

NRC WHITE PAPER FOR DISCUSSION

Leveraging Information from Previously Authorized Facilities

PURPOSE

This white paper is part of the U.S. Nuclear Regulatory Commission's (NRC's) strategy to establish an expedited pathway to approve reactor designs that have been previously authorized and tested and that have demonstrated the ability to function safely by the U.S. Department of Energy (DOE) or the U.S. Department of War (DOW). This paper provides the NRC staff's initial views on the overlap between DOE and NRC requirements. Future documents will include transitioning from a DOW authorization to an NRC license.

This white paper provides information on NRC requirements and associated guidance documents in foundational areas of reactor safety and the protection of public health and safety and describes related considerations for expedited pathways for NRC approval of advanced nuclear technologies previously authorized by DOE. This white paper identifies the NRC staff's initial views on the types of information and prior determinations that may be leveraged, the considerations relevant to determining their applicability to NRC requirements, and the basis on which the NRC staff will determine the appropriate extent of reliance on prior authorizations in NRC licensing. This focus is on key, foundational topical areas that are important in NRC licensing reviews and that may benefit from leveraging information generated from a prior DOE authorized facility. The intended outcome of this coordinated approach is to facilitate commercial reactor licensing and industry deployment efficiencies by enabling the NRC staff to adjust the scope and depth of their review, where appropriate, when assessing the risks associated with newly proposed commercial reactor facilities.

The NRC staff encourages developers to consider this information before and during development of their DOE authorization application and related NRC commercial licensing strategy, including pre-application engagement with the NRC staff. This white paper is intended to facilitate discussion related to how applicants and the NRC staff can best maximize the overlap between what would be required to obtain a DOE authorization and an NRC license. During its review of future applications, the NRC staff may be able to leverage more information that has been previously reviewed and accepted by DOE when the NRC staff performs its subsequent review of a commercial license application.

This white paper is best read in conjunction with other closely related NRC activities, strategies, and stakeholder engagement opportunities as summarized in the section titled "Guidance," below. This discussion covers construction permits (CPs) and operating licenses (OLs) under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities;" design certifications (DCs), combined licenses (COLs), standard design approvals (SDAs), and manufacturing licenses (MLs), under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants;" CPs, OLs, COLs, MLs, SDAs, and DCs under 10 CFR Part 53, "Risk-Informed, Technology Inclusive Regulatory Framework for Commercial Nuclear Plants;" or CPs, OLs, MLs, and SDAs under the proposed Part 57, "Licensing Requirements for Microreactors and Other Reactors With Comparable Risk Profiles."

This white paper supports streamlined review of applications by enabling NRC staff to adjust the scope and depth of its review, when appropriate, in areas where DOE authorized reactor facilities provide pertinent information. The NRC staff anticipates that select methods and approaches to receive an authorization to construct and operate a reactor facility under DOE authorities will overlap

with NRC licensing requirements in the topical areas addressed in the appendices of this white paper. Although this white paper focuses on certain topical areas, additional areas of overlap between DOE and NRC applications are expected. Preapplicants and applicants should engage with the NRC early to identify any further technical information needed to support a complete NRC application. While this information is primarily intended for applicants for NRC licenses that first pursue a DOE authorization, it may also be useful for prospective applicants not pursuing DOE authorization as a part of their NRC licensing strategy.

This white paper does not affect the NRC's commitment to conducting independent reviews of applications, nor does it alter NRC's regulatory requirements. Regardless of whether an applicant references a DOE authorization, it remains the applicant's responsibility to demonstrate compliance with all applicable requirements in the Atomic Energy Act of 1954, as amended (AEA), and NRC regulations. An applicant must submit a complete application, and the NRC staff will conduct the assessments necessary to determine whether the application meets NRC requirements. References to information from a design authorized and tested by DOE must be clearly identified in an NRC license application. The applicant is responsible for justifying the applicability and relevance of the information it seeks to rely upon, identifying any limitations, and providing supplemental information when the referenced analyses or testing do not fully demonstrate compliance with NRC regulations. Where appropriate, the NRC staff will tailor the scope and depth of its review on new and significant information in the license application and build on DOE's related assessments and findings that are relevant to NRC licensing.

BACKGROUND

Since the 1970s, applicants and the NRC staff have considered and used DOE-generated technical information in the licensing of commercial nuclear reactors and plants under section 103 of the AEA. For example, applicants for NRC licenses have used information generated by DOE and its national laboratory infrastructure to support addressing NRC requirements in individual topical areas (e.g., test results supporting the qualification of a specific material or data supporting the validation of an analytical tool). This approach is consistent with existing NRC regulations, including paragraph (e) of 10 CFR 50.43, "Additional standards and provisions affecting applications for class 103 licenses and certifications for commercial power," as applied to the NRC licensing or other approval of new reactor facility designs that would use simplified, inherent, passive, or other innovative means to accomplish their safety functions. The regulations in 10 CFR 53.440(a) and proposed 10 CFR 57.60(a) also include similar requirements.

There is growing interest among prospective NRC applicants in pursuing DOE or DOW authorizations for the construction and operation of prototypical reactor facilities for the purpose of demonstrating various reactor designs (or major portions thereof). Prospective NRC applicants could then reference, as appropriate, these DOE- or DOW-authorized demonstrations in applications for NRC licenses, approvals, or certifications for identical or similar reactor facility designs.

Executive Order (E.O.) 14300, "Ordering the Reform of the Nuclear Regulatory Commission," Section 5, "Reforming and Modernizing the NRC's Regulations," Paragraph (d) directs the NRC to establish an expedited approval pathway and states:

Establish an expedited pathway to approve reactor designs that the DOD or the DOE have tested and that have demonstrated the ability to function safely. NRC review of such designs shall focus solely on risks that may arise from new applications permitted by NRC licensure, rather than revisiting risks that have already been addressed in the DOE or DOD processes.

In response to the direction in EO 14300 5(d), the NRC completed a rulemaking that adds a new 10 CFR 50.43(f) and updates 10 CFR 53.440(a) (ML26295A067). Further, the NRC has a proposed

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rule that would establish new regulations in 10 CFR part 57 (91 FR 23628). This completed rulemaking and the proposed rule, if approved, would improve NRC licensing review efficiency, where applicable, by providing additional means for applicants for NRC licenses, approvals, or certifications to demonstrate the safety functions of their reactor designs. The white paper and its appendices reference rulemakings that are in progress as proposed rules, recognizing that the content may change in the final rule or the final rule may not be published.

