final rule also does not explain how emergency planning will work for multi-unit deployments, large designated areas, remote sites, industrial sites, military or federal sites, and deployments that rely on a manufactured reactor or standardized operational program.

The final rule should make clear that the absence of a defined EPZ is a consequence of the § 57.25(a) eligibility demonstration, and that subsequent emergency planning requirements are calibrated to that requirement. Emergency planning requirements should not be reimported through guidance, staff review practice, or adjacent rule parts without explaining why those requirements remain necessary for a reactor whose accident consequences are already bounded. If site-specific conditions—proximity to hazardous industrial facilities, unusual meteorological conditions, shared-site configurations— create hazards not captured in the eligibility demonstration, the emergency plan should address them, but the baseline should remain scaled to the consequence.

2. Align Physical Security with Risk

Part 57 establishes a two-track security structure. Under § 57.60(a)(8)(v)(A)(3), applicants must demonstrate whether a Design Basis Threat (DBT)-initiated event could produce offsite doses exceeding the § 50.34(a)(1)(ii)(D) reference values even if all mitigation and recovery actions were unavailable. Applicants whose design satisfies that criterion are not required to establish a full operational physical security program. Applicants that cannot satisfy it are subject to proposed Subpart J, which establishes performance-based requirements for intrusion detection and assessment, security communications, response capabilities, coordination with local law enforcement, target set identification, cybersecurity, insider mitigation, and individual and vehicle search programs. Subpart J does not prescribe how these elements are met; it requires reasonable assurance that a DBT-initiated event would result in offsite doses below the § 50.34(a)(1)(ii)(D) values. Force-on-force exercises are not required for Part 57 facilities, given their low risk profiles.

The two-track structure is well-designed. It extends the consequence-based logic of the eligibility criteria into the security domain: a reactor whose design bounds the consequences of a security-initiated event is regulated differently from one that requires operational security measures to keep consequences in bounds. The use of "reasonable assurance" language in Subpart J is consistent with the Atomic Energy Act's (AEA) mandate and should be preserved in the final rule. The § 57.30(f) security-by-design attribute supports this structure by crediting engineering and physical protection features in the applicant's licensing basis, which is appropriate.

Several issues need clarification:

First, the reference values in § 57.325(b)(2) should be drawn from the technology-inclusive physical security requirements of § 73.100(b)(3) rather than from § 50.34(a)(1)(ii)(D). The § 73.100(b)(3) value, adopted from § 53.210, reflects the modernized, technology-inclusive security framework and aligns Part 57 with Part 53. Conforming § 57.325(b)(2) to § 73.100 would also enable a design that established its physical-security basis under Part 53 to carry that basis into Part 57 for nth-of-a-kind deployment without re-pegging to a light-water-reactor-era reference value, consistent with the FOAK-to-NOAK transferability the rule should support.

Second, the rule does not explain how § 57.30(f) security- by-design features are evaluated in the § 57.60(a)(8)(v)(A)(3) assessment or how they interact with Subpart J requirements. Which features are credited, how their reliability and durability are demonstrated, and whether security-by-design alone can satisfy the consequence criterion or must be combined with operational measures are questions the rule leaves to case-by-case staff judgment. That gap will create uncertainty for applicants designing for the security-by-design attribute.

Third, the rule does not address the transition of security responsibilities during transport and redeployment. Proposed § 57.197(e) references Parts 71 and 73 for transport and security of manufactured reactors in transit, but the rule does not explain how security coverage transitions between the manufacturing licensee, the transport phase, and the operating licensee at the deployment site. For Part 57 designs that may be recovered and redeployed, this gap is significant: the applicable law enforcement jurisdiction, the security plan, and the coordination arrangements will all change with the deployment location, and the rule does not identify who is responsible for those arrangements during the transition.

Relatedly, a gap remains for the theft and diversion of special nuclear material at remote, unattended, and remotely operated sites, which neither the security entry criterion nor the § 57.60(a)(8)(v)(A)(3) assessment reaches. Delay-and-denial features sized to meet the § 73.100(b)(3) reference values against sabotage may still leave intact fuel removable by a determined adversary during the window before law enforcement can respond at a remote site. The final rule should require the residual materials-protection posture of remote and unattended facilities to be evaluated as a function of site controllability, including expected response capability, so that these deployments receive graded materials-protection treatment.

Lastly, as with safety, the rule does not specify how the consequence determinations in § 57.60(a)(8)(v)(A)(3) treat co-located units. A reactor whose individual source term meets the § 73.100(b)(3) reference values may, when co-located with others within a single attack footprint, present an aggregate target. Since these determinations now govern whether a facility meets a possible security entry criterion and whether it receives the further relief of operating without an operational security program, the final rule should state whether the assessment is conducted per unit or per site and define the co-location distance and common-cause assumptions that govern aggregate consideration.

3. Cybersecurity

Under § 57.60(a)(8)(v)(B), Part 57 requires licensees to implement a cybersecurity program under either existing § 73.54 or new proposed § 73.110. Section 73.54 is the traditional prescriptive cybersecurity framework used by the current large LWR fleet. Section 73.110 is a new consequence-based and graded framework developed specifically to accommodate the diverse technologies and risk profiles of reactors licensed under Part 57. Under § 73.110, a licensee must provide reasonable assurance that digital systems are protected against cyberattacks capable of adversely impacting safety and security functions that prevent off-site doses from exceeding the § 50.34 reference values or compromising the physical security requirements. The level of protection required is graded by consequence: if the analysis shows that a cyberattack cannot produce doses exceeding those thresholds even with digital assets compromised, only a narrow set of baseline requirements applies.

NUREG-2271 Section 11.8 maps these requirements to Part 57's operational architectures explicitly. For remote monitoring and operation, the application must describe the defensive computer security architecture, access controls, and connectivity protocols, and must demonstrate how the confidentiality, integrity, and availability of data at the remote monitoring or operating location will be protected. Proposed § 57.60(a)(1)(v) separately requires design features for remote monitoring to protect sensitive plant data that could be used to aid a physical or cyber attack. For autonomous functions, the cybersecurity program must ensure that a cyberattack cannot adversely impact the digital assets necessary for safety and security functions. For fleet deployment, the rule defines cybersecurity programs as programmatic controls that may be standardized at the corporate or institutional level and administered fleet-wide. Applicants choosing § 73.54 are directed to RG 5.71; those choosing § 73.110 are directed to proposed RG 5.96.

The cybersecurity framework is more complete than a reading of the rule text alone suggests. The § 73.110 track addresses the consequence-based logic that motivates Part 57 and maps meaningfully to remote, autonomous, and fleet-wide operational architectures through NUREG-2271. The graded approach—which limits requirements to a baseline set where digital assets cannot produce dose-exceeding consequences even if compromised—is consistent with the rule's broader proportionality principle. The fleet-level standardization of programmatic controls is also appropriate and aligns with the high-volume deployment architecture discussed in Sections II and V.

Two issues nevertheless warrant attention in the final rule.

First, the rule gives applicants a free choice between § 73.54 and § 73.110 without guidance on which track is more appropriate for which deployment model. Section 73.54 was designed for large light-water reactors with staffed control rooms and physical separation of safety-critical systems. Its application to operator-independent, remotely monitored, or autonomous Part 57 facilities may produce either over- inclusive requirements not calibrated to the actual risk profile or an awkward mapping exercise that consumes applicant and staff resources without corresponding safety benefit. The NRC should provide updated guidance when finalizing NUREG-2271 on when § 73.54 is appropriate for Part 57-eligible designs and when § 73.110 better reflects the facility's consequence and operational profile. For most Part 57 deployment models, the § 73.110 track appears better suited, and the NRC should say so clearly rather than leaving the choice unguided.

Second, the rule addresses the structure of fleet-level cybersecurity governance but not the change management process for post-deployment updates. Standardizing cybersecurity programs as programmatic controls at the corporate or institutional level reduces duplicative licensing review across fleet deployments. But it does not resolve how software updates, security patches, and configuration changes that affect cybersecurity-relevant digital assets are managed after initial deployment across a fleet of licensed reactors. If a cybersecurity vulnerability is discovered in a common software component across a deployed fleet, the mechanism for remediation—and whether it requires individual license amendments, a fleet-level amendment, or can be managed through the standardized programmatic control without licensing action—should be defined in the rule or guidance. The NRC should also clarify whether cybersecurity programmatic controls, once standardized and reviewed in a FOAK or fleet-level proceeding, are eligible for generic finality under § 57.142(e) such that subsequent NOAK applicants can reference them without full re-review.

4. Material Control, Accounting, and Safeguards

The MC&A and safeguards framework is appropriately structured. Applying Part 74 Subpart B as the baseline rather than the full fabrication and enrichment facility requirements reflects the actual material control profile of Part 57 reactors. The 6-month physical inventory cycle for unattended HALEU facilities is a reasonable response to the diversion risk posed by high-enrichment fuel in remotely operated configurations. The § 75.4 amendment correctly brings Part 57 licensees into the IAEA safeguards framework at the one-effective-kilogram threshold. The manufacturing interface and criticality prevention provisions address a genuine safety and security gap for factory-fueled designs. However, there are issues that warrant clarification in the final rule.

First, the preamble does not explain the technical basis for the 6-month physical inventory cycle. The stated rationale—that increased frequency provides additional assurance against diversion of higher-enriched material—is a policy conclusion rather than a technically grounded derivation. The NRC should explain in the preamble how the 6-month interval was determined in relation to the specific material control risks of HALEU fuel in unattended facilities, including the role of remote monitoring data, access control systems, and security-by-design features in maintaining material accountability between physical inventories. Without that explanation, the threshold appears administrative rather than technically justified, and applicants will lack a basis for proposing alternative intervals where facility-specific conditions might support them.

Second, the NMMSS location tracking requirement addresses the reporting obligation for geographic moves but does not identify who holds reporting responsibility during transport. Part 57 reactors may transition through at least three possession states— manufacturing licensee, transport (under Part 71 and Part 73), and operating licensee at the deployment site. The proposed rule does not specify whether the ML holder, the operating licensee, or the transportation licensee bears the NMMSS reporting obligation during each transition, or how possession handoffs are documented. For redeployable designs that may move through this sequence multiple times, that ambiguity compounds across the facility's operational life.

Third, the SGI separation requirement—that Safeguards Information must be submitted separately from the FSAR—creates a tension with the rule's generic finality provisions. For Part 57 applicants pursuing standardized designs and fleet deployment, security- sensitive design information that supports a finality determination under § 57.142(e) may fall within the SGI or SGI-M categories. If that information cannot appear in the FSAR, it is not clear whether it can be incorporated by reference in subsequent NOAK applications in a way that preserves finality. The NRC should explain in the preamble or in implementation guidance how security-sensitive design information that has received generic finality is handled in NOAK applications, given the FSAR separation requirement.

Fourth, the § 73.58 safety-security interface requirement applies to all Part 57 operating licensees, but its application to standardized fleet deployments is not addressed. For high-volume NOAK deployments, configuration changes based on standardized approved designs may trigger § 73.58 assessments that are substantively identical across the fleet. The rule should clarify whether fleet-level § 73.58 assessments for standardized configuration changes can be conducted once and referenced across NOAK applications, or whether each deployment site requires independent assessment.

Table 8. Recommendations – Emergency Planning, Security, and Safeguards

Affected Section	Recommendation
Preamble § II.F; § 57.60(a)(8)(iv)	State that the absence of a defined EPZ flows from the § 57.25(a) eligibility demonstration and that emergency planning requirements are calibrated to that showing.
§ 57.60(a)(8)(iv); implementation guidance	Resolve the inconsistency in emergency plan requirements across rulemakings. Clarify the scope and content of site-specific emergency plans for remote, operator-independent, and multi-unit deployments; identify which elements may be standardized or afforded finality.
§ 57.30(f); § 57.60(a)(8)(v)(A)(3); Subpart J; implementation guidance	Clarify how § 57.30(f) security-by-design features are evaluated and credited in the § 57.60(a)(8)(v)(A)(3) consequence assessment and how they interact with Subpart J requirements.
§ 57.197(e); Subpart J; conforming amendments to Parts 71, 73	Explain how physical security responsibilities transition during transport, initial deployment, and redeployment, including how offsite law enforcement coordination applies when deployment location changes.
§ 57.60(a)(8)(v)(B); RG 5.96; implementation guidance	Provide guidance on when § 73.54 versus § 73.110 is the more appropriate cybersecurity track for Part 57 deployment models; indicate that § 73.110 is the better fit for most remote, operator-independent, and autonomous configurations.
§ 57.60(a)(8)(v)(B); § 57.142(e); implementation guidance	Clarify how post-deployment software updates and configuration changes affecting cybersecurity-relevant digital assets are managed across a fleet, and whether standardized cybersecurity programmatic controls are eligible for generic finality under § 57.142(e).
§ 57.360(b); Preamble § II.N	Explain the technical basis for the 6-month physical inventory cycle in § 57.360(b), connecting it to specific material control risks of HALEU fuel in unattended facilities.
§ 57.360; conforming amendments to Part 74	Clarify which entity holds NMMSS reporting responsibility during each possession transition for manufactured and redeployable reactors.