In addition to this white paper and the rulemakings mentioned, there are other documents the NRC has issued related to leveraging information from DOE and DOW authorizations:

- Memorandum to NRC Staff, “Expectations for NRC Staff Involvement with Prospective NRC License Applicants Planning to Refer to DOE and DOW Authorization Processes” (ML25322A208).
- Addendum No. 9 to the Memorandum of Understanding titled, “Coordinating DOE and NRC Technical Expertise and Knowledge on Advanced Nuclear Reactor and Advanced Reactor Fuel Technologies,” between the NRC and DOE (ML25303A288).

OVERVIEW OF NRC LICENSING AND DOE AUTHORIZATION PROCESSES

The NRC licensing and DOE authorization processes have several core overarching principles in common: protecting public health, safety and the environment and using risk-informed and performance based principles in the review of a design’s safety basis, analyses and supporting data. However, the NRC and DOE have their own requirements and specific review and approval processes driven by their different statutory authorities and functions. For example, DOE generally provides for authorization of research, demonstration, and prototype testing reactors focus on data collection, whereas the NRC licenses commercial reactors that may operate for decades at full power.



Figure 1: DOE Authorization Process versus NRC Licensing Process

NRC Commercial Licensing Process:

The NRC’s licensing process typically includes the major phases discussed below, with the specific milestones and requirements being tailored to the specific licensing framework an applicant pursues

(e.g., 2-step CP and OL licensing framework of 10 CFR Part 50 compared to the one-step combined license framework of 10 CFR Part 52).

Activities supporting the NRC licensing process can begin well before an applicant submits a license application. Prospective applicants typically open engagement by sending a Letter of Intent to the NRC, which signals their plans and allows the agency to begin preparing for the review. This early notification enables NRC staff to open a docket,¹ assemble an appropriate review team, schedule pre-application meetings, and plan resources for the upcoming licensing action. In some cases, the NRC staff may also determine that a site visit is warranted to better understand the proposed facility or technology.

Following this initial notification, applicants may choose to enter a period of pre-application engagement designed to promote early, transparent communication. Through pre-application meetings and white papers, applicants can outline their licensing strategy, describe their technologies, processes, and safety approaches, and receive feedback from the NRC staff. Applicants that choose to submit topical reports can also receive early NRC review and regulatory certainty for key topics related to their design. NRC staff use these pre-application interactions to provide early feedback on potential safety or regulatory issues and inform license review schedules. Applicants may also request that NRC staff perform a readiness assessment of the draft application, to receive feedback on items that might affect application acceptance, generate requests for additional information, or have an impact on the efficiency of the review.

Once ready, applicants submit their license applications. The quality of an application has a significant impact on the resources expended in the review process, and applicants should ensure they include complete and high-quality administrative and technical information consistent with applicable content of application guidance. The NRC staff then conducts an acceptance review, assessing whether the application is sufficiently complete to support a full technical review. If acceptable, the NRC issues an acceptance letter that formally docketed the application, outlines the proposed review schedule and resource estimates, and may include preliminary observations on areas of the application that will likely require additional engagement with the applicant—such as through audits—during the NRC’s review.

With the application docketed, NRC staff perform the detailed technical review, generally supported by one or more audits. A regulatory audit allows the NRC staff to conduct its review more efficiently and gain insights on the applicant’s programs or processes by verifying information provided on the docket and identifying additional information that needs to be docketed to satisfy the applicable requirements. Throughout the formal license review process, the NRC may issue requests for additional information (RAI), which allows staff to resolve safety or environmental issues on the docket that are not fully addressed in a license application. An applicant typically has 30 calendar days from the issuance date to respond to an RAI.

Technical reviews culminate in the NRC staff producing a safety evaluation report (SER), which documents the technical and regulatory basis for approval or denial of the requested action. The SER documents the NRC staff review of the applicant’s safety analysis report, including conclusions that proposed nuclear plants will sufficiently protect public health and safety. It includes evaluations of design criteria, safety features, and necessary modifications to address any inadequacies identified during the review process. Information submitted by applicants and information developed by the NRC staff, including the safety evaluations, is publicly available to the extent possible

¹ The NRC staff create a unique docket number to track specific nuclear power plants, materials facilities, or commercial activities. The first three digits correspond to the relevant part of Title 10 of the Code of Federal Regulations (10 CFR).

considering proprietary and other types of information restricted from public access in accordance with 10 CFR 2.390, “Public inspections, exemptions, requests for withholding.”

For CPs, OL, COLs, and MLs, hearings—whether mandatory or contested—provide opportunities for affected parties to participate in the process. After safety, environmental, and hearing requirements are satisfied, the NRC issues its final licensing decision and publishes a notice of issuance on its public website and in the *Federal Register*.

Additionally, in accordance with Sections 29 and 182b of the Atomic Energy Act (42 U.S.C. 2039, 2232(b)), the Advisory Committee on Reactor Safeguards (ACRS) shall “advise the Commission with regard to the hazards of proposed or existing reactor facilities and the adequacy of proposed safety standards.”

As part of its implementation of EO 14300; the Accelerating Deployment of Versatile, Advanced Nuclear for Clean Energy Act of 2024 (ADVANCE Act); and the Nuclear Energy Innovation and Modernization Act (NEIMA), the NRC updated the NEIMA milestone schedules for issuance of the final safety evaluations of CPs, OLs, and COLs to 18 months.

DOE Authorization Process

Standard DOE-STD-1271-2026 provides guidance for authorization of nuclear facilities by the DOE’s Office of Nuclear Energy (NE). The standard identifies applicable authorization requirements and provides a methodology for meeting those requirements for NE authorized facilities. All DOE Hazard Category 1-3² facilities are required to meet 10 CFR 830 Subpart B, “Nuclear Safety Management,” requirements, which defines requirements for documented safety analyses, technical safety requirements, and DOE reviews and approvals. DOE developed orders, additional standards, and guides to translate the 10 CFR 830 requirements into policies, procedures, and tools for facility safety, operations, and quality assurance³.

The DOE authorization process for nuclear facilities is structured around a series of phases designed to ensure safe and efficient development of advanced nuclear technologies. It begins with early alignment between DOE and the contractor through formal contracts. Next, both parties negotiate expectations, responsibilities and regulatory requirements (including potential exemptions to DOE requirements). A key product of this phase—the Nuclear Safety Design Agreement (NSDA)—lays the foundation for how safety will be analyzed, documented, and integrated into the evolving design. The NSDA lays the foundational relationship with the contractor and documents design requirements, safety analysis approach, regulatory engagement process, key safety decisions, and Code of Record. The Code of Record details the applicability of consensus codes and standards to the design at a system level, addresses the potential need for code variances, equivalencies, and exemptions, and includes references to technical evaluations demonstrating that any alternative approaches provide equivalent or better safety compared to the applicable code. The Code of Record continues to be matured and developed as the DOE authorization process progresses and supports other safety analysis documentation.

As the project moves into preliminary design, the contractor develops and documents system concepts, hazard analyses, and early performance assumptions. Once the design reaches approximately 50% maturity as determined by DOE, the contractor submits the Preliminary

² NE-STD-1027-2025, “Hazard Categorization of DOE Nuclear Facilities,” defines Hazard Category 1 as showing the potential for significant off-site consequences, Hazard Category 2 as showing the potential for significant on-site consequences, and Hazard Category 3 as showing the potential for significant localized consequences.

³ It should be noted that DOE Orders (DOE-O) and DOE Policies (DOE-P) are supplements to DOE regulations and the contractors must meet the requirements. DOE standards are similar to NRC regulatory guides (RGs) and standard review plans; however, they are not optional in the sense NRC RGs are.

Documented Safety Analysis (PDSA). The PDSA represents DOE's first opportunity to formally review the developing safety case in an integrated manner. Approval of the PDSA by DOE reflects confidence that the proposed approach is reasonable, establishes regulatory certainty on the proposed approach, and supports subsequent procurement actions by clearly establishing performance requirements on key SSCs necessary to ensure safety of the overall facility.

As system and component designs are finalized, verified, and placed under configuration control, the contractor submits the Final Documented Safety Analysis (DSA). The purpose of final DSA is to establish the safety basis and operating controls necessary to safely operate the facility and validate that the identified design has met those criteria. Notably, authorization applicants may begin construction at risk before the DSA is approved.

At each of these stages, DOE conducts structured reviews of contractor submittals. The DOE Review Team has a goal to complete its review in 90 days following formal acceptance of the NSDA, PDSA, or DSA. During these reviews, DOE staff focus comments on issues affecting worker or public safety. To promote efficient and timely reviews, draft comments are provided to the contractor for early resolution. The contractor is encouraged to meet with the DOE Review Team frequently to discuss the status of the design and future operations. Similar to the NRC, DOE documents its basis for approval in a safety evaluation report; however, the applicant submittals and DOE safety evaluation and review team summaries are generally nonpublic due to security of research of the proprietary nature of the facility.

After the DSA is approved, the facility enters the startup phase, beginning with a DOE Readiness Review that verifies operational readiness through inspection of procedures, verification of equipment operability, safety management programs and DSA implementation, emergency management drills, and staff training. Successful completion results in Startup Approval, allowing the facility to proceed into initial commissioning under a Joint Test Group (JTG) and ultimately to full operations under the approved safety basis. The JTG is responsible for overseeing the initial startup testing, which includes verifying that all systems and components function as intended and meet safety requirements and that the key behavioral predictions in safety analyses are correct as demonstrated in the actual operating plant.

Key Takeaways on the NRC Licensing and DOE Authorization Processes

Both NRC and DOE processes ensure safety and security through the licensing or authorization of nuclear reactor facilities, consistent with their missions.

One key distinction between DOE authorization and oversight and NRC licensing and oversight is that DOE, as the grantor of the contract with an applicant, is directly responsible for the safety of its facilities. In contrast, NRC license holders have full responsibility for ensuring the safe design, construction, and operation of their facilities, subject to NRC inspection and oversight. This distinction impacts the manner in which NRC and DOE interacts with applicants to ensure the shared goal of safe and secure technology deployment.

DOE reactor authorizations (under the Reactor Pilot Program and Launchpad) are also generally limited in scope and duration to support specific technology demonstration while initial NRC commercial licenses generally involve full scope operation over a longer period (up to 40 years). By nature of these differences, a DOE authorization may consider a subset of the topics that would be relevant for an application for an NRC license for a commercial facility. For example, a short-duration DOE authorization may not require significant consideration of passive materials performance, while an application for a longer-term commercial NRC license would need to more thoroughly consider and address passive materials aging and the potential for safety functions to be ensured over time.

The NRC encourages developers to begin engagement early to discuss their intended deployment approaches and the information from a DOE authorization they plan to reference in a future NRC license application.

DISCUSSION

The NRC's existing guidance may be used to review applications that reference a DOE or DOW authorization. Applicants who reference a DOE or DOW authorization when demonstrating compliance with applicable regulations, may credit prior analyses, testing, prototype operation, and operating experience, as appropriate. The NRC notes that the extent to which prior DOE or DOW information can be leveraged depends on the information an applicant provides and the associated justification as to how it meets NRC requirements. For example, the scope of a DOE or DOW authorization or associated testing may not fully represent a commercial reactor configuration or the breadth of analyses necessary for an NRC licensing basis. To support NRC staff in leveraging the maximum amount of information from a previous DOE or DOW authorization, the guidance below emphasizes the importance of developers and industry stakeholders "having the end in mind" when pursuing authorization—namely, ensuring that the technical basis and regulatory analyses developed for DOE or DOW approval are aligned, to the extent practical, with those that will ultimately be required for an NRC license application. Early and sustained preapplication engagement with the NRC staff further helps applicants understand where DOE or DOW information overlaps with commercial licensing requirements and where additional information may be needed to address differences. Additionally, applicants may, as an option, provide an equivalency map illustrating NRC regulations that may be satisfied by prior authorization information to aid in the NRC staff's review. The NRC staff may consider leveraging when (1) the relevant design feature(s) and safety function(s) are materially similar to those previously authorized, and (2) the NRC determines that the prior assessment remains applicable and technically valid to the subject application.

Types of NRC Leveraging of DOE Information

A key to establishing an expedited pathway to approve reactor designs previously authorized by DOE is effectively leveraging applicable information generated from these prior reviews and demonstrations. The NRC staff can use that information in two ways:

- Tailoring the scope and depth of the NRC staff review; and
- Developing NRC technical and regulatory findings informed by DOE authorization findings.