Affected Section	Recommendation
§ 57.142(e); § 57.360; §§ 73.21—73.23	Explain how security-sensitive design information subject to generic finality under § 57.142(e) is handled in NOAK applications, given the FSAR/SGI separation requirement.
§ 73.58; implementation guidance	Clarify whether § 73.58 safety-security interface assessments for standardized configuration changes may be conducted at the fleet level and referenced across NOAK applications.
§ 57.325(b)(2); § 73.100(b)(3); cf. § 53.210; conforming amendments to Part 73	Conform the physical-security reference values in § 57.325(b)(2) to the technology-inclusive requirements of § 73.100(b)(3) (adopted from § 53.210) rather than § 50.34(a)(1)(ii)(D), aligning Part 57 with the modernized security framework and Part 53 and enabling a design's physical-security basis to transfer from Part 53 into Part 57 without referencing to a light-water-reactor-era reference value.
§ 57.25 (new subsection (c)); § 57.30(f); § 57.325	Add § 57.25(c) as a physical-security eligibility criterion parallel to § 57.25(a)—(b): a reactor is eligible for Part 57 only if the § 73.100(b)(3) reference values can be met against the design-basis threat through inherent design characteristics, § 57.30(f) security-by-design features, and standardized programmatic delay-and-denial measures, without crediting a site-specific armed interdiction-and-neutralization response requiring case-by-case adversary-pathway and response-timeline analysis. A reactor that cannot meet this criterion is not eligible for licensing under Part 57.
§ 57.60(a)(8)(v)(A)(3); § 57.325	Reposition the unmitigated assessment in § 57.60(a)(8)(v)(A)(3) as a post-entry, further-streamlining determination rather than the security entry test: a design admitted under § 57.25(c) that also meets the reference values with no credited mitigation or operator action operates without an operational physical security program, while an admitted design relying on standardized programmatic measures implements them under § 57.325.

IX. Environmental Review and Siting

Environmental review and siting are central to whether Part 57 can support high-volume deployment. A reactor safety pathway that enables standardized review will not succeed if environmental review remains entirely bespoke, site-by-site, and disconnected from the consequence-based safety screen. At the same time, the NRC must preserve legally durable NEPA compliance and avoid overstating the circumstances in which categorical treatment or standardized environmental information is appropriate.

The final rule should integrate Part 57 with Part 51 and the agency's broader NEPA modernization work. The staff should also consider whether these recommendations are more appropriate to be included in Part 51, where they would be generically available across the NRC's regulatory framework, including Part 57. Reactors that satisfy Part 57's consequence-based eligibility criteria may have environmental profiles that differ materially from large light-water reactors: smaller footprints, lower water use, lower offsite accident consequences, modular construction, factory fabrication, deployment at industrial or previously disturbed sites, and limited emergency planning needs. Those features should be reflected in the environmental-review framework, and the NRC should explain clearly which issues may be resolved generically, which may be standardized through site parameter envelopes, and which must remain site-specific.

1. Use Categorical Exclusions Where the Record Supports Them

The NRC should use categorical exclusions (CatEx) for Part 57 activities that normally do not have significant environmental effects, but the final rule should structure those exclusions around bounded environmental impacts rather than broad technology classes alone. Categorical exclusions can be an important tool for high-volume deployment, especially for administrative approvals, minor or previously disturbed site-preparation activities, limited construction-related activities, manufactured-reactor approvals, or deployments at sites where relevant impacts are already bounded. But their legal and practical value depends on whether the exclusion is tied to plant, site, and activity-based criteria.

Under proposed Subpart K (§ 57.350), the NRC determines that the issuance of initial or renewed licenses for microreactors and reactors with comparable risk profiles normally does not significantly affect the human environment. To qualify for this categorical exclusion, an application must demonstrate compliance with several site-specific and design criteria: the reactor's plant parameter envelope (PPE) and site parameter envelope (SPE) values must fall within those established in the Generic Environmental Impact Statement for Licensing of New Nuclear Reactors (NR GEIS),[22] location on previously disturbed land, absence of direct surface-water or groundwater withdrawals or discharges for cooling, de minimis air emissions, and consistency with applicable state and local requirements. This type of bounded, criteria-based approach is more durable than an exclusion based solely on reactor size, technology class, or nominal power level.

SPEs are especially important to the proposed categorical-exclusion framework. Under proposed § 57.350, a Part 57 applicant seeking to rely on a categorical exclusion would need to show that the proposed action falls within the environmental PPE and SPE analyzed in the NR GEIS.[23] Where the reactor design and proposed site remain within those envelopes, the NRC may be able to conclude that certain previously analyzed environmental issues do not require additional site-specific review. That approach can reduce duplicative review and support repeat deployment, provided the underlying envelope assumptions are clear, conservative enough to be relied upon, and tied to identifiable environmental effects.

As proposed, the categorical exclusion framework is in conflict with itself. § 57.350(b)(1) requires the applicant to meet the PPE and SPE of Part 51, Appendix C, Table C-1. Proposed NUREG-2271 is more flexible than Part 51, allowing applicants to substitute some values in Table C-1 with the values from NUREG-2271 Table E-1. The conflict is that some of the values in NUREG-2271 E-1 exceed those in Part 51, Appendix C, Table C-1, so an applicant would have to request an exemption in order to use the guidance for a Part 57 license. However, Page 16-5 of draft NUREG-2271 states that an applicant cannot request an exemption from Part 57 to be eligible for a

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[22] *Generic Environmental Impact Statement for Licensing of New Nuclear Reactors*, 91 Fed. Reg. 22,394 (final Apr. 24, 2026).

[23] The Breakthrough Institute, Comment on NUREG-2249, "Generic Environmental Impact Statement for Licensing of New Nuclear Reactors," Docket ID NRC-2020-0101, ML24355A170.

CatEx.[24] The NRC should resolve this conflict by revising §57.350(b)(1) to include values from NUREG 2271 Table E-1 when appropriate.

The NRC should also preserve flexibility to use categorical exclusions where they provide real value. The agency should avoid defining categorical exclusions so narrowly that they become practically unavailable, especially for repeat deployments, manufactured-reactor approvals, limited site-preparation actions, and projects at previously disturbed or otherwise bounded sites. At the same time, the NRC should not rely on categorical exclusions where material site-specific impacts are reasonably foreseeable or where the proposed activity falls outside the assumptions supporting the exclusion.

§ 57.350(b)(2)(i) requires the site to be within a previously disturbed area to be eligible for categorical exclusion. The definition of "previously disturbed area" conflates "previously disturbed" with "no ESA habitat and no potential for historic/cultural resources," when those are separate determinations. A heavily disturbed industrial site might still have subsurface historic resource potential, and a pristine greenfield might be entirely devoid of such resources. The definition creates a binary eligibility gate rather than a sensitive-resource screen calibrated to actual impacts. This restriction is unnecessarily narrow and not environmentally justified for reactors whose physical footprint is a fraction of large LWRs.

The proposed rule also bars the use of a CatEx for facilities using surface water or groundwater for cooling, even though the vast majority of previous NRC environmental reviews, including reviews for much larger facilities than anticipated under Part 57, have found cooling system effects to be small or insignificant. As above, a screening process should be made available to avoid penalizing a design choice (i.e., water cooling) rather than deterministically eliminating that option.

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[24] *This creates another inconsistency at the policy level. The categorical exclusion framework for Part 57 is analogous to the Licensing Modernization Project (LMP) methodology designed to exist within the Part 50 and 52 licensing frameworks. LMP provides a systematic, reproducible process based in guidance for reactor developers to use as a pathway through regulation. LMP is also considered more strict than the rule in terms of its depth and breadth of safety analysis, but is nonetheless a viable pathway and an applicant does not have to request an exemption to use LMP. The difference is that the Commission approved the LMP methodology in policy as a reasonable approach for establishing the licensing basis within the existing regulatory framework.*

§ 57.350(b)(2)(iv) requires the licensed activity to be "in accordance with applicable state and local requirements (such as land use planning, zoning requirements, and coastal zone management)." While it is reasonable for the NRC to establish an understanding of the applicant's plans for non-NRC permits, it is not within the NRC's regulatory purview to require compliance with non-NRC permits as a requirement of the NRC license. The NRC should remove the provisions in § 57.350(b)(2)(iv).

The final rule should make this CatEx framework transparent, predictable, and administrable. The NRC should identify the categories of Part 57 actions expected to qualify, the environmental assumptions that support each category, and the extraordinary circumstances that would require additional review. The NRC should also clarify what information an applicant must submit to justify reliance on a categorical exclusion and what conditions would disqualify an action from categorical treatment.

2. Early Site Permitting

The rule does not provide a way to obtain approval of siting analysis independently of a specific reactor application and to reuse it across deployments. A Part 57-compatible early site permit (ESP) would provide this function. Pre-approval of an area would provide efficiency and enable high-volume licensing when an area is known, but a specific design has not been selected. That mirrors the purpose of an ESP in Part 52 and Part 53.

The proposed rule resolves the analytical core of an envelope-based siting approach: § 57.16(b)(2) treats as resolved those matters decided in a referenced Part 52 early site permit proceeding, § 57.18(b) authorizes incorporation by reference of approvals issued under Parts 50 and 52, and § 57.60(d) allows an applicant to designate one or more large geographic areas, identify unsuitable areas, bound environmental and siting issues at the area level, and confirm a specific location within the approved bounds before construction. The site-parameter-envelope concept in draft NUREG-2271, aligned to the New Reactor GEIS for categorical exclusions, supplies the technical method.

Two changes would close that gap. First, the referenceability provisions should reach Part 53 and should name early site permits expressly. Sections 57.16(b)(2) and 57.18(b) currently extend only to approvals issued under Parts 50 and 52, and § 57.18(b) does not list early site permits among the approvals it covers, even though Part 53 established an equivalent early site permit process in Subpart H. BTI develops this transferability recommendation in Section III. Due to the more restrictive environmental limits imposed in Part 57, a site that has received an ESP under Part 52 or Part 53 should be able to carry those findings into a Part 57 application without repeating the siting review. Second, the NRC should provide a standalone site approval that decouples the siting review from the reactor-specific application.

As proposed, § 57.60(d) is available only within a joint application for a construction permit and associated operating licenses, with the area's life-cycle impacts analyzed at the construction-permit stage and specific locations selected later under that same authorization. That structure integrates siting into a single reactor application; it does not produce a severable site approval that a later or different deployment application can reference. To support repeated deployment, the NRC should allow the § 57.60(d) area-siting analysis to be issued as a standalone, technology-neutral, bankable approval, using the same area-designation methodology, site-parameter envelopes, and unsuitable-area screening the rule already defines, so that the siting review is performed once and reused rather than repeated in each joint application.

The NRC should structure that standalone approval in stages. If the site approval is too abstract, it will not sufficiently scope the ESP and may invite later disputes over whether the selected location was actually analyzed; if it requires full location-level analysis for every possible site within a designated area, it will be too burdensome to support repeated deployment. The workable middle is to resolve at the area level the issues that can be bounded there, identify the issues that remain genuinely location-specific, require confirmation that the selected location falls within the approved bounds before construction or operation, and foreclose relitigation of matters already resolved within the envelope. The NRC already has experience with ESP applications that are technology-neutral and bounded by PPE and SPE characteristics. A site approval built this way can be performed once and referenced across many deployments, which is the high-volume purpose it should serve.

3. Clarify Environmental Review of Large Geographic Areas and Multiple Deployment Sites

The NRC should clarify how Part 57 environmental review will operate for applications covering multiple specific sites, large designated geographic areas, co-located reactors, deployments at host facilities, and multi-unit deployments. Part 57 may be used for deployments at industrial facilities, data centers, mining sites, defense or other federal sites, brownfields, existing nuclear sites, or other host locations where the reactor is only one element of a larger facility. These deployment models can support efficient siting, make use of existing infrastructure, and expand the set of commercially useful reactor applications. But the final rule should define the environmental-review consequences of these pathways with enough precision to provide usable finality without making the review either too abstract to be legally durable or too burdensome to support deployment.

Under § 57.18(a)(2), the proposed rule allows an applicant to seek authorization for one or more reactors at multiple specific sites, provided the application includes the necessary site evaluation and environmental information for each location. That pathway is relatively straightforward, where the applicant can identify each site in advance and demonstrate that each site is bounded by the site parameter envelope for the reactor design. In that circumstance, the NRC should clarify which environmental findings may be made generically for the design, which may be made for all listed sites collectively, and which must be made separately for each site.

The broader and more novel issue is how environmental review should work for a large designated area (§ 57.60(d)) when the exact final deployment location has not yet been selected. A broad-area review could be useful where the applicant can characterize the full area, identify unsuitable subareas, and demonstrate that any later-selected location will remain within the approved site parameter envelope and environmental assumptions. But the final rule should specify the level of information required at the initial application stage, including mapping, site-evaluation factors, environmental constraints, excluded areas, and procedures for confirming the suitability of the final location before construction begins. The same clarification is needed for cumulative impacts, especially where an application covers multiple reactors, multiple potential locations, or phased deployments over time. The environmental review should evaluate the combined impacts of the authorized deployment, not merely the incremental effects of a single unit considered in isolation.

The NRC should identify which environmental issues can be bounded across the entire designated area and which issues require later site-specific confirmation. Some issues may be susceptible to bounding analysis if the applicant can define conservative area-wide assumptions, such as maximum land disturbance, bounding meteorological and hydrologic values, transportation assumptions, or cumulative deployment limits. Other issues may be difficult to resolve before the specific location is known, including cultural resources, endangered species, wetlands, water resources, environmental justice, local land use, transportation routes, emergency-planning interfaces, and certain cumulative impacts. The final rule should make clear how those issues will be handled without reopening matters that were already bound and resolved.

For multi-unit deployments, the final rule should require the environmental analysis to account for the total effects of all authorized units at a site or within a region. That includes combined land use, water use, emissions, waste management, transportation, radiological impacts, and construction effects where those impacts are relevant. If reactors share infrastructure or if construction, operation, and decommissioning activities for one unit could affect other units at the same site, the applicant should identify those interactions and show that the combined effects remain within the approved environmental and safety assumptions.

Where reactors are co-located with industrial facilities or deployed as part of a multi-unit site, the environmental analysis should account for the combined effects of the authorized nuclear deployment and should explain how those effects interact with existing site conditions. Existing permits, prior site characterization, previously disturbed land, or brownfield status may support a more efficient review, but they should not substitute for identifying which environmental assumptions are actually bounded. The NRC should explain when existing information may be relied upon, what supplemental information is required, and what site changes would trigger additional review.

The NRC should preserve the usefulness of these flexible pathways by avoiding two opposite errors. A broad-area review that is too vague will not provide durable finality and may invite later disputes over whether the selected site was actually analyzed. A broad-area review that requires full site-level analysis for every possible location within the area will be too burdensome to serve its intended purpose. The final rule should instead adopt a staged approach: resolve issues that can be bounded at the design, site-envelope, or area level; require clear procedures for later site-specific confirmation; and identify the circumstances that would require supplemental environmental review.

4. Allow Use of Existing Site Characterization Information

Part 57 should allow applicants to rely on existing site characterization information where that information is reliable, relevant, and sufficient for the licensing decision. Requiring applicants to recreate information that is already current and decision-relevant would add cost and delay without improving the quality of NRC's environmental review.

The final rule should clarify what types of existing information may be used and what showing an applicant must make before relying on it. Potentially relevant sources may include prior NEPA documents, NRC licensing records, environmental monitoring data, geotechnical investigations, hydrology and flood studies, meteorological data, cultural-resource surveys, endangered-species reviews, industrial site permits, state and federal agency datasets, and information developed for adjacent or co-located facilities. The NRC should not require new site investigations where existing information adequately characterizes the relevant site conditions, but it should require applicants to explain why the information remains representative of the current and reasonably foreseeable site conditions.

A rigid expectation for one to two years of onsite meteorological tower data would be poorly matched to Part 57's purpose.[25] The question should not be whether data were collected from an onsite tower, but whether the applicant has provided sufficient technically justified meteorological information to support the licensing finding. The NRC should clarify in guidance when offsite or regional data, supplemented by site-specific confirmatory information as needed, is acceptable for Part 57 applications.

[25] Adam Stein and Spencer Toohill, The Breakthrough Institute, *Maximizing the Use of Available Weather Data to Accelerate Advanced Nuclear Deployment*, https://thebreakthrough.org/issues/energy/maximizing-the-use-of-available-weather-data-to-accelerate-advanced-nuclear-deployment.

BTI applauds Part 57's departure from the deterministic population density siting requirements under 10 CFR Part 100.[26] For reactors that have already demonstrated under § 57.25(a) that a bounding accident would not exceed 1 rem TEDE at the site boundary, a rigid population density exclusion zone is not necessary to protect public health and safety. The consequence-based eligibility screen is the appropriate tool for right-sizing siting requirements, and Part 57 is correct not to replicate Part 100's deterministic approach for this reactor class.

The NRC should define the basic acceptance criteria for existing information. Applicants should identify the source of the data, the methods used to collect and analyze it, the age of the information, the geographic relationship between the data and the proposed site footprint, any quality-assurance limitations, and any material changes since the information was generated. Where existing information is incomplete or outdated, the applicant should identify the gap and provide supplemental information commensurate with the significance of the issue.

The final rule should also recognize that the appropriate level of new data collection may vary with deployment duration, site conditions, and the environmental issue being evaluated. A short-term deployment of a lower-risk reactor (e.g., NSC-1, NSC-2, or NSC-3) at a previously characterized site may not require the same level of new field investigation as a long-term deployment at a previously undeveloped site. Similarly, offsite or regional data may be sufficient for some meteorological, hydrological, or demographic questions but insufficient for site-specific issues such as wetlands, cultural resources, endangered species, subsurface conditions, or host-site hazards. The NRC should make that graded approach explicit so that applicants understand when existing information is enough and when additional site-specific work is required.

Existing site information should also be integrated with the Part 57 envelope-based review framework. Reliable existing information may help demonstrate that a proposed site falls within an approved site parameter envelope, that plant and site parameters remain bounded by a prior generic environmental analysis, or that extraordinary circumstances are absent for purposes of a categorical exclusion. Conversely, if existing information shows that a site condition falls outside

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[26] Adam Stein and Spencer Toohill, The Breakthrough Institute, *Advanced Nuclear Needs Modern Siting Regulations*, https://thebreakthrough.org/issues/energy/advanced-nuclear-needs-modern-siting-regulations.

an approved envelope or raises a material site-specific issue, the applicant should provide additional environmental analysis rather than relying on the envelope or categorical treatment.

Table 9. Recommendations – Environmental Review and Siting

Affected Section	Recommendation
§ 57.350; Subpart K; implementation guidance	Resolve the internal conflict between the rule and guidance on categorical exclusion eligibility: the rule requires applicants to meet the PPE and SPE of Part 51, Appendix C, Table C-1, while NUREG-2271 offers more flexibility but prohibits the exemption that would be needed to use it. The NRC should align the rule and guidance so that the flexibility NUREG-2271 offers is legally accessible; clarify that SPEs define the conditions under which prior generic environmental analysis remains applicable and not a substitute for identifying which site-specific issues remain; and make the categorical exclusion framework transparent, predictable, and administrable by identifying the categories of Part 57 actions expected to qualify and the extraordinary circumstances that would require additional review.
§ 57.350(b)(2)(i)—(ii); Subpart K; implementation guidance	Replace the categorical bars on previously disturbed area eligibility and surface water or groundwater cooling with sensitive-resource screens calibrated to actual impacts; the current definitions conflate site disturbance history with ESA habitat and cultural resource determinations, and the cooling bar penalizes a design choice that prior NRC reviews have consistently found to have small or insignificant environmental effects. Develop alternative screening mechanisms that evaluate actual impacts rather than imposing deterministic eligibility gates.
§ 57.350(b)(2)(iv); Subpart K; implementation guidance	Remove the requirement that licensed activity be in accordance with applicable state and local requirements as a condition of categorical exclusion eligibility; state and local permits will often not be finalized on a timeline consistent with NRC licensing, and conditioning NRC categorical exclusion eligibility on non-NRC permit completion exceeds the NRC's regulatory jurisdiction. The NRC should remove the provisions in § 57.350(b)(2)(iv).

Affected Section	Recommendation
§ 57.16(b)(2); § 57.18(b); § 57.60(d); NUREG-2271; implementation guidance	Support the proposed envelope-based siting analysis (the § 57.60(d) large-area designation and the site-parameter envelopes in NUREG-2271 aligned to the New Reactor GEIS) and make it referenceable and reusable. Extend § 57.16(b)(2) and § 57.18(b) referenceability to Part 53 approvals and name early site permits expressly, as developed in the Part 57 and Part 53 sections. Because § 57.60(d) is available only within a joint construction-permit and operating-license application and cannot yield a severable site approval, allow the § 57.60(d) area-siting analysis to be issued as a standalone, bankable approval that later or separate deployment applications can reference, structured in stages with confirmation that the selected location falls within the approved bounds and finality against relitigation within the envelope.
§ 57.18(a)(2); § 57.60(d); implementation guidance	Clarify how environmental review applies to applications covering multiple specific sites and large designated geographic areas; specify the level of information required at the initial application stage, identify which environmental issues can be bounded area-wide versus requiring later site-specific confirmation, resolve inconsistent terminology across site boundary, unrestricted area, exclusion area, and large designated area, and require cumulative impact analysis accounting for all authorized units rather than only the incremental effects of a single unit.
§ 57.350; implementation guidance	Allow applicants to use scientifically justified offsite, regional, or existing meteorological data where representative of the proposed site and adequate for dose, emergency planning, and environmental analyses; clarify in guidance when offsite or regional data supplemented by site-specific confirmatory information is acceptable for Part 57 applications rather than defaulting to an expectation of one to two years of onsite meteorological tower data, which is inconsistent with Part 57's high-volume deployment purpose.

X. FUEL, TRANSPORTATION, STORAGE, DECOMMISSIONING, AND FINANCIAL PROTECTION

Part 57 will not function as a high-volume licensing pathway if fuel, transportation, storage, decommissioning, and financial protection requirements are left to be resolved after the reactor-safety review. For many microreactors and NRC-defined small modular reactors, the deployment model may depend on factory fueling, sealed cores, transportable reactor modules, centralized refurbishment, limited on-site fuel handling, or standardized decommissioning plans. Those features are central to the business model and the safety case, not peripheral implementation details.

The final rule should identify how adjacent approvals under Parts 70, 71, 72, and 140 will interact with Part 57. The NRC should distinguish which issues can be resolved in the Part 57 application, which may be resolved through a manufacturing license or referenced approval, which require separate licensing, and which must remain site-specific. Without that sequencing, applicants may satisfy Part 57's entry criteria but still face hidden critical paths that prevent deployment.

1. Clarify Part 57 with Part 70 Special Nuclear Material Requirements

The final rule should specify which special nuclear material activities are authorized under a Part 57 license and which require a separate Part 70 license or approval, and should provide for the manufacturing license to resolve standardized special nuclear material possession, sealed-core verification, and material-control questions once for a reactor design so that deployment-site applications can incorporate that resolution by reference rather than repeat it. Many Part 57 deployment models seek to minimize or eliminate routine on-site fuel handling through factory fueling, sealed cores, replaceable modules, or centralized fueling and refurbishment, and those approaches raise questions of possession, transfer, storage, safeguards, and accountability that the reactor-safety review does not by itself resolve.

The proposed rule addresses some of these questions but does not lay out the full special nuclear material interface for the deployment-site license. What the rule does not yet make explicit is which possession, receipt, loading, inspection, sealed-core verification, damaged-fuel, and return-to-manufacturer activities fall within the Part 57 license itself and which require separate Part 70 authorization.

Many of these considerations are well-suited to be resolved at the manufacturing-license level. The final rule should provide that the manufacturing license can establish finality for standardized special nuclear material matters (such as fuel form, enrichment level, quality assurance, and more), with deployment-site Part 57 applications limited to site-specific and applicant-specific possession questions. That approach would extend to the fuel cycle the same product-and-fleet standardization that the proposed rule applies to the reactor design.

Possession limits and material control and accounting should be scaled to the actual sealed-core inventory rather than to assumptions developed for fuel-cycle facilities. Criticality safety should be addressed as a distinct consequence pathway rather than folded into the reactor accident analysis. Because material control and accounting for this reactor class is addressed under the proposed Part 74 conforming provisions discussed in the emergency planning, security, and safeguards section, the final rule should make the handoff between Part 70 possession authorization and Part 74 accountability explicit so that sealed-core item accountancy does not fall between the two.

The allocation of activities between the Part 57 license and a separate Part 70 authorization should be stated in the preamble and implementation guidance, and the manufacturing-license finality for standardized special nuclear material matters should be reflected in the regulatory text defining manufacturing-license scope.

2. Coordinate Part 71 Transportation Approvals with Manufactured and Transportable Reactors

The final rule should clarify the conditions under which the proposed risk-informed transportation methodology may be used in place of the prescriptive Part 71 testing and performance requirements, should specify the safety standards that govern that alternative, and should explain how transportation package approvals for repeated shipments of identical reactor modules can be resolved at the manufacturing-license level rather than repeated for each shipment. Transportation is central to manufactured and transportable reactor models in which a module is fabricated, fueled, shipped, installed, removed, refurbished, or returned, and the sequencing of transportation approvals can determine whether such a model is deployable.

BTI supports maintaining rigorous transportation safety, and the risk-informed option is sound for a sealed module whose containment, shielding, and criticality-safety performance can be demonstrated by analysis. To be administrable and durable, the alternative should be bounded. The final rule should specify when the alternative is available, what technical demonstration is required, and what dose, criticality-safety, heat-removal, shielding, and accident-condition standards govern it, together with post-transport inspection expectations.

In specifying those standards, the final rule should draw the package's radiological and fissile basis from the reactor's demonstrated bounded inventory, while recognizing that a transportation accident is a distinct exposure scenario from the reactor accident analyzed under Section 57.25(a), with a different receptor, environment, and exposure pathways. For repeated shipments of identical modules, the package design, accident-condition analysis, and associated controls can be reviewed generically. The proposed rule already routes the transport procedures for an unfueled reactor with unirradiated components to the manufacturing-license application and requires manufacturing-license holders to impose carrier-compliance obligations through transport contracts. The final rule should extend that logic by explaining how package certification for fueled or irradiated modules can attain finality at the manufacturing-license level for a standardized design.

Where the risk-informed methodology is used in place of physical package testing, the final rule should explain how the NRC will document and address route, accident, sabotage, and emergency-response considerations in the licensing record. Without defined availability conditions and governing standards, the alternative methodology could create litigation risk and uncertainty for applicants, and unresolved sequencing between package approval and the reactor license could make transportation a critical path to deployment.

3. Clarify Storage and Part 72 Interfaces

The final rule should clarify how the proposed Part 72 storage authorizations apply to transportable and return-to-manufacturer deployment models, and should ensure that storage requirements are scaled to the demonstrated decay heat and radiological inventory of the stored module rather than to assumptions developed for large spent-fuel inventories. Depending on the deployment model, irradiated fuel or a sealed module may be stored at the deployment site, removed for return to a manufacturer, or held at a centralized facility, and the regulatory treatment should make each of those pathways continuous.

The proposed rule establishes a layered storage scheme. An operating-license holder receives a general license under Part 72 for the disposition of irradiated fuel, which may be stored within the reactor or in a certified dry storage system at the operating site, and a site-specific Part 72 license is required for continued on-site storage after the operating license is terminated. A manufacturing-license holder may store irradiated fuel returned from an operating site if it holds a Part 70 license for the material and a site-specific Part 72 license for storage. This structure fits a store-on-site model, but its transition points are less clear for a transportable model in which the module does not remain on the deployment site. The final rule should explain how the general license, transportation, and manufacturing-site storage authorizations connect so that an operator can move a module from operation to transport to return without encountering an authorization gap.