First, the NRC staff consistently seeks to be risk-informed in its licensing reviews by focusing review effort on areas of greatest risk and safety significance. Leveraging information from prior DOE-authorized and tested facilities enables further tailoring of NRC review scope and depth. The NRC staff will also consider what aspects of the facility have already been evaluated in the prior DOE authorization and otherwise demonstrated to operate safely. Therefore, in general, the NRC staff should seek all opportunities to leverage information from prior authorization reviews and demonstrations to appropriately focus on areas of greatest risk and safety significance. The extent of NRC staff review on a given topic depends on the information from the prior DOE authorization demonstrated to be applicable by the applicant. The guidance below provides additional detail on how the NRC staff will tailor its review generically and in selected topical areas covered by the appendices.

Second, findings from prior DOE reviews may be directly leveraged, in full or in part, to support NRC technical and regulatory findings. Although the NRC staff generally considers information and analyses referenced by an applicant from other entities (including DOE), direct leveraging of DOE findings would be a more novel approach relative to past NRC practice. This may be appropriate when the applicability of the DOE review and demonstration to the NRC license requirement(s) is

clearly established and the basis for the prior DOE review findings is clearly documented. For each review of a commercial license application referencing a DOE authorization, the NRC staff should consider, consistent with factors described below, how the NRC staff may leverage or build on the basis for prior DOE findings. The NRC staff should focus on the information provided by the applicant to justify the relevance of information from its prior DOE authorization for addressing NRC requirements.

In some cases where prior review applicability and similarity between NRC and DOE requirements is high, the NRC staff review may directly leverage, with appropriate confirmation based on its independent review, the same basis for certain NRC technical and regulatory NRC findings as the prior DOE review. In other cases, more in-depth staff evaluation may be warranted to appropriately verify the applicability of authorization information referenced and to confirm that the additional information supports the basis needed to satisfy the NRC requirements. Application of the DOE graded approach (10 CFR 830.7) should also be considered to determine if the level of analysis, documentation, and actions used to meet DOE requirements is sufficient to meet NRC requirements. Finally, there may be limited leveraging in cases where the basis for the prior DOE review findings for a certain topical area is not clearly documented or the relevance of the DOE review and demonstration is not clearly established by the applicant.

Types of DOE-authorized Facilities

From an NRC licensing perspective, there are two general categories of facilities seeking DOE authorization:

- Limited-scope demonstrations, which may be either:
 - Zero or low-power criticality tests or
 - Reactor facilities operated at appreciable power levels to lay the foundation for a future NRC license application, and
- Full-scale commercial design demonstrations planning to transition the as-built DOE-authorized facility to an NRC license in the future.

Information from the approved design from either type of demonstration could be used in future NRC licensing reviews; however, how that information is leveraged differs.

Limited-scope demonstrations are similar to how NRC reviews have traditionally considered and leveraged prior research and demonstration work in specific topical areas, such as fuel and materials qualification. The NRC staff expects the information associated with limited-scope demonstrations to generally include data generated from the testing or operations that can inform NRC licensing.

The NRC staff anticipates that full-scale commercial design demonstrations will provide greater potential for leveraging because the reactor designs and scope of their operation are expected to be similar to that of commercial reactors. There may be applicable reactor operation and testing experience to demonstrate safety as well, particularly in areas where the DOE review scope, depth, and documentation and the relevance of the DOE review and demonstration to the NRC license review are clearly established.

The guidance below provides additional detail on how the NRC staff will do this generically and in selected topical areas covered by the appendices.

Leveraging Factors and Degree of Leveraging

Given the broad range of technologies and designs being proposed, the NRC staff will need to apply judgement on an application-specific and topical area-specific basis for how information from DOE authorizations can be appropriately leveraged. As noted in the “Overview of NRC Licensing and

DOE Authorization Processes” section above, there are important differences in the scope and purpose of a DOE authorization review compared to an NRC license review. Key differences include but are not limited to:

- To date, facilities authorized by DOE under the [Reactor Pilot](#) and [Launchpad](#) Programs have generally been for limited-duration, limited-scope operations, which do not involve consideration of longer-term operation, such as a 40-year NRC license term. For instance, in some cases, the planned DOE project duration may limit the fuel burnup and amount of fission products produced. This most directly impacts considerations associated with materials and component aging and the need for inspection, monitoring, and maintenance programs to ensure safety functions can be performed through the life of the facility. This shorter time period may also influence considerations associated with risk over time as a longer time period of operation may require a different standard or approach to appropriately manage or mitigate relevant hazards. Reactor performance and potential degradation will need to be considered over the full NRC license period.
- DOE PDSAs and DSAs are required to establish nuclear safety design criteria, which are somewhat analogous to principal design criteria (PDC) for NRC licensing. However, DOE PDSAs and DSAs may not contain the same explicit mapping of PDC to design requirements and design bases for specific SSCs fulfilling those PDC as are required for NRC licensing. DOE contractors are allowed some flexibility in their approach to mapping PDC. Similarly, the level of detail and specificity of design information for NRC licensing may need to be increased for a commercial license application where the reactor design is more complex and the overall facility hazard is different than that considered for the DOE authorization.

Table 1, below, describes five key factors to be considered in assessing the degree of leveraging. A similar table appears in each topical area appendix to provide guidance on how the first three factors in Table 1 will be considered for leveraging within that topical area. The final two factors are more dependent on the review performed by DOE and the type of documentation generated from the prior review that may be leveraged and are therefore not discussed in each appendix.

Table 1: Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	Areas where NRC and DOE requirements and guidance are more similar or overlapping provide greater potential for NRC leveraging.
Application quality and justification of relevance of DOE authorization to NRC review	Areas where the application content is thorough and complete and the applicant has clearly justified or demonstrated the relevance of the DOE authorization to the NRC review provide greater potential for NRC leveraging. Operational data generated from limited-scope demonstrations should be appropriately considered for relevance to the design to be licensed with NRC.
Safety-significance of information leveraged ⁴	NRC frameworks generally provide greater flexibility in areas of lower safety-significance and involve greater regulatory

⁴ For programmatic aspects of the licensing basis, there may be cases where a program applies to SSCs of varying safety or security significance. In those cases, this factor should not be weighed heavily in determining how DOE authorization information can be leveraged, especially when other considerations—such as alignment with NRC processes, documentation quality, or regulatory independence—play a more influential role.

	controls in areas of higher safety-significance. Therefore, information that is less significant to plant safety may have greater potential for NRC leveraging, while information that is more significant to plant safety should be a greater focus for the NRC staff review. Information used to establish the degree of safety-significance (e.g., design criteria, safety analysis methodologies) should be considered carefully for NRC leveraging.
Similarity of the scope and depth of DOE and NRC reviews	Areas where the scope and depth of DOE's review are similar to NRC's review (including with confirmatory technical basis work in some cases) provide greater potential for NRC leveraging.
Documentation of basis for DOE authorization in NRC license application	Areas where the DOE authorization review findings and basis are clearly documented provide greater potential for NRC leveraging. This would increase NRC staff confidence in the basis for DOE's finding and allow NRC staff to leverage the DOE authorization to the appropriate extent as relevant to the NRC review.