Storage requirements should be proportioned to the stored module. The confinement, thermal, criticality, and shielding requirements that implement the Part 72 dose standards turn on decay heat and radionuclide inventory, and for a single sealed low-consequence module, those quantities differ substantially from the large-cask and spent-fuel-pool inventories that inform existing storage practice. Decay heat is a function of fuel mass, burnup, and cooling time, which is the same physical relationship that makes a fixed heavy-metal mass limit an imprecise proxy for safety consequence, as discussed in the eligibility analysis. Scaling storage requirements to the module's demonstrated decay heat and inventory would keep this part of the lifecycle consistent with the consequence-based logic of the rule.

For sealed-core and transportable models, storage is closely connected to transportation and to decommissioning, and the regulatory treatment should reflect that integrated lifecycle rather than addressing storage in isolation. The proposed rule's requirement to submit an irradiated fuel management and funding plan around permanent cessation of operations should be coordinated with the return and refurbishment pathway so that the plan accounts for material that leaves the site rather than remaining in place. The final rule should avoid a result in which a low-consequence reactor can be deployed but cannot practically store, remove, or return its fuel or module, and storage should appear in the applicant-facing pathway map alongside the other lifecycle approvals.

4. Provide a Lifecycle Framework for Decommissioning, Refurbishment, and License Termination

The final rule should specify how the decommissioning and decommissioning-funding obligation is allocated, and where necessary transferred, for non-traditional deployment pathways. The proposed rule adapts its decommissioning framework from the rules for non-power production and utilization facilities and for power reactors, appropriately departing from the 60-year decommissioning provision in 10 C.F.R. § 50.82(a)(3) and allowing a decommissioning plan to be approved as part of the initial joint construction permit and operating license application. The framework's funding and termination mechanics remain anchored to the operating-site license; however, the proposed rule's storage provisions contemplate that irradiated fuel and reactor modules may leave the operating site and be held by the manufacturing license holder.

The proposed rule allows a manufacturing license holder to store irradiated fuel returned from an operating site if it holds a Part 70 license for the material and a site-specific Part 72 license for storage. That provision recognizes the return-to-manufacturer model on the storage side, but the decommissioning provisions do not appear to address how the funding obligation follows the material. Under a return or centralized-refurbishment model, the obligation is divided. The operating site retains a residual decommissioning scope, including radiological characterization of the pad and controlled area and unrestricted release under 10 C.F.R. Part 20, Subpart E,[27] while the larger radiological inventory travels with the module to a facility whose Part 70 and Part 72 authorizations govern possession and storage rather than decommissioning funding. The proposed rule does not state at what point the operating-site licensee's funding obligation is reduced or discharged, what funding assurance the receiving facility must hold for the returned inventory, or how a centralized facility's own decommissioning funding accounts for an aggregate inventory that rises and falls as modules cycle through.

[27] 10 C.F.R. Part 20, Subpart E.

The proposed rule's funding-assurance methods—namely, prepayment, external sinking funds, and surety or insurance, together with the requirement to submit an irradiated fuel management and funding plan around permanent cessation of operations—are suited to a facility that operates for a long period at a single site. That does not translate to a transportable module that may reside on a deployment site briefly before return. A sinking fund designed to accrue over a multi-decade operating life cannot assure decommissioning for a module removed after a short residence, and the funding plan's cessation trigger may arrive only after the material has already left the site. The final rule should explain which funding-assurance instrument is appropriate for short-residence transportable modules and whether a manufacturing license holder should maintain a fleet-level funding assurance scaled to the rolling inventory of deployed and returned cores.

The amount of decommissioning funding assurance should be scaled to the actual radiological inventory, contamination potential, and site footprint of the deployment rather than to assumptions derived from large, permanently sited light-water reactors. The proposed rule's departure from the 60-year provision reflects the correct recognition that a manufactured or transportable reactor presents a different lifecycle, and the funding-amount determination should follow the same consequence-based logic that supports the rest of Part 57, consistent with the principle stated in the eligibility analysis that requirements following the § 57.25(a) showing should be calibrated to that showing.

The final rule should also clarify partial site release and individual-unit license termination at multi-unit deployments. The proposed rule provides that where a plant has shared portions used by multiple reactors, the entire plant must be decommissioned before the final reactor's operating license can be terminated. For a high-volume site where units are deployed, replaced, or retired in stages over a long period, that provision could hold the license termination of an early-retired unit behind the decommissioning of the last unit on the site. The agency should explain how partial site release operates for staged multi-unit deployments so that the shared-facilities provision does not become an administrative barrier to terminating obligations for units that have completed decommissioning.

5. Part 140 Financial Protection and Indemnity

The final rule should explain the basis on which the NRC will exercise its discretion to set financial protection and indemnity requirements for Part 57 facilities, should connect that determination to the § 57.25(a) accident consequence demonstration, and should resolve how financial protection applies across modules and units at multi-unit and fleet deployments. The proposed rule states that, because microreactors may pose lower risks than the existing fleet, the NRC may reduce the amount of financial protection or indemnification fees required under the Price-Anderson Act and 10 C.F.R. Part 140. BTI supports exercising that discretion where it is grounded in demonstrated consequences, but the final rule should explain how the discretion will be applied so that determinations are consistent across applicants and adequately supported in the record.

Part 140 already establishes a graded structure that distinguishes reactors by size and consequence.[28] Many Part 57 reactors will fall below the large-reactor tier under the existing structure, so the operative question is less whether such reactors will be treated like large light-water reactors and more how the NRC will calibrate the amount within its existing discretionary tiers.

The § 57.25(a) eligibility demonstration provides the consequence metric that financial protection is intended to reflect. A reactor admitted to Part 57 has demonstrated that a bounding or credible accident keeps offsite dose below the 1 rem TEDE criterion at the unrestricted-area boundary, which is a direct measure of the public liability consequence the financial-protection amount is meant to address. The final rule should explain how that demonstrated consequence informs the financial-protection amount, so that the determination is consequence-based and consistent with the proportionality logic applied elsewhere in Part 57, rather than resting on a power-rating proxy.

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[28] *A reactor authorized to operate above one but not more than ten megawatts thermal is required to maintain $2,500,000 in financial protection under 10 C.F.R. § 140.11(a)(3); a reactor not addressed in § 140.11 is subject to the formula in § 140.12, under which the required amount is never less than $4,500,000 or more than $74,000,000; and a reactor designed for the production of electrical energy with a rated capacity of 100,000 electrical kilowatts or more is required to carry the maximum primary financial protection together with the secondary retrospective pool under § 140.11(a)(4).*

The final rule should also identify the boundary of the NRC's discretion. The maximum primary-plus-pool requirement under § 140.11(a)(4) implements the financial-protection structure of Section 170 of the AEA, and the agency's discretion to reduce the per-reactor amount operates below the statute's 100,000 electrical kilowatt threshold.[29] Because the NRC defines a small modular reactor to include modules of up to approximately 300 megawatts electrical,[30] a Part 57 module may sit at or above that statutory threshold even while qualifying under the consequence-based eligibility criteria.

The final rule should resolve how financial protection aggregates at multi-unit and fleet deployments. Sections 140.11 and 140.12 are written on a per-reactor basis, and the proposed rule contemplates multi-unit sites, large designated areas, and centralized fleet operators. The agency should clarify whether financial protection applies per reactor, per module, per plant, or per site, and whether it scales to the aggregate demonstrated consequence of a site or sums per unit. This is the same unit-of-measurement question the proposed rule leaves open for the fuel-mass and dose criteria, and resolving it consistently across the eligibility criteria and the financial-protection requirement would reduce uncertainty for applicants, host sites, and lenders evaluating multi-unit deployments.

The final rule should confirm that the construction-period financial protection required of construction permit holders who also hold a Part 70 license for fuel, set at $1,000,000 in the proposed rule, is calibrated to the material actually possessed before operation. To make the financial-protection framework administrable and durable, the agency should explain in the preamble the consequence-based basis on which it will exercise its Part 140 discretion, identify in the record the statutory boundary that the 100,000 electrical kilowatt threshold establishes, and resolve the per-reactor, per-module, per-plant, or per-site question in regulatory text. Keeping the financial-protection determination tied to the § 57.25(a) demonstration would align this requirement with the consequence-based structure on which Part 57 depends.

[29] Atomic Energy Act § 170, 42 U.S.C. § 2210. Indemnification authority under the Price-Anderson system was extended through December 31, 2065. See Further Consolidated Appropriations Act, 2024, Pub. L. No. 118-47, div. G, § 107.

[30] 1,000 megawatts thermal per module; see also 10 C.F.R. § 171.5.

6. Fuel-Cycle and Lifecycle Guidance

Applicants need to understand not only how to obtain a Part 57 construction permit, operating license, manufacturing license, or standard design approval, but also how to sequence the adjacent approvals under Parts 70, 71, 72, and 140 that make deployment possible. Guidance should identify which approvals are prerequisites for each step in the deployment lifecycle: manufacturing, fuel loading, shipment, site installation, possession of special nuclear material, operation, module removal, storage, refurbishment, decommissioning, and license termination. It should also distinguish which of these matters can attain finality at the design, manufacturing-license, or fleet level and which must be resolved for each site. For the standardized, manufactured, and transportable models that Part 57 is intended to enable at high volume, the difference between an approval resolved once and an approval that must be repeated for every unit is the difference between a usable pathway and a sequence of hidden critical paths.

Presenting these approvals in an integrated map would allow an applicant to confirm at the outset that a deployment model is feasible end-to-end, rather than discovering after the reactor-safety review that a fuel-cycle, transportation, storage, decommissioning, or financial-protection approval cannot practically be obtained on the assumed schedule. The pathway map can be provided as implementation guidance, with the sequencing relationships it reflects grounded in the final-rule preamble, so that applicants and staff can rely on them.

Table 10. Recommendations – Fuel, Transportation, Storage, Decommissioning, and Financial Protection

Affected Section	Recommendation
10 C.F.R. Part 70; proposed rule manufacturing-license provisions (Subpart D); preamble and guidance	Specify which special nuclear material activities fall within a Part 57 license and which require separate Part 70 authorization; allow the manufacturing license to establish finality for standardized possession, sealed-core verification, quality assurance, and material control; scale possession limits and accountancy to the sealed-core inventory; and make the Part 70 to Part 74 handoff explicit.

Affected Section	Recommendation
10 C.F.R. Part 71; proposed rule risk-informed transportation methodology; Subpart D	Specify when the risk-informed methodology may replace prescriptive Part 71 testing and the dose, criticality-safety, heat-removal, shielding, accident-condition, and post-transport inspection standards governing it; provide manufacturing-license-level finality for package certification of identical modules; and document route, accident, sabotage, and emergency-response considerations in the record.
10 C.F.R. Part 72, §§ 72.104, 72.106; proposed rule Part 72 general license and storage provisions	Connect the general license, transportation, and manufacturing-site storage authorizations so transportable and return-to-manufacturer models face no authorization gap; scale confinement, thermal, criticality, and shielding requirements to the module's demonstrated decay heat and inventory; and coordinate the irradiated fuel management and funding plan with the return pathway.
10 C.F.R. § 50.82(a)(3), § 50.75; 10 C.F.R. Part 20, subPart E; proposed rule decommissioning and storage provisions	Allocate and, where needed, transfer the decommissioning and funding obligation between the operating-site licensee and the manufacturing-license holder under return and centralized-refurbishment models; ensure continuous funding assurance scaled to demonstrated inventory; specify a funding instrument suited to short-residence modules; and clarify partial site release and individual-unit license termination at multi-unit sites.
10 C.F.R. §§ 140.11, 140.12; Atomic Energy Act § 170 (42 U.S.C. § 2210); proposed rule financial-protection provisions	Explain the consequence basis, tied to the § 57.25(a) demonstration, on which the NRC will exercise its Part 140 discretion; identify the statutory boundary at 100,000 electrical kilowatts; resolve whether financial protection applies per reactor, per module, per plant, or per site; and confirm the construction-period amount is calibrated to material possessed before operation.
Preamble; implementation guidance	Include fuel, transportation, storage, decommissioning, and financial-protection approvals in the applicant-facing pathway map, identifying prerequisites for each lifecycle step and which matters attain finality at the design, manufacturing-license, or fleet level versus remaining site-specific.

XI. Defects, Reporting, Enforcement, and Implementation

The final Part 57 rule should be supported by implementation, reporting, and enforcement provisions that are clear, proportionate, and compatible with standardized deployment. A high-volume licensing pathway cannot rely on case-by-case staff expectations to define basic compliance obligations after the rule is finalized. Applicants and licensees need to know how defects, noncompliance, reporting, enforcement, effective dates, guidance, and conforming amendments will operate for manufactured, standardized, remotely operated, or fleet-deployed reactors. The NRC should therefore ensure that implementation requirements preserve the rule's high-volume logic through reporting thresholds, enforcement practice, or guidance documents.

1. Clarify Defects and Noncompliance Requirements for Manufactured and Fleet-Deployed Reactors

The NRC should clarify how defects and noncompliance requirements will apply to Part 57 licensees, manufacturing license holders, suppliers, contractors, and fleet operators. For manufactured reactors, a defect may not be limited to a single site. It may affect the approved product configuration, a production process, a fleetwide software platform, a standardized operational program, or a component installed across multiple deployments. The final rule or implementation guidance should identify who has reporting responsibility, what must be reported, and when the reporting clock begins.

The NRC should distinguish defects in the manufactured product from defects in site-specific characteristics. It should also explain how defect reporting interacts with manufacturing-license finality, fleetwide change control, and corrective action programs. The requirements should be tailored to the Part 57 model. A fleetwide issue should be reported and corrected efficiently across the fleet, while minor site-specific deviations should not trigger duplicative or disproportionate reporting across unrelated deployments.

2. Right-Size Reporting and Administrative Requirements

The NRC should ensure that reporting and administrative requirements are proportionate to the consequences, operational model, and standardized licensing basis of Part 57 facilities. Reporting should focus on information that is material to safety, security, safeguards, environmental protection, licensing-basis maintenance, or NRC oversight.

The final rule should clarify how event reporting, operational reporting, cyber incident reporting, safeguards reporting, environmental reporting, and change reporting apply to Part 57 facilities. It should also explain whether reports may be submitted at the fleet level, manufacturing-license level, or site-specific license level. For standardized reactors, some information may be more efficiently reported once for the affected fleet or product configuration rather than repeated separately for every site.

The NRC should avoid creating a reporting system that is formally lighter than large-reactor requirements but practically burdensome because it lacks clear thresholds. Applicants and licensees should understand which events require immediate notification, written reports, license amendments, corrective actions, or periodic updates. Clear thresholds would improve compliance and reduce unnecessary administrative burden.

3. Preserve Enforcement Accountability While Recognizing Standardized Deployment

Part 57 should preserve strong enforcement authority, but enforcement should be adapted to standardized and distributed deployment models. The NRC should identify how it will assign responsibility among site licensees, manufacturing license holders, contractors, suppliers, remote operations centers, and fleet support organizations. Where multiple entities contribute to a Part 57 deployment, enforcement accountability should be clear before the operation begins.

The NRC should also distinguish between enforcement issues that are site-specific and those that arise from a fleetwide product, program, or process. If a violation is tied to the approved manufactured configuration or a fleetwide software platform, the enforcement response should be capable of addressing all affected deployments without forcing duplicative site-by-site proceedings. Conversely, if a violation is limited to a host site or local procedure, enforcement should remain focused on that site.

This approach would preserve accountability while supporting efficient corrective action. It would also align enforcement with the product-level and fleet-level finality concepts that Part 57 needs in order to support high-volume deployment.

4. Finalize, Maintain, and Periodically Update NUREG-2271

BTI supports the NRC's decision to issue draft implementation guidance alongside the proposed Part 57 rule. NUREG-2271 can improve predictability for both applicants and staff by explaining expected application content, review approaches, environmental information, operational programs, manufacturing-license issues, and related implementation matters. That is especially important for a new pathway intended to support standardized and high-volume deployment.

The NRC should, however, ensure that NUREG-2271 remains guidance rather than a de facto mandatory application checklist. The guidance should preserve applicant flexibility to use alternative methods where those methods are supported by a sufficient technical basis and meet the rule's requirements. If applicants believe every chapter, analysis, or information item in NUREG-2271 is effectively required, the guidance could recreate the same large-reactor-style burden that Part 57 is intended to avoid.

The NRC should also align NUREG-2271 with the final rule's purpose and scope. Any guidance on MHA/MCA methods, design criteria attributes, site characterization, staffing, cybersecurity, environmental review, manufacturing licenses, or transportation should remain tied to the Part 57 safety case and the demonstrated consequences of the facility.

Finally, the NRC should commit to maintaining and updating NUREG-2271 after early applications and deployments. If experience shows that portions of the guidance are unnecessary, duplicative, or inconsistent with high-volume licensing, the NRC should revise the guidance promptly. A new licensing pathway should be supported by adaptive guidance that helps Part 57 work in practice, not by static guidance that hardens early assumptions before deployment experience is available.

5. Revisit the Rule and Guidance after Implementation

The NRC should also commit to revisiting Part 57 and its implementation guidance after the first applications and early deployments. Part 57 is intended to support a new high-volume licensing model, and the agency should not assume that the initial rule and guidance will perfectly anticipate how applicants, staff, manufacturers, host sites, and intervenors will use the pathway in practice.

The NRC should evaluate whether Part 57 is achieving its central purpose: enabling predictable, risk-informed, high-volume licensing of microreactors and NRC-defined small modular reactors that satisfy consequence-based eligibility criteria. That review should consider whether entry criteria are creating unnecessary cliff effects, whether manufacturing licenses and generic finality are providing real deployment value, whether environmental review and adjacent approvals are becoming hidden critical paths, whether guidance is imposing excessive burden, and whether applicants are able to move from FOAK review to NOAK deployment efficiently.

If experience shows that the rule or guidance is not achieving that purpose, the NRC should revise the guidance, update review procedures, issue clarifications, or pursue targeted rule amendments as needed. Post-implementation review should be treated as part of the Part 57 framework, not as an afterthought. A new licensing pathway should be measured by whether it works in practice, not only by whether it is internally coherent on paper.

Table 11. Recommendations – Rule Implementation

Affected Section	Recommendation
§ 57.270; § 57.60(a)(3); Subpart F; implementation guidance	Clarify how defects and noncompliance requirements apply to Part 57 licensees, manufacturing license holders, suppliers, and fleet operators; distinguish defects in the manufactured product from defects in site-specific installation, operational-program implementation, software or digital systems, remote monitoring infrastructure, and shared SSCs; explain how defect reporting interacts with manufacturing-license finality, fleetwide change control, and corrective action programs; and specify whether event reporting, cyber incident reporting, and change reporting may be submitted at the fleet or manufacturing-license level rather than repeated for each deployment site.
Subpart O; implementation guidance	Clarify enforcement accountability for Part 57 deployments before operation begins; identify how enforcement responsibility is assigned among site licensees, manufacturing license holders, contractors, suppliers, remote operations centers, and fleet support organizations; and distinguish enforcement actions arising from a fleetwide product, program, or process — which should be addressed across all affected deployments without duplicative site-by-site proceedings — from violations limited to a specific host site or local procedure.

Affected Section	Recommendation
Implementation guidance	Finalize NUREG-2271 as genuine guidance rather than a de facto mandatory application checklist; preserve applicant flexibility to use alternative methods supported by sufficient technical basis; align guidance terminology with the final rule's reactor-class framing; add practical implementation tools including a minimum-information matrix by pathway, Part 53-to-Part 57 transferability guidance, and lifecycle sequencing tables for related approvals under Parts 70, 71, 72, 73, 74, 75, and 140; and commit to updating NUREG-2271 after early applications and deployments if experience shows that portions are unnecessary or inconsistent with high-volume licensing.

XII. Cross-Rule Coordination

Part 57 should not be finalized or implemented as a stand-alone reform. It is being developed in the middle of a broader NRC rulemaking cycle driven by EO 14300 and the ADVANCE Act, and it is being carried out alongside Part 53 implementation, environmental review modernization, adjudicatory streamlining, and other ongoing reforms. Part 57's value depends not only on the text of new Part 57, but also on whether adjacent rules allow the pathway to operate as intended. A streamlined reactor-licensing framework can still fail in practice if hearing procedures, environmental review requirements, change-control rules, security requirements, transportation approvals, or financial protection requirements become uncoordinated critical paths.

Part 57 intersects directly with several other rulemakings in the EO 14300 package, and its assumptions depend on how those rulemakings are resolved. The most consequential are the proposed revision of the contested hearing process,[31] the rulemaking implementing the agency's NEPA obligations in Part 51,[32] the proposed modernization of security requirements affecting Parts 26 and 73,[33] the modernization of transportation package certification under Part 71,[34] and

[31] *Streamlining Contested Adjudications in Licensing Proceedings*, 91 Fed. Reg. 10450 (proposed Mar. 3, 2026) (Docket ID NRC-2025-1501; RIN 3150-AL58).

[32] *Implementation of the National Environmental Policy Act*, Docket ID NRC-2025-0478 (proposed rule, 10 C.F.R. Part 51, 2026).

[33] *Modernizing Security Requirements*, Docket ID NRC-2025-1303 (proposed rule, 10 C.F.R. Parts. 26, 50, 52, 72, 73, 95, 2026).

[34] *Modernizing Package Certification Requirements*, Docket ID NRC-2025-1667 (proposed rule, 10 C.F.R. Part 71, 2026).

the modernization of materials licensing affecting Parts 70, 72, and 140.[35] Part 57 also will also intersect with the broader effort to modernize reactor licensing, oversight, and siting.[36] The final rule or implementation guidance should identify these dependencies expressly and explain how applicants should sequence the related approvals.

The NRC should treat cross-rule coordination as a substantive implementation issue, not a housekeeping matter. Related rulemakings may change the assumptions on which Part 57 depends, including when review begins, what information must be complete, what standards govern agency review, what issues receive finality, and what adjacent approvals must be resolved before construction, fuel loading, operation, or deployment. Finalizing Part 57 before those assumptions are settled risks locking in a mechanism that a later rule invalidates. The NRC should not finalize Part 57 as though it can be cleanly separated from those adjacent reforms.

1. Coordinate Part 57 with Part 2 Adjudication and Hearing Procedures

The NRC should expressly coordinate Part 57 with its Part 2 adjudicatory rules. The Streamlining Contested Adjudications rulemaking proposes a category of "highly expedited proceedings" for hearings that need to move faster than ordinary contested adjudications, with shorter review schedules and narrower issues, and contemplates that the NRC could later designate additional application types—including applications tied to high-volume licensing of microreactors and modular reactors—as highly expedited. A related rulemaking on the mandatory hearing process is also part of the package.[37]

Part 57 is intended to support high-volume licensing, standardized applications, generic finality, and rapid deployment following site selection. Those features assume hearing procedures calibrated to standardized, design-referencing applications. If Part 57 applications instead proceed under hearing timelines and contention practices designed for bespoke large-reactor

[35] *Modernizing Materials Licensing*, Docket ID NRC-2025-1370 (proposed rule, 10 C.F.R. Parts. 30, 37, 40, 70, 72, 140, 2026).

[36] *Modernizing Reactor Licensing, Safety Oversight, and Siting Practices*, Docket ID NRC-2025-0975 (proposed rule, 10 C.F.R. Parts 2, 50, 51, 52, 54, 71, 100).

[37] *Increased Flexibility in the Mandatory Hearing Process*, Docket ID NRC-2025-1502 (final rule, 10 C.F.R. Parts. 2, 50, 51, 52, published Apr. 15, 2026).

licensing, the practical effect may be to recreate the delay and uncertainty Part 57 is intended to avoid.

The NRC should coordinate Part 57 with the Part 2 rulemaking and identify which Part 57 applications qualify for highly expedited proceedings. That treatment is appropriate for NOAK Part 57 applications that reference an NRC-approved design, manufacturing license, standardized operational program, or other licensing basis carrying generic finality, where the hearing scope should be limited to genuinely unresolved site-specific, applicant-specific, or new-and-material issues.

2. Coordinate Part 57 Environmental Review with Part 51 Modernization and NEPA/FRA Implementation

The NRC should coordinate Part 57's environmental-review framework with the agency's broader Part 51 modernization, NEPA implementation, and Fiscal Responsibility Act implementation efforts. Part 57 depends on several environmental-review tools that are also being revised or clarified in other proceedings, including categorical exclusions, applicant-prepared environmental documents, narrowed purpose-and-need statements, alternatives analysis, mitigated findings of no significant impact, and the use of generic or bounded environmental analyses. The final rule should make those interactions explicit so that Part 57 does not lock in assumptions that are later revised in Part 51 or related guidance.

This coordination is especially important for categorical exclusions and envelope-based review. Proposed Part 57 appears to rely on plant parameter envelopes, site parameter envelopes, and the New Reactor Generic Environmental Impact Statement to support streamlined environmental review for certain deployments. That approach can be efficient and legally durable if the NRC clearly identifies which environmental issues are bounded, which issues require site-specific confirmation, and which extraordinary circumstances would require additional review. But if Part 51 modernization changes the agency's definitions, procedures, or standards for categorical exclusions, Part 57 should not become misaligned with the agency's generally applicable NEPA framework.

The NRC should also clarify how Part 57 will treat applicant-prepared environmental assessments or environmental impact statements. Applicant-prepared documents may reduce staff burden and support high-volume licensing, but the final rule should preserve the NRC's responsibility for supervision, independent review, and final environmental conclusions. The NRC should avoid creating Part 57-specific procedures that conflict with, duplicate, or preempt generally applicable Part 51 procedures being developed for applicant-prepared environmental documents.

The same coordination is needed for purpose and need, alternatives analysis, and the scope of environmental effects reviewed by the agency. A Part 57 environmental review should focus on the NRC licensing action and the environmental effects sufficiently connected to that action and the NRC's statutory responsibilities. The final rule or associated guidance should explain how that principle applies to microreactor and small modular reactor deployments, including when the no-action alternative is sufficient, when site or deployment alternatives must be considered, and when broader energy-system alternatives are outside the useful scope of NRC review.

The NRC should also preserve flexibility for environmental issues that mature later in the licensing process. Part 57's safety and environmental submissions may proceed on different timelines, particularly for applications involving large designated areas, multiple potential sites, host facilities, or standardized reactor deployments. The final rule should identify how later-maturing environmental information will be handled without unnecessarily reopening matters already resolved and without foreclosing legitimate site-specific issues that could not have been evaluated earlier.

Part 57 should therefore include explicit coordination language recognizing its relationship to ongoing Part 51 modernization and NEPA/FRA implementation. The NRC should state whether Part 57 references to Part 51 will automatically conform to later Part 51 revisions, whether conforming amendments will be needed, and how applicants should proceed if Part 57 and Part 51 guidance are updated on different timelines. This would reduce procedural uncertainty and help avoid disputes over which environmental-review framework controls.

Beyond aligning its Part 51 references, the NRC should use Part 57 to advance a more proportional reactor environmental-review framework rather than running a streamlined safety pathway against environmental procedures still built for bespoke, one-off projects. The agency should develop broad-area and multi-site review tools, allow reliance on adequate existing site characterization, and clarify the site-boundary and partial-release questions that arise when a standardized design is deployed across many locations. Looking forward, Part 51 modernization should move away from automatic EIS treatment for reactor licensing actions whose impacts are bounded, small, or already addressed through generic analysis, preserving environmental review quality while reducing duplication for low-consequence reactors deployed at scale.