The NRC staff should use the factors in Table 1 on an application-specific and topical area-specific basis to holistically assess, as appropriate, the information from a DOE authorization to be leveraged. These factors are qualitative and intended to promote consistency and clarity in NRC approaches to leveraging of DOE information, while enabling the NRC staff to apply appropriate judgement to a given licensing review.

NRC Staff Approach to Leveraging DOE Information

NRC staff should consider the factors identified in Table 1 above to determine the degree of leveraging that is appropriate. While recognizing that DOE authorization decisions and NRC regulatory requirements serve different purposes, this approach allows NRC staff to leverage relevant DOE safety reviews when technically justified. It is the applicant's responsibility to provide enough information to demonstrate that the planned facility satisfies all relevant NRC regulations.

NRC leveraging of DOE information can generally be organized into three categories:

Maximum leveraging:

- Technical information developed or evaluated as part of a DOE authorization may provide a direct basis for leveraging in an NRC review when the underlying information remains valid and the relevant facility design features and hazards are sufficiently comparable. The NRC staff would perform a review to the level of scope and depth necessary to verify the findings and basis from the prior DOE authorization are applicable and appropriate to satisfy relevant NRC requirements. The potential for direct leveraging is greatest when the information addresses a technical attribute or analysis that is not dependent on DOE-specific regulatory requirements or acceptance criteria.
- Expected to lead to significant efficiency gains for areas where most or all of the factors in Table 1 support significant NRC leveraging.

Moderate leveraging:

- Under a moderate leveraging scenario based on the factors above, certain DOE information may be technically useful but require additional detailed evaluation before it can support findings by the NRC staff. For these areas, NRC staff should compare the DOE

methodology, assumptions, design features, design criteria, and performance or acceptance criteria to those applicable to the NRC-regulated nuclear facility. Where differences exist, the applicant should identify those differences and provide additional analysis or justification as necessary to establish applicability to the NRC licensing basis.

- Expected to lead to moderate efficiency gains for areas where the factors in Table 1 indicate moderate support for NRC leveraging.

Limited leveraging:

- Under a limited leveraging scenario based on the factors above, NRC staff could use the DOE information as background and supplemental technical information to support its review and evaluation. The DOE information would inform the NRC's review but would not be a significant part of the technical basis for the NRC's safety determination. Such information may nevertheless be leveraged to the extent that it demonstrates relevant technical characteristics or provides a sound technical basis for NRC evaluation. Where DOE requirements or determinations are relied upon, the applicant should identify the applicable DOE and NRC requirements, explain the relationship between them, and provide sufficient information to demonstrate that the applicable NRC requirement is satisfied.
- Expected to lead to limited efficiency gains for areas where few or none of the factors in Table 1 support NRC leveraging.

Considerations for Applicants when Preparing an Application Based on a Prior DOE Authorization

Early Engagement and Pre-application

Industry stakeholders planning to pursue significant leveraging of prior DOE authorization are encouraged to have the "end in mind" when developing the technical and applicable regulatory basis necessary to receive DOE authorization to construct and operate the reactor facility. Although the DOE would have jurisdiction over the facilities with respect to the specific DOE authorization, there will likely be many opportunities for industry applicants to consider related NRC requirements and implementing guidance when developing their respective DOE authorization application content. Considering those NRC requirements when developing a DOE application is expected to result in efficiency gains when a follow-on NRC application is developed and submitted for NRC review.

The benefits of this early engagement align well with NRC's establishment and implementation of the "core team"⁵ application review approach. The use of core teams has resulted in efficiency gains in the NRC's licensing reviews of new reactor facilities, including reductions in both application review duration and review resources. Early engagement further enables the NRC to better leverage directly applicable information from DOE reviews. This engagement is expected to facilitate staff familiarity with the DOE or DOW review scope, depth, and documentation and identify areas where more detailed discussion of how the DOE authorization may be leveraged in an NRC application.

Similarly, industry stakeholders are encouraged to engage in preapplication interactions with the NRC to facilitate mutual understanding of the application that is either under review by DOE or that was previously reviewed by DOE. Such interactions aid the NRC staff and applicants in understanding whether and to what extent attributes of the prior DOE authorization may be

⁵ Core teams support the reviews of applications in a holistic, risk-informed manner. These core teams rely on a small number of staff who are dedicated full-time (or significant part-time) to each review. Supplemental subject matter experts support the core team through well-defined, discrete assignments. A project charter defines this project structure and staffing philosophy at the outset of each review. Each charter identifies the core team members and subject matter experts needed to carry out the major activities involved in each review. Roles and responsibilities are also outlined to ensure that staff and management involved in each review clearly understand the expectations for the project.

leveraged in an NRC review. The NRC staff has performed observation audits of the DOE authorization reviews for some vendors. These observations have been beneficial for NRC staff and the vendors to enable early identification and engagement on key areas where additional information may be needed for NRC review.

Prospective NRC applicants are encouraged to establish early preapplication engagements with the NRC to communicate their plan for addressing these and other key topics associated with the development of a complete NRC application. This early preapplication engagement, preferably performed in parallel with the DOE authorization effort preceding the planned follow-on NRC licensing action, is key to facilitating the efficient leveraging of those prior authorizations. Additional information on the preapplication and NRC licensing processes can be found on the NRC's website "Guidance for Prospective Applicants" (<https://www.nrc.gov/reactors/new-reactors/advanced/new-app/general-guidance>).

Prospective applicants are also encouraged to consider other NRC application process requirements, including those in 10 CFR Part 2, "Agency Rules of Practice and Procedure". Associated information on the NRC's adjudications (hearings) processes, including opportunities for public involvement, can be found on the agency's website (<https://www.nrc.gov/about-nrc/regulatory/adjudicatory>).

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DOE requirements, standards, criteria, and previously approved authorization information should be used as supporting technical bases or evidence where appropriate and should not be treated as automatic substitutes for NRC regulations or NRC licensing findings.

The applicant should provide a justification of the applicability of the prior DOE authorization information to the NRC license application by addressing:

1. The specific DOE documented authorization information being relied upon;
2. The DOE requirement, standard, or criterion under which the information was developed;
3. What DOE reviewed, approved, or demonstrated;
4. The design, site, source term, and configuration to which the information applies, including changes from the DOE-authorized facility;
5. The corresponding NRC requirement or review criterion; and
6. Any difference or gap between the DOE requirement, standard, criterion, or prior DOE determination and the corresponding NRC requirement or review criterion that requires supplementation.

NRC Leveraging at Different Stages of the DOE Authorization Process

Based on NRC staff experience as detailees to DOE and NRC staff observation audits of DOE authorization reviews, the NRC staff have noted the following regarding the potential for leveraging and NRC preapplication engagement at different stages of the DOE authorization process:

- The NSDA stage generally provides very high-level design information that is of limited value for leveraging in an NRC application.
 - It may be prudent to consider engaging NRC staff for pre-application audits to begin to familiarize the NRC staff with the design and regulatory approach, including potential exemptions from DOE requirements.
- The PDSA stage for the DOE authorization process represents the point at which the DOE determines the design is approximately 50% complete. While the DOE PDSA and NRC PSAR both represent a preliminary level of design maturity, the NRC staff has noted through DOE authorization observations that there may be significant differences between the scope

and maturity of information provided in a DOE PDSA and that required for an NRC PSAR under 10 CFR Parts 50 and 53. For example, NRC regulations require a complete preliminary design where all aspects of the design provide a preliminary level of design information. The DOE authorization PDSA expectation for design maturity does not necessarily require such a level of design maturity across the complete design.

- If there are plans to leverage the content of a DOE PDSA for an NRC PSAR in a CP application, the applicant is strongly encouraged to engage NRC staff to observe the DOE PDSA review to enable the NRC staff to be familiar with the DOE review and thereby able to maximize leveraging of the DOE PDSA review. The NRC staff and preapplicant should be mindful of the differences noted here between a DOE PDSA and NRC PSAR as well as any additional observations the NRC staff provide through the observation audit for their specific application.
- The DSA stage of the DOE authorization process contains the final design of the facility and serves to validate that the as-built facility meets the safety basis required for operations. In general, because the DSA represents a more mature final design, the NRC staff anticipates that information from a DSA should generally be able to be leveraged to a greater degree than from an NSDA or PDSA.⁶
 - Similar to the PDSA stage, an applicant planning to leverage from a DOE DSA is encouraged to engage NRC staff to observe the DOE DSA review to enable the NRC staff to be familiar with the design and the DOE review. The applicant should consider observations the NRC staff provide through the observation audit for their specific application.
- Finally, the NRC staff expects some applicants to seek to leverage information from - authorized facilities after some period of operation to demonstrate safety. At least initially, the NRC staff expects that the facilities providing operational data to be limited-scope demonstration facilities. Therefore, an important consideration is for the applicant to clearly establish the relevance of the data generated to the NRC license application.
 - An applicant planning to leverage information from an operated DOE facility is further encouraged to engage NRC staff to observe the DOE authorization process to enable the NRC staff to become familiar with the design and the scope of the planned DOE-authorized operations. The applicant should consider feedback the NRC staff provides throughout the observation of the authorization process.
 - Applicants wishing to transition their operational facility from a DOE authorization to an NRC commercial license should be aware of unique considerations associated with verifying the quality of construction activities. The NRC is required to make a finding of reasonable assurance that a facility was constructed and will operate in accordance with NRC requirements. However, the NRC has no regulatory authority in the oversight of a facility's construction or testing under DOE's authorization, and, therefore, the NRC cannot conduct the inspection activities necessary to support a finding that a facility is constructed and will operate in accordance with its licensing basis. To perform that review for an applicant requesting a commercial NRC license for a facility constructed under the DOE authorization process, the NRC staff intends to verify the quality of construction activities through a process described in greater detail in letters issued by NRC to prospective NRC reactor license applicants (ML26268A085, ML26259A330).

⁶ The NRC staff has more limited experience to-date with the level of detail and contents of DOE DSAs for the types of facilities intended for NRC licensing. Most of the DOE DSAs to date have been for limited-scope demonstration facilities with very short and limited scope operations planned.

Topic-Specific Appendices

This paper includes a series of appendices on selected NRC licensing topical areas. These appendices are not intended to reflect all topics required to be addressed in an application for an NRC license. However, these topics generally form much of the foundation for the assessment of reactor facility risk and documentation of the facility's associated safety basis, to be reflected in the SAR portion of a license application. These topics are also anticipated to overlap many of the requirements to receive an authorization to construct and operate under DOE authorities. The appendices are as follows:

- A. Generic Content of Application Requirements and Guidance
- B. Quality Assurance Program
- C. Licensing Event Selection and Classification of Structures, Systems, and Components
- D. Principal Design Criteria
- E. Source Term
- F. Consequence Analysis
- G. Emergency Preparedness
- H. Security
- I. Fire Protection
- J. Technical Specifications
- K. Site Characterization and Design for External Hazards
- L. Fuel Performance
- M. Analytical Methods Development, Verification, and Validation
- N. Materials Performance
- O. Codes and Standards
- P. Instrumentation and Controls
- Q. Cybersecurity

APPENDIX A—Generic Content of Application Requirements and Guidance

The purpose of this appendix is to identify key generic content of application guidance and requirements for different technologies and NRC licensing frameworks as well as the DOE authorization process. In the development of applications for NRC licenses, certifications, permits, and approvals, an applicant must provide sufficient information to demonstrate that they meet applicable regulatory requirements. The information and level of detail required varies according to the design and type of application sought (e.g., Construction Permit (CP), Standard Design Approval (SDA), Design Certification (DC), Early Site Permit (ESP), Manufacturing License (ML), Operating License (OL), Combined License (COL)). The content of the application and the NRC staff's review approach are further defined by the reactor technology, the design and licensing approach selected by the applicant, and the extent to which the applicant proposes to reference a prior DOE authorization of the same or similar design. The NRC provides a variety of licensing frameworks to support efficient licensing of a broad range of reactor designs.