3. Help Applicants Choose Among Parallel Licensing Pathways

If Part 57 is finalized as a pathway parallel to Parts 50, 52, and 53, the NRC should provide clear pathway-selection guidance so applicants can understand when Part 57 is likely preferable, when Part 53 remains the better fit, and when Parts 50 or 52 should still be used. Without that guidance, additional optionality may increase uncertainty rather than reduce burden. This matters most for applicants already engaged under another framework or for designs that could plausibly qualify under more than one pathway; the NRC should explain how prior analyses, preapplication interactions, design information, environmental work, and safety evaluations may be reused or transferred when an applicant changes pathways.

The transferability asymmetry between Part 53 and Part 57—including the case of a first-of-a-kind design licensed under Part 53 whose later units would be deployed under Part 57—is addressed in the Part 57 and Part 53 section and in the recommendation to revise § 57.18(b). The coordination point here is forward-looking: clear pathway-selection and transferability guidance would reduce duplication, support efficient applicant planning, and help ensure that Parts 50, 52, 53, and 57 function as a coherent licensing system rather than as disconnected alternatives.

4. Resolve the Section 50.59 Change-Control Interface

Section 50.59 was developed around Part 50 licensing-basis concepts whose terminology and assumptions may not map cleanly onto Part 57's consequence-based structure. If Part 57 licensees must apply Section 50.59 without tailored guidance, the rule may import legacy analytical categories poorly matched to low-consequence, standardized, factory-manufactured, or remotely operated facilities. Part 53 addressed this concern by providing a risk-informed change-evaluation process in § 53.1550 rather than relying on Section 50.59; the NRC should explain how Part 57 licensees are expected to handle changes by comparison.

The NRC should either provide Part 57-specific change-control criteria or issue guidance explaining how Section 50.59 concepts apply to Part 57 licensing-basis information. That guidance should address manufactured reactors, fleetwide changes, changes approved at the manufacturing-license level, operational-program changes, and changes affecting remote monitoring, remote operation, or autonomous functions. The goal is to preserve regulatory control over safety-significant changes without requiring duplicative amendments for standardized reactors deployed across multiple sites.

5. Conforming Amendments and Coherent Definitions

Finally, the NRC should ensure that conforming amendments and definitions are complete and coherent. Part 57 will interact with multiple existing rules that were drafted around Parts 50 and 52. If those rules are amended only superficially, applicants may face uncertainty about whether legacy terminology applies to Part 57 facilities and how Part 57-specific concepts should be interpreted.

Part 57 appropriately carries forward the existing utilization-facility concept rather than attempting to create an entirely separate category. However, the NRC should clarify how that concept applies across the Part 57 lifecycle. In particular, the final rule or guidance should explain when a manufactured reactor, fueled module, reactor module in transport, factory-loaded reactor, module undergoing testing, or refurbished module is itself a utilization facility, part of a utilization facility, or not yet part of the licensed utilization facility until installation or operation at a licensed site. This clarification is important for manufacturing licenses, possession of special nuclear material, transportation approvals, inspection authority, safeguards, defect reporting, and enforcement responsibility.

The NRC should review definitions and cross-references involving construction, operation, production, or utilization facility, nuclear plant, reactor module, manufactured reactor, site boundary, exclusion area, operator, licensee, safety-related SSCs, change control, emergency planning, physical security, safeguards, and financial protection. Where Part 57 uses different concepts, the NRC should either define them expressly or explain how existing terms apply.

6. Coordinate Adjacent Approvals to Prevent Hidden Critical Paths

Several adjacent requirements could become hidden critical paths if they are not coordinated with Part 57 implementation. This comment addresses the substance of those interfaces where they arise: special nuclear material, transportation, storage, and financial protection under Parts 70, 71, 72, and 140 in the Fuel, Transportation, Storage, Decommissioning, and Financial Protection section; physical security and safeguards under Parts 73, 74, and 75 in the Emergency Planning, Security, and Safeguards section; and fitness-for-duty under Part 26 in the Operational Program and Oversight section. Several of these interfaces are themselves the subject of pending EO 14300 rulemakings—security requirements, package certification, and materials licensing—which makes coordinated sequencing more important, not less.

The coordination point here is narrower than the substantive recommendations in those sections. For each interface, the NRC should identify which adjacent approval or compliance demonstration is expected to be resolved in the Part 57 application, which may be standardized or afforded finality, and which must remain site-specific, and should state whether a conforming amendment is sufficient or whether implementation guidance is needed. For example, the final Part 57 should clearly identify which requirements under Parts 73, 74, and 75 must be resolved before construction authorization, fuel possession, fuel loading, initial operation, or deployment to a subsequent site. A conforming amendment may suffice for administrative coverage, but it is not sufficient where the adjacent rule supplies a substantive requirement that determines whether high-volume deployment is possible. Applicants should not encounter late-stage barriers that effectively determine whether a Part 57 deployment model is viable, particularly for redeployable or multi-site configurations where the sequencing of regulatory approvals may differ from a conventional single-site licensing process.

The rule also does not confirm whether the conforming amendments to Parts 73, 74, and 75 are sufficient to support the full range of Part 57 deployment models. Remote, operator-independent, redeployable, and multi-unit configurations each present regulatory interfaces that the existing parts were not designed to accommodate. If additional guidance is needed before applicants can rely on those parts for non- conventional deployment scenarios, the NRC should identify that need and commit to addressing it concurrently with the final rule rather than leaving it to emerge case-by-case.

For these reasons, the final Part 57 rule should include a cross-rule coordination roadmap. That roadmap should identify the related dockets, describe how Part 57 applicants should sequence related approvals, explain how finality will operate across rule parts, preserve flexibility to revisit Part 57 guidance if the combined framework proves misaligned after implementation, and commit the NRC to issuing implementation guidance before the first Part 57 applications are expected. This would make the framework more predictable, reduce avoidable regulatory fragmentation, and better align the rule with its purpose of enabling risk-informed, high-volume licensing of microreactors and NRC-defined small modular reactors that satisfy consequence-based eligibility criteria.

Table 12. Recommendations – Rule Coordination

Affected Section	Recommendation
10 C.F.R. Part 2; Streamlining Contested Adjudications and mandatory hearing rulemakings; final rule or guidance	Coordinate Part 57 with the Part 2 rulemakings and identify which Part 57 applications qualify for highly expedited proceedings, in particular NOAK applications referencing an approved design, manufacturing license, or standardized program carrying generic finality, with hearing scope limited to genuinely unresolved site-specific, applicant-specific, or new and material issues.
10 C.F.R. Part 51; NEPA and Fiscal Responsibility Act implementation; conforming amendments or guidance	Use categorical exclusions, plant and site parameter envelopes, and the New Reactor GEIS where supported, identifying which environmental issues are bounded and which require site-specific confirmation; preserve NRC supervision and final responsibility for applicant-prepared environmental documents; focus purpose and need and alternatives on the NRC licensing action; provide for later-maturing environmental information without reopening resolved matters; and state how Part 57's environmental provisions conform to Part 51 modernization and on what timeline.
Preamble; implementation guidance; Part 57 and Part 53 section (§ 57.18(b))	Provide pathway-selection guidance explaining when Part 57, Part 53, or Parts 50 and 52 is preferable, and how prior analyses, preapplication interactions, design information, environmental work, and safety evaluations may be reused or transferred when an applicant changes pathways.

Affected Section	Recommendation
10 C.F.R. § 50.59; cf. § 53.1550; Part 57-specific criteria or guidance	Provide Part 57-specific change-control criteria or guidance explaining how § 50.59 concepts apply to Part 57 licensing-basis information, addressing manufactured reactors, fleetwide changes, changes approved at the manufacturing-license level, operational-program changes, and changes affecting remote monitoring, remote operation, or autonomous functions.
10 C.F.R. Parts. 26, 70, 71, 72, 73, 74, 75, 140; conforming amendments or guidance; cross-references to the dedicated sections	For each adjacent interface, identify which approval is resolved in the Part 57 application, which may be standardized or afforded finality, and which must remain site-specific, and state whether a conforming amendment suffices or guidance is needed; a conforming amendment does not suffice where the adjacent rule supplies a substantive requirement that determines whether high-volume deployment is possible.

XIII. Conclusion

Part 57 is a significant component of the NRC's rulemaking activities in response to EO 14300 and a meaningful milestone in the agency's ongoing effort to adapt its regulatory framework to a new era of nuclear deployment characterized by advanced technologies, standardized designs, and repeatable licensing. BTI supports the NRC's decision to develop a framework that uses consequence-based eligibility criteria, standardized applications and approvals, manufacturing-license tools, generic finality, and targeted construction authority to support more predictable, repeatable, and efficient licensing. These provisions are directionally correct and should be preserved in the final rule.

The final rule needs to resolve the structural issues that could limit Part 57's effectiveness. The NRC should clarify that Part 57's central purpose is high-volume licensing for reactors that satisfy defined consequence-based eligibility criteria, rather than a loosely bounded technology category. The 1 rem TEDE criterion should remain the primary eligibility gate, while the 10 metric ton fuel-mass limit should be replaced, supplemented, or more clearly justified as a screen tied to safety-significant performance. A physical-security entry criterion should be added to enable high-volume licensing by limiting applications with unbounded complexity. The NRC should strengthen the manufacturing-license and standardized-operational-program provisions so that finality can be afforded where issues are genuinely resolved, while clearly identifying the site-specific matters that must remain for later review. The agency should also ensure that emergency planning, security, fitness-for-duty, environmental review, transportation, change control, and adjudicatory procedures are implemented in a way that is calibrated to consequence and administrable at fleet scale.

Finalized with these changes, Part 57 can become an important complement to Part 53 and the existing Part 50 and Part 52 frameworks. A durable final rule should also provide clear coordination mechanisms across licensing frameworks, support the transferability and reuse of approvals where issues have already been resolved, and facilitate the transition from first-of-a-kind demonstrations to nth-of-a-kind deployment without unnecessary re-litigation of standardized matters. Such a framework should preserve adequate protection, provide clear pathways for applicants and staff, and make regulatory treatment proportional to the hazards, uncertainties, and deployment configurations presented by the facility under review.

More broadly, the success of Part 57 should be evaluated by whether it establishes a stable and predictable regulatory pathway to achieve large-scale deployment. The rule's long-term value will depend on its ability to reduce unnecessary regulatory variability, enable efficient use of NRC and applicant resources, and provide sufficient certainty to support investment, manufacturing, and deployment decisions while maintaining the NRC's safety and security objectives. Consistent with the NRC's updated mission to protect public health and safety and the environment while enabling the safe and secure use of civilian nuclear technologies, Part 57 can support broader societal and environmental benefits by facilitating the deployment of advanced reactors that provide reliable energy, strengthen energy resilience, and contribute to emissions reduction goals.

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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Appendix. Addressing Specific Requests for Comment

Q1-1: In lieu of applying a deterministic material limit on the quantity of SNM to ensure safety, should the Commission consider an alternative performance-based entry criterion? Please explain the basis for your recommendation.

Yes. The NRC should replace or supplement the proposed 10 metric ton SNM/fuel-mass limit with performance-based entry criteria tied directly to safety-relevant outcomes, including site-boundary dose, decay heat removal capability, source term, radionuclide retention, accident progression, and the ability to maintain fuel and fission product barriers without active systems or prompt operator action. The NRC has itself identified an adiabatic heat rate threshold as a candidate performance-based alternative; BTI supports that approach as a more direct measure of decay heat removal capability and accident progression than heavy-metal mass inventory. As discussed in Section IV.1.C and Table 4, fuel mass is only a proxy for these safety characteristics and may exclude designs that can demonstrate low consequences while admitting designs whose safety case is less directly tied to the rule's consequence-based rationale. If the NRC retains the fuel-mass limit, it should explain the technical basis for the threshold, clarify whether it applies per reactor, per module, per plant, or per site, and identify whether an alternative eligibility showing is available for designs that satisfy the dose criterion but exceed the mass limit.

Q2-1: Besides the general license approach for certain construction activities in the proposed rule, are there other general licensing approaches for important components parts of utilization facilities that would benefit high-volume licensing or other regulatory processes for microreactors and other reactors with comparable risk profiles? Please explain the basis for your recommendation.

BTI does not recommend adding a broad new general license for important component parts on the present record. Instead, as discussed in Sections V.1.D, VI, and X.1—X.3, the NRC should ensure that the Part 57 manufacturing license, generic finality provisions, and adjacent Part 70, 71, and 72 interfaces can resolve standardized reactor design, component fabrication, quality assurance, inspection, criticality, transport, storage, and site-interface issues once and allow later deployment applications to rely on those determinations without duplicative review. That approach would provide the high-volume licensing benefit the question identifies while preserving clearer accountability than a freestanding component-part general license. A freestanding component-part general license would also create definitional uncertainty about when a manufactured component becomes part of a licensed utilization facility, with implications for inspection authority, safeguards obligations, and enforcement responsibility that are better addressed through the manufacturing license framework.

Q2-2: Given that the NRC anticipates that a review timeline for the required part 70 license will align with the timeline to complete a safety and security review of reactors via proposed part 57, would there be any benefits provided by a general license for a reactor in addition to the general license for construction activities proposed in part 57? Please provide your explanation.

BTI does not recommend adding a separate general reactor license on the current record. As discussed in Sections V.1.B, V.1.D, V.1.G, and X.1, the more important benefit would come from making the existing Part 57 tools work as an integrated high-volume licensing architecture: the manufacturing license should resolve standardized design, fabrication, quality assurance, configuration-management, and appropriate fuel-interface issues with meaningful finality, while the Part 57 application and any required Part 70 authorization should be sequenced clearly to avoid duplicative review. If the NRC concludes that a separate general reactor license is necessary, it should be limited to matters already resolved generically and should not substitute for clear allocation of Part 57 and Part 70 responsibilities.

Q3-1: Should any requirements in proposed part 57 be eliminated or made less burdensome or more flexible? If so, which ones? For existing requirements in 10 CFR chapter I that are referenced by proposed part 57, should any of them be similarly revised to the extent that they are relied upon by a proposed part 57 requirement? If so, which ones? Please explain the basis for your recommendation.

Yes. As discussed throughout the comment, the NRC should make several proposed Part 57 requirements less burdensome or more flexible where doing so would better align the requirement with Part 57's consequence-based and high-volume licensing rationale. The principal examples are replacing or supplementing the 10 metric ton fuel-mass limit with performance-based criteria; clarifying that MHA and MCA are consequence-based methods rather than PRA-style frequency screens; bounding the security case through a standardized physical-security eligibility criterion; making manufacturing licenses, generic finality, QA, codes-and-standards determinations, and standardized operational programs usable across repeat deployments; and replacing deterministic categorical-exclusion bars with impact-based environmental screens. The NRC should also revise or clarify referenced Chapter I requirements where they become critical-path constraints for Part 57, including Part 53 transferability, 10 C.F.R. § 50.59 change control, Part 51 environmental review, Part 70 SNM licensing, Part 71 transportation, Part 72 storage, Part 73 security/cybersecurity, Part 74/75 safeguards and accounting, Part 26 fitness-for-duty, and Part 140 financial protection. These recommendations are addressed in more detail in Sections IV—XII and Tables 4—11.

Q3-2: Recognizing that part 57 shares similar features with part 53, are there any provisions in part 57 that should be adapted for part 53 to enhance their complementary nature? For example, should the NRC include provisions in part 53 that would provide a general license for partial reactor construction or allow applicants to reference a general area for siting? If so, what, if any, modifications to the language in part 57 would be needed for it to be appropriate in part 53?

Yes. As discussed in depth in Section III and Table 3, the NRC should adapt for Part 53 the Part 57 tools that are broadly useful risk-informed regulatory mechanisms rather than features uniquely dependent on Part 57's six-month review model or 1-rem eligibility gate. These include flexible codes and standards, graded quality assurance, manufacturing-license concepts, generic finality, a general license or at-risk construction concept, remote and autonomous operation provisions, and change-control tools for standardized designs. Any Part 57-derived general construction or general-area siting provision incorporated into Part 53 should be modified so that it is justified by the applicant's design maturity, site-parameter envelope, environmental showing, and safety case, rather than automatically tied to Part 57's consequence-based eligibility criteria.

Q3-3: Because the proposed part 57 directs licensees to use 10 CFR 50.59, which uses the term "important to safety," and that term is not used in part 57, should the NRC explain in a guidance document how a part 57 licensee should use 10 CFR 50.59 or should the final part 57 include its own specific 10 CFR 50.59-like process?

Yes. The NRC should either include a Part 57-specific change-control process or issue guidance explaining how 10 C.F.R. § 50.59 concepts apply to Part 57 licensing-basis information. As discussed in Section XII.4 and Table 12, § 50.59 was developed around Part 50 licensing-basis concepts and terminology that may not map cleanly onto Part 57's consequence-based, standardized, manufactured, remote, or autonomous deployment models. Any final rule or guidance should address manufactured reactors, fleetwide changes, changes approved at the manufacturing-license level, operational-program changes, and changes affecting remote monitoring, remote operation, or autonomous functions, with the objective of preserving NRC control over safety-significant changes without requiring duplicative amendments for standardized reactors deployed across multiple sites.

Q3-4: Is a single notice in the Federal Register for each joint application for a construction permit and associated operating license(s) sufficient and appropriate for notice for large geographic areas? Or should additional measures be employed to put the public on notice of a hearing opportunity for a large geographic area, and if so, what measures?

A single Federal Register notice should not be treated as categorically sufficient for every joint CP/OL application involving a large designated geographic area. As discussed in Sections IX.2-IX.3 and XII.1, applications covering large areas raise distinct questions about which issues are resolved at the area level, which require later site-specific confirmation, and how hearing procedures should distinguish generic issues from genuinely site-specific or applicant-specific matters. For large geographic-area applications, the NRC should supplement Federal Register notice with additional measures reasonably designed to reach affected communities, such as applicant-maintained project notice pages, direct notice to relevant state, local, Tribal, and host-site authorities, and geographically targeted notice when later site selection or site-specific confirmation occurs within the approved area.

Q3-5: Should the NRC look holistically at the duration of renewals for manufacturing licenses, design certifications, and standard design approvals across all parts?

Yes. The NRC should review renewal duration and reopening standards for manufacturing licenses, design certifications, and standard design approvals holistically across Parts 50, 52, 53, and 57 so that standardized designs are not subject to inconsistent renewal-driven re-review depending only on the licensing pathway selected. As discussed in Sections III, V.1.D, V.1.E, and XII, the central regulatory value of these tools is the ability to resolve design, manufacturing, interface, and programmatic issues once and carry those resolutions forward into later deployments, subject to defined reopening standards for materially new information or safety-significant departures. The NRC should preserve indefinite Part 57 SDA duration, clarify the circumstances for reopening, and align renewal and finality expectations across related approvals so that high-volume deployment is not undermined by unnecessary duplicative review.

Q3-6: Should the NRC consider periodicities other than the proposed 5-year interval for FSAR updates?

BTI does not take a position in this comment on a specific alternative FSAR update periodicity. The NRC should, however, consider whether a fixed 5-year interval is the best fit for standardized, manufactured, fleet-deployed, or remotely operated Part 57 reactors, or whether a graded approach would better align FSAR updates with safety-significant changes, operating experience, fleetwide configuration changes, and changes approved under the applicable change-control process. As discussed in Sections V.1.F, VIII.3, and XI.4, the key objective should be to preserve current, auditable licensing-basis information without creating duplicative update obligations for standardized reactors whose common design and programmatic issues have already been resolved generically.

Q4-1: Should a proposed part 57-compatible early site permit process be developed? Describe the potential value of creating a proposed part 57-compatible ESP process, including the benefits and drawbacks of such an approach for applicants and stakeholders, and whether this process could facilitate more timely and predictable licensing outcomes.

Yes, with qualification: the proposed rule already contains the siting analysis such a process requires, so the need is to make that analysis standalone and referenceable rather than to create new analytical requirements. Section 57.60(d) already lets an applicant designate one or more large geographic areas, screen unsuitable areas, and bound environmental and siting issues at the area level, and the site-parameter-envelope concept in draft NUREG-2271, aligned to the New Reactor GEIS, supplies the method. As proposed, however, § 57.60(d) is available only within a joint construction-permit and operating-license application, with the area's life-cycle impacts heard at the construction-permit stage, so it cannot yield a severable site approval that a later or separate deployment application can reference. A Part 57-compatible early site permit should therefore decouple that siting review from the reactor-specific application and issue it as a standalone, bankable approval, using the area-designation methodology, site-parameter envelopes, and unsuitable-area screening the rule already defines. The NRC should also extend the referenceability provisions in §§ 57.16(b)(2) and 57.18(b), which currently reach only Parts 50 and 52, to early site permits issued under Part 53. Structured this way, a Part 57-compatible early site permit would make licensing outcomes more timely and predictable by separating reusable site and environmental findings from deployment-specific matters. These points are developed in Sections IX.2 and IX.3 and in Section III.

Q4-2: What types of site issues (e.g., seismic, emergency planning, tribal consultations) would benefit most from early resolution under such a process?

The site issues that would benefit most from early resolution are those that are either capable of being bounded across a site or geographic area, or that can create critical-path uncertainty if deferred until the CP/OL stage. These include seismic, geotechnical, meteorological, hydrological, flooding, and external-hazard parameters; cultural-resource, tribal-consultation, endangered-species, wetlands, water-use, land-use, environmental-justice, and transportation-route issues; host-site hazards and interfaces; emergency-planning and offsite-response coordination; and cumulative impacts for multi-unit or phased deployments. As discussed in Sections IX.2-IX.4, the NRC should distinguish issues that can be resolved generically or area-wide from issues that require later site-specific confirmation, so early resolution produces real finality without becoming either overbroad or prematurely site-specific.

Q4-3: Would a part 52-type ESP process reduce licensing uncertainty and costs for developers, and if so, how?

Yes. A Part 52-type ESP process, adapted to Part 57's consequence-based and high-volume deployment framework, could reduce licensing uncertainty and cost by allowing applicants to resolve site suitability, environmental constraints, external hazards, emergency-planning interfaces, consultation obligations, and cumulative-impact assumptions before submitting a full CP/OL or deployment application. As discussed in Sections V.1.H, IX.2, and IX.3, early resolution of bounded site and environmental issues would allow later Part 57 reviews to focus on what is new, changed, unresolved, or site-specific rather than reopening issues already addressed within an approved site-parameter envelope. The cost-benefit would come from reducing duplicative studies, avoiding late-stage siting surprises, supporting host-site and financing decisions earlier, and giving stakeholders a clearer point at which to raise site-related concerns. The process should be structured to preserve finality only for issues actually bounded and resolved, with later confirmation for issues that depend on the final deployment location or configuration.

Q5-1: Besides the volume of waste, would there be differences in the process for refurbishment versus decommissioning of the reactor, if both occurred at the same facility, that would be important to consider with regard to enabling more efficient and safe streamlining of the decommissioning licensing and the license termination processes? Please provide a rationale supporting your comment.

Yes. Refurbishment and decommissioning may occur at the same facility, but they are not the same regulatory process and should not be streamlined only by reference to waste volume. As discussed in Sections X.3, X.4, and X.6, refurbishment may involve continued possession, inspection, repair, sealed-core verification, storage, return-to-service decisions, transportation interfaces, and potential redeployment, while decommissioning and license termination require final radiological characterization, disposition of residual contamination, funding assurance, and release of the site or unit from licensed obligations. The NRC should distinguish the two processes so that reusable or returnable reactor modules can move through storage, refurbishment, or redeployment without being treated as if they are necessarily entering final decommissioning, while also ensuring that decommissioning obligations and funding assurance remain continuous for any material, module, site, or facility that retains radiological responsibility.

Q5-2: The NRC's current regulations generally restrict the use of decommissioning trust funds to activities conducted after permanent cessation of operations, unless an exemption is granted. The NRC has received stakeholder interest in accessing decommissioning funds during reactor operation for the removal or replacement of major components when those activities would ultimately be necessary for decommissioning. The NRC is seeking stakeholder input on whether, and under what conditions, limited access to decommissioning trust funds for such activities during reactor operation should be considered. For example, is there an anticipated need to access radiological decommissioning funds during operations to facilitate the removal of a reactor for refurbishment or other major radioactive component disposal? Please provide a rationale supporting your comment.

Yes, but only under defined conditions. For transportable, replaceable, or return-to-manufacturer reactors, limited access to decommissioning funds during operation may be appropriate where removal, replacement, disposal, or transfer of a major radioactive component is a radiological activity that would otherwise be necessary for decommissioning or license termination. As discussed in Section X.4, the NRC should not rely solely on decommissioning-funding mechanics designed for a single long-lived operating site, because a transportable module may leave the site before permanent cessation of operations and the radiological inventory may transfer to a manufacturing, storage, refurbishment, or disposal facility. Any access should be conditioned on maintaining adequate remaining funding assurance, documenting the reduction or transfer of the operating-site obligation, and ensuring that the receiving facility or fleet-level funding mechanism covers the radiological inventory after transfer.

Q6-1: Should the NRC include a specific provision for releasing a part of a nuclear plant or site for unrestricted use before license termination in proposed part 57? If so, how should the NRC consider adapting the approach in § 50.83 and § 53.1080 to make the provision applicable to licensees under proposed part 57?

Yes. The NRC should include a Part 57-specific provision allowing partial site release and individual-unit license termination before termination of the entire plant or site license, adapted from § 50.83 and § 53.1080 to account for staged, multi-unit, manufactured, and transportable deployment models. As discussed in Section X.4, Part 57 deployments may involve units that are deployed, removed, replaced, refurbished, or retired at different times, and a rule requiring the entire plant to remain under decommissioning control until the last shared facility is decommissioned could create unnecessary administrative barriers. The adapted provision should allow unrestricted release where the licensee demonstrates that the released area or unit satisfies Part 20, Subpart E, that shared SSCs or shared radiological areas remain under appropriate license control, and that decommissioning and funding obligations for remaining units, modules, and shared facilities remain continuous.

Q7-1: Provide feedback on the need for alternate dose rates for transportable microreactors, the technical basis for those alternate dose rates, and the safety implications for those alternative dose rates.

BTI does not take a position in this comment on a specific alternate dose-rate limit for transportable microreactors. The NRC should consider alternate dose rates only where the applicant provides a technical basis showing that the transportable reactor's package design, shielding, radionuclide inventory, decay heat, criticality safety, accident-condition performance, route controls, and post-transport inspection program provide an equivalent or superior safety basis to the generally applicable Part 71 framework. As discussed in Sections X.2 and XII, transportable microreactors present a distinct exposure scenario from the reactor accident analyzed under § 57.25(a), so any alternate dose-rate framework should be justified under transportation-specific conditions rather than inferred solely from Part 57 eligibility. The safety implication is that flexibility may be appropriate for standardized sealed modules, but only if the NRC defines the governing dose, shielding, heat-removal, criticality, accident-condition, inspection, route, sabotage, and emergency-response standards clearly enough to avoid case-by-case uncertainty.

Q7-2: Are there cost-benefit considerations beyond the costs and benefits associated with rulemaking (e.g., the costs of additional shielding due to lower dose rates) that the NRC should consider with respect to alternate dose rates for transportable microreactors? Please provide a basis for your response.