1. NRC Content of Application Regulatory Requirements

10 CFR Part 50

Title 10 of the *Code of Federal Regulations* (10 CFR) 50, "Domestic Licensing of Production and Utilization Facilities," provides a two-step licensing process for commercial power reactors. For applicants using the 10 CFR Part 50 process, application requirements include those in 10 CFR 50.34, "Contents of applications; technical information." 10 CFR 50.34(a) contains requirements for a preliminary safety analysis report (PSAR) in support of a CP application. Meanwhile, 10 CFR 50.34(b) contains requirements for a final safety analysis report (FSAR) in support of an OL application.

10 CFR Part 52

10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," provides for a one-step licensing process based on a COL application. For applicants using the 10 CFR Part 52 process, application requirements are provided, in part, in 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report." Additionally, the application requirements under the Part 52 process for standard ESPs, DCs, SDAs, and MLs include those in 10 CFR 52.17, 52.47, 52.137, and 52.157, respectively.

10 CFR Part 53

A risk-informed, performance-based, and technology-inclusive regulatory framework for commercial advanced nuclear reactors is provided under 10 CFR Part 53, which incorporates expanded use of risk-informed evaluations, enhanced operational flexibility, streamlined siting and manufacturing processes, and graded security and code requirements. For applicants using 10 CFR Part 53, application requirements include those in 10 CFR 53.1306 and 1309 for CPs, 10 CFR 53.1366 and 53.1369 for OLs, and 10 CFR 53.1413 and 53.1416 for COLs. Additionally, the application requirements under the Part 53 process for standard ESPs, DCs, SDAs, and MLs include those in §§ 53.1144 and 53.1146, §§ 53.1236 and 53.1239, §§ 53.1206 and 53.1209, and §§ 53.1276 and 53.1279, respectively.

Proposed 10 CFR Part 57

A streamlined, risk-informed, and performance-based framework for microreactors and other reactors with comparable risk profiles would be provided under proposed 10 CFR Part 57 (91 FR 23628, Docket ID NRC-2025-0379). Proposed 10 CFR Part 57 would provide for expedited review and licensing timelines, use of generic finality supporting standardized deployments, large geographical areas for siting, pathways for fabrication and assembly of components of a

manufactured reactor, transport allowances, and commensurate safety and operational requirements. Under proposed 10 CFR Part 57, an applicant would be required to include the information specified in proposed 10 CFR 57.60 in a joint application for a CP and associated OL(s) based on a final design.

2. NRC Guidance and Related References

The following NRC guidance documents and references provide generic guidance on developing applications across different NRC licensing frameworks:

LWR Technologies Using a Predominantly Deterministic Approach

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)” (ML070810350), provides guidance to NRC staff in performing safety reviews of CP or OL applications, including requests for amendments under 10 CFR Part 50, and ESP, DC, COL, SDA, or ML applications under 10 CFR Part 52. The principal purpose of NUREG-0800 is to ensure the quality and uniformity of staff safety reviews. The guidance is also intended to make information about regulatory matters widely available and to improve communication among the NRC, interested members of the public, and the nuclear power industry, thereby increasing understanding of the NRC’s review process. Though direct applicability is limited to light water reactors (LWRs), it provides detailed, hazard specific guidance that may apply more generically as a baseline to safety reviews of applications.
- Regulatory Guide 1.70 Revision 3, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition” (ML011340122), provides a standard format and content for LWR applications.
- Regulatory Guide 1.206, Revision 1, “Applications for Nuclear Power Plants” (ML18131A181) was revised in 2018 with a new title, “Applications for Nuclear Power Plants.” This RG is specifically applicable to power reactors using LWR technology, with the clarification that the NRC staff also considers it to be generally applicable to other types of reactors (e.g., non-LWRs).

Non-LWR Technologies Using a Risk-Informed and Performance-Based (Licensing Modernization Project) Approach

- RG 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors” (ML20091L698), contains the NRC’s endorsement of the Licensing Modernization Project (LMP) methodology in Nuclear Energy Institute (NEI) 18-04, Revision 1, “Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development” (ML19241A472), for selecting licensing-basis events (LBEs), classifying structures, systems, and components (SSCs), and assessing the adequacy of defense in depth (DID).
- RG 1.253, “Guidance for a Technology-Inclusive Content of Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approval for Non-Light-Water Reactors” (ML23269A222), provides guidance for developing specific portions of the safety analysis report for non-LWR applications using the LMP methodology. This RG endorses the methodology described in NEI 21-07, Revision 1, “Technology Inclusive Guidance for Non-Light Water Reactors, Safety Analysis Report Content: For Applicants Using the NEI 18-04 Methodology” (ML21250A378).
 - DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications – Roadmap” (ML23277A139), provides additional guidance for

the preparation and review of non-LWR applications using the LMP process. The roadmap is accompanied by specific topical area guidance under the Advanced Reactor Content of Application Project (ARCAP) (ML23277A105):

- DANU-ISG-2022-02 ARCAP Chapter 2 Site Information.
- DANU-ISG-2022-03 ARCAP Chapter 9 Normal Effluents.
- DANU-ISG-2022-04 ARCAP Chapter 10 Occupational Dose.
- DANU-ISG-2022-05 ARCAP Chapter 11 Organization and Human-System Consideration.
- DANU-ISG-2022-06 ARCAP Chapter 12 Post-construction Inspection, Testing, and Analysis Program.
- DANU-ISG-2022-07 ARCAP Risk-informed Inservice Inspection and Inservice Testing.
- DANU-ISG-2022-08 ARCAP Risk-informed Technical Specifications.
- DANU-ISG-2022-09 ARCAP Fire Protection for Operations.

Non-power Reactor Facilities under 10 CFR Part 50

- NUREG-1537, Part 1, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content” (ML042430055), provides guidance to non-power reactor licensees and applicants on the format and content of applications to the NRC for licensing actions. These licensing actions include CPs and initial OLs, license renewals, amendments, conversions from highly enriched uranium to low-enriched uranium, decommissioning, and license termination.
- NUREG-1537, Part 2, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria” (ML042430048), provides guidance to the NRC staff on the conduct of licensing reviews for non-power reactor applications.

Proposed 10 CFR Part 57 Approach

- Proposed NUREG-2271, “Guidelines for Preparing and Reviewing Applications Under 10 CFR Part 57,” provides guidance to support rapid licensing and high-volume deployment of microreactors and other reactors with comparable risk profiles. NUREG-2271 provides applicants direction on the information to include in an application and provides NRC staff with guidance for conducting their reviews. The guidance also describes the licensing pathways proposed under Part 57, which include SDAs, MLs, CPs, and OLs.