Yes. In evaluating alternate dose rates for transportable microreactors, the NRC should consider cost-benefit factors beyond the administrative cost of rulemaking, including shielding mass and design impacts, transportability, route availability, shipment frequency, worker and public exposure, package certification burden, post-transport inspection needs, emergency-response planning, security requirements, and the possibility that overly restrictive dose-rate limits could increase lifecycle risk by requiring more shipments, more handling, or more complex refurbishment logistics. As discussed in Sections V.1.D and X.2, transportation can become a recurring bottleneck for standardized manufactured reactors if package approval, inspection, route, security, and emergency-response requirements are resolved case by case rather than through predictable performance-based criteria. The NRC should therefore evaluate alternate dose rates as part of the full transportable-reactor lifecycle, not only as a shielding-cost issue.

Q7-3: Provide feedback on the impact to international and interstate shipments if there were alternate transportation package dose rate limits for transportable microreactors.

Alternate transportation package dose-rate limits could affect interstate and international shipments by changing route availability, carrier acceptance, state and local coordination needs, emergency-response planning, security planning, and compatibility with non-NRC transportation requirements. For international shipments, the NRC should also consider whether any alternate limits would create misalignment with importing, exporting, transit-country, or international package-approval expectations, which could reduce the practical value of a transportable-reactor model even if the package is acceptable under Part 71. As discussed in Sections V.1.D, X.2, and XII, any alternate dose-rate framework should be coordinated with Part 71 modernization and should be structured so that repeated shipments of identical modules can rely on clear, predictable package, route, inspection, emergency-response, and security standards rather than recurring case-by-case determinations.

Q7-4: What assumptions should the NRC use when estimating the number of shipments, exposure scenarios, and expected dose rates for fresh and irradiated transportable microreactors? Please provide a basis for your response.

Assumptions should distinguish initial factory shipment, fuel loading, site installation, replacement or module removal, shipment for refurbishment, return shipment after refurbishment, shipment to storage, and final shipment for decommissioning or disposal; they should also distinguish fresh, irradiated-but-cooled, damaged, and end-of-life module conditions. As discussed in Sections X.2 and X.6, the NRC should base dose-rate and exposure assumptions on the module's demonstrated inventory, burnup, cooling time, shielding, decay heat, criticality-safety controls, route controls, handling frequency, worker activities, public exposure pathways, post-transport inspection, emergency-response assumptions, and whether repeated shipments of identical modules can rely on a generically approved package and transport program. The analysis should avoid assuming either a one-time shipment model or a large-reactor spent-fuel model where the applicant's transportable-reactor lifecycle is materially different.

Q8-1: To support licensees developing an FFD program tailored to their own specifications, what core elements (such as program policy and governance; program scope and applicability; behavioral observation; specimen collection and testing; substances tested; pre-employment screening; for-cause and post-event measures; periodic medical fitness evaluations for licensed reactor operators; program-related training; program audits and corrective actions; and supportive resources, such as an employee assistance program or other equivalent substance abuse counseling) should the NRC include in its program requirements or guidance to help licensees ensure the trustworthiness, reliability, and fitness of personnel and to support FFD program consistency within the industry? Please provide a basis for your response.

The NRC should identify a core set of FFD elements for licensee-specific programs, but those elements should be graded to the facility's staffing model, operator-dependence, maintenance activities, and safety significance of credited human actions. At a minimum, guidance should address program governance, scope and covered personnel, behavioral observation, pre-access and pre-employment screening, for-cause and post-event testing, substances tested, specimen collection and testing methods, medical fitness where licensed operators or GLROs perform safety-significant functions, training, audits, corrective actions, recordkeeping, and access to employee-assistance or equivalent support resources. The NRC should distinguish operator-independent, remotely monitored, remotely operated, and maintenance-intensive facilities so that FFD requirements support trustworthiness and reliability without importing large-reactor staffing assumptions into Part 57 deployments. The program should be consistent enough to preserve industry-wide expectations, but flexible enough to account for facilities with few onsite personnel, fleetwide operations centers, GLRO-based staffing, or no credited operator action for safety.

Q8-2: What approach or methodology should be used to determine whether a credible operator or maintenance error could result in exceeding the dose-based entry criterion specified in proposed § 57.25(a)? Please provide a basis for your response.

The NRC should use a consequence-based human-error screening methodology tied to the same MHA or MCA framework used to demonstrate compliance with § 57.25(a), rather than a generic PRA-style frequency screen. The applicant should identify credible operator and maintenance errors associated with safety-significant functions, remote or autonomous controls, maintenance configurations, bypasses, testing, module replacement, and post-maintenance restoration, and then evaluate whether those errors, alone or in credible combinations with equipment states they could cause, could result in exceeding the 1 rem TEDE criterion. As discussed in Sections IV.1.B, VII.1, VII.2, and Table 7, the analysis should distinguish operator-independent facilities from facilities that credit human action, and should determine whether the design remains below § 57.25(a) without operator action or whether the facility must transition to more conventional operator licensing, staffing, training, inspection, or change-control requirements.

Q8-3: What alternative criteria could be applied to proposed § 26.3(f)(3) to determine whether a licensee should be permitted to implement an FFD program of its own specification or be required to implement either the requirements of part 26 except subparts K and P or the program described in proposed subpart P of part 26? Please provide a basis for your response.

The NRC should use consequence- and function-based criteria to determine whether a Part 57 licensee may implement an FFD program of its own specification under proposed § 26.3(f)(3). Relevant criteria should include whether the facility is operator-dependent or operator-independent; whether any human action is credited to maintain compliance with § 57.25(a); whether personnel perform safety-significant maintenance, surveillance, testing, remote-operation, cybersecurity, security, or emergency-response functions; whether the facility is staffed onsite, remotely monitored, remotely operated, or autonomous; and whether fleetwide operations centers or standardized maintenance teams perform covered functions across multiple sites. Licensees whose human actions are not credited for safety and whose personnel functions are limited, standardized, and auditable should be eligible for a tailored FFD program, while facilities that rely on personnel to perform safety-significant actions should be subject to more prescriptive Part 26 requirements or the proposed Subpart P framework.

Q9-1: Consistent with the objectives of this proposed rule to support high-volume licensing of microreactors and other reactors with comparable risk profiles, should the NRC include certain proposed part 57 applications within the definition of "highly expedited proceeding" if the NRC issues a final rule modifying the NRC's contested hearing process with special requirements for highly expedited proceedings? Specifically, when a proposed part 57 application references an NRC approval providing finality on the design in the adjudicatory proceeding, the scope of issues for adjudication would be narrow, supporting an even more expedited schedule for filings and decisions. Licensee-initiated amendments to proposed part 57 licenses should be similarly narrow. Therefore, should the NRC include these types of proposed part 57 applications within the § 2.4 definition of "highly expedited proceeding" and thereby apply requirements for highly expedited proceedings to these applications?

Yes. The NRC should include within the § 2.4 definition of "highly expedited proceeding" Part 57 applications that reference an NRC-approved design, manufacturing license, standardized operational program, or other licensing basis that has been afforded generic finality, where the remaining hearing issues are limited to genuinely unresolved site-specific, applicant-specific, departure-related, or new and material issues. As discussed in Section XII.1 and Table 12, Part 57's high-volume licensing model depends on hearing procedures calibrated to standardized, design-referencing applications; otherwise, contested proceedings could recreate the delay and uncertainty that Part 57 is intended to avoid. The same treatment should apply to licensee-initiated amendments where the amendment scope is narrow and does not reopen matters already resolved with finality.

Q9-2: What hearing schedule requirements should apply to proposed part 57 joint applications for construction permits and operating licenses that would not be included within the proposed definition of "highly expedited proceeding"?

Part 57 joint CP/OL applications that do not qualify as "highly expedited proceedings" should still be governed by a bounded and front-loaded hearing schedule tailored to Part 57's rapid licensing purpose. As discussed in Sections V.1.A, V.1.F, and XII.1, the NRC should distinguish applications that reference resolved generic matters from applications involving less mature designs, unresolved site-specific issues, material departures, or first-of-a-kind questions that require a fuller hearing record. For non-highly-expedited Part 57 joint applications, the schedule should require early identification of contested issues, early separation of generic issues from site-specific and applicant-specific issues, prompt resolution of threshold contention admissibility questions, and defined milestones for discovery, testimony, and initial decision. The schedule should be less compressed than the "highly expedited proceeding" schedule, but still materially more structured than an ordinary bespoke large-reactor adjudication where the issues are narrower and the Part 57 application is otherwise complete.

Q10-1: Should the NRC allow remote operations and autonomous operations of nuclear power plants that demonstrate low consequences? What, if any, additional requirements and guidance are necessary for the regulatory review of remote operation and autonomous operation as part of the rapid licensing envisioned under part 57? Please provide a basis for your response.

Yes. The NRC should allow remote operation and autonomous operation for Part 57 facilities that demonstrate low consequences, but it should review those functions through a graded framework that distinguishes remote monitoring, remote operation of non-safety-related functions, remote operation of safety-related functions, supervised autonomous operation, and fully autonomous operation. As discussed in Sections VII.1—VII.3 and Table 7, the NRC should provide additional requirements or guidance on the evidentiary showing for autonomous operation, cybersecurity and data integrity, communications availability and latency, redundancy, loss-of-link response, fallback safe-state behavior, manual-shutdown expectations, OIF/GLRO classification, maintenance-error treatment, event-triggered inspections, and change-control triggers. That approach would preserve Part 57's flexibility while ensuring that remote and autonomous functions are reviewed in proportion to their safety significance and role in the facility's consequence-based safety case.

Q11-1: To what extent should the proposed part 57 implementation guidance consider the single failure criterion as a desired attribute to enhance reliability and defense in depth, rather than as a limiting factor in determining whether reasonable assurance of adequate protection exists for advanced reactor *designs with enhanced margins of safety and/or that use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions? Please provide a basis for the response.*

The NRC should treat the single failure criterion as a potentially useful reliability and defense-in-depth attribute, not as a categorical prerequisite for reasonable assurance of adequate protection under Part 57. For reactors that demonstrate compliance with § 57.25(a) through simplified, inherent, passive, or other innovative safety functions, implementation guidance should ask whether the design provides adequate reliability, margin, functional independence, and defense in depth for the safety functions credited in the licensing basis, rather than mechanically requiring satisfaction of a deterministic single-failure formulation developed for large light-water-reactor architectures. As discussed in Sections IV.1.A—B, IV.1.C, VII.1, and Table 4, Part 57's streamlined treatment should remain tied to the applicant's consequence-based safety showing, and the NRC should avoid reintroducing large-reactor assumptions through guidance where they do not add safety value for a design that has already demonstrated bounded consequences.

Q11-2: Are there criteria or methods that can be included in the proposed part 57 implementation guidance that provide balance between the use of deterministic methods such as the single failure criterion and applicant-derived risk information to provide for reasonable assurance of adequate protection of public health and safety? Please provide a basis for the response.

Yes. Part 57 implementation guidance should provide a structured method for balancing deterministic design attributes, such as the single failure criterion, with applicant-derived risk information and consequence-based analysis. The NRC should require applicants to identify the safety functions credited in the licensing basis, explain the deterministic attributes used to support those functions, provide applicant-derived risk or reliability information where it improves the decision, and demonstrate that the resulting design provides adequate margin, functional independence, defense in depth, and compliance with the § 57.25(a) consequence criterion. As discussed in Sections IV.1.A—B, IV.1.E, and Table 4, deterministic criteria should be treated as tools for evaluating reliability and defense in depth, not as automatic substitutes for evaluating whether a particular Part 57 design maintains reasonable assurance of adequate protection.

Q12-1: Are the NRC's conclusions—existing pathways designed for large or specialized facilities (e.g., part 52 with inspections, tests, analyses, and acceptance criteria (ITAAC) or part 50 requirements tailored to *large LWRs) would impose unnecessary burden and extend review timelines for microreactors—accurate and sufficiently supported?*

Yes. The conclusion is accurate, and it is adequately supported, though the support is most durable when grounded in the fact that Part 57's purpose is for high-volume deployment. As discussed in Sections II, III, and V.1.A, existing Part 50 and Part 52 pathways, including Part 52's ITAAC structure and large-LWR-oriented requirements, may impose unnecessary burden and extend review timelines for reactors that satisfy Part 57's consequence-based entry criteria and are mature enough for standardized, repeat deployment.

Q12-2: What additional, intermediate, or hybrid alternatives (e.g., targeted modifications to part 52, streamlined ITAAC constructs, or scoped use of part 53 elements) should the NRC evaluate to meet the statutory objectives while minimizing cost and schedule impacts? Please provide data, examples, or suggested regulatory text that could enable rapid, high-volume licensing of microreactors within or alongside existing regulations.

BTI recommends the scoped use of Part 53 elements and does not recommend streamlined ITAAC constructs. As discussed in Section V.1.A, Part 57 intentionally avoids the ITAAC structure of Part 52 and instead achieves efficiency by front-loading a complete design submission; that complete-design requirement is what makes a single safety review and a 6-to-12-month timeline possible, so grafting on progressive ITAAC closure would undermine the very feature that enables rapid review. The more productive path is to align Part 53 and Part 57, as detailed in Sections III and V: revise § 57.18(b) to allow incorporation by reference of Part 53 approvals on the same basis already extended to Parts 50 and 52; incorporate broadly useful Part 57 tools into Part 53 (such as graded quality assurance, flexible codes and standards, etc.); and allow prior Part 50, 52, or 53 approvals to support NOAK use of Part 57's general construction license where the design satisfies Part 57 eligibility criteria and the relevant generic issues are already resolved. The Hermes and Hermes 2 reviews illustrate the payoff: the NRC completed the Hermes 2 review using about 60 percent fewer resources than expected by drawing on the earlier Kairos review, and BTI recommends formalizing that generic-finality approach across Parts 53 and 57.