3. Summary of DOE Requirements, Orders, and Standards

- 10 CFR 830, “Nuclear Safety Management,” establishes the core regulatory framework for DOE nuclear facilities. Part 830 requires the development of a Documented Safety Analysis (DSA) and Technical Safety Requirements (TSRs) that demonstrate adequate protection of workers, the public, and the environment.
 - 10 CFR 830, Subpart A, “Quality Assurance Requirements,” establishes requirements for a Quality Assurance Program (QAP) for DOE nuclear facilities, including management, design, procurement, work processes, inspection, testing, and assessments. NE O 414.1, “Quality Assurance,” establishes additional requirements for developing, implementing, and maintaining quality assurance programs for facilities and activities authorized by DOE’s Office of Nuclear Energy, including provisions for software quality assurance and the application of a graded approach.

- Appendix A to Subpart B to Part 830, “General Statement of Safety Basis Policy,” sets forth acceptable methodologies for preparing a DSA⁷:
 - For a reactor—Using the method in U.S. Nuclear Regulatory Commission Regulatory Guide 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,” or successor document, such as DOE-STD-1237-2021, “Documented Safety Analysis for DOE Reactor Facilities”.
 - For a nonreactor nuclear facility—Using the method in DOE-STD-3009, Change Notice No. 1, January 2000, “Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports,” July 1994, or successor document.
- 10 CFR 830.7, “Graded approach,” provides a graded approach for meeting Part 830 requirements, provided that the basis for the approach is documented, and the documentation is submitted to DOE. However, the graded approach may not be used in implementing the unreviewed safety question (USQ) process or technical safety requirements (TSRs).
- DOE requires a Worker Safety and Health (WSH) Program that meets the requirements of 10 CFR Part 851, “Worker Safety and Health Program.” The WSH Program addresses general industrial safety hazards and includes provisions for fire prevention, fire response, worker training, and other occupational safety and health activities. DOE approved a variance (No. 1037) in January 2026 for Office of Nuclear Energy advanced reactor contractors that provides alternative implementation provisions for specified requirements of Part 851. The variance is subject to applicable conditions, and requirements not covered by the variance remain in effect.
- DOE P 420.2, “Nuclear Safety Policy for Onsite Nuclear Facilities,” and P 420.3, “Nuclear Safety Policy for Offsite Nuclear Facilities,” establishes the DOE nuclear safety policy for nuclear facilities authorized by the Office of Nuclear Energy (NE). This policy establishes a quantitative safety objective and design goal as “aiming points” for safety goals that guide the development of NE’s nuclear safety directives and technical standards. The policy states: “The design goal for new nuclear facilities is that the worst-case design basis accident (greater than 10E-6 events/year) will be 5 rem at the facility’s public access boundary.”
- DOE-STD-1271-2026, “Authorization Pathway for Nuclear Facilities,” implements DOE Policy P 420.2 and P 420.3 and provides a pathway for the authorization by NE of nuclear facilities. This Standard specifies the requirements and responsibilities for engineering/design, safety analysis, operations and interactions essential for the safe design, construction and operation of nuclear facilities, including test reactors, fuel manufacturing and recycling facilities, and major modifications. The Standard also identifies key considerations for the integration of safety into design.
- DOE-STD-1237-2021, “Documented Safety Analysis for DOE Reactor Facilities” provides an acceptable methodology for preparing a DSA for DOE reactor facilities. It supports selection and documentation of a reactor-appropriate safety-analysis approach and may use or adapt methods from RG 1.70, NUREG-1537, ANSI/ANS-15.21, DOE-STD-3009-2014, NEI 18-04, or another DOE-approved methodology, as applicable to the reactor and the approved

⁷ These methodologies are not required to be utilized. DOE-STD-1271-2026, “Authorization Pathway for Nuclear Facilities,” provides additional detail on the approach for format and content of the Preliminary Documented Safety Analysis and the DSA which may include the application of existing standards or development of an individualized approach.

safety-basis strategy. Additionally, the reactor developer may propose an alternate methodology, with DOE approval.

- NE Order (O) 420.1, “Facility Safety,” establish facility and programmatic safety requirements for the DOE NE, including advanced nuclear reactors and other nuclear facilities authorized by DOE NE, for nuclear safety design criteria; fire protection; criticality safety, and natural phenomena hazards (NPH) mitigation
- NE-STD-1020-2025, “[Hazards Analysis and Design Criteria for DOE Facilities](#),” provides requirements and guidance for the analysis and design of facility SSCs necessary to implement the NPH requirements of DOE NE O 420.1 and the use and reference of industry building codes and voluntary consensus standards in the NPH analysis and design of SSCs in DOE facilities.
- NE-STD-1027-2025, “[Hazard Categorization of DOE Nuclear Facilities](#),” provides requirements and guidance for determining if a DOE nuclear facility is a Hazard Category (HC) 1, 2, 3, or below HC-3 nuclear facility, as required by 10 CFR Part 830.
- NE O 425.1, “Verification of Readiness to Startup or Restart Hazard Category 1 and 2 Nuclear Facilities,” establishes the requirements for NE readiness reviews. This includes verification of the engineering processes used to determine operational capability, the effectiveness of safety systems and safety management programs, and the proper implementation of the documented safety basis.
- Additional DOE Office of Nuclear Energy Orders establish supporting programmatic requirements for nuclear facility authorization and operation, as applicable. These include NE O 151.1, “Comprehensive Emergency Management System”; NE O 422.1, “Conduct of Operations”; NE O 426.2, “Personnel Selection, Training, Qualification, and Certification Requirements”; NE O 433.1, “Maintenance Management Program for DOE Nuclear Facilities”; and NE O 470.1, “Office of Nuclear Energy Safeguards and Security.” Collectively, these directives address emergency preparedness, operational controls, personnel competency, equipment maintenance, and safeguards and security.

APPENDIX B—Quality Assurance Program

1. NRC Requirements

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, 10 CFR Part 52, 10 CFR Part 53, and proposed 10 CFR Part 57, an applicant must have a description of the quality assurance (QA) program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility.

For reactors licensed under 10 CFR Parts 50 and 52, quality assurance programs (QAPs) must meet the requirements in 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” unless an exemption can be justified or an alternative approach, such as the proposed 10 CFR 50 Appendix T, is applied. The Appendix B excerpt below summarizes the required scope of an NRC applicant’s QAP:

- Nuclear power plants and fuel reprocessing plants include structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. This appendix establishes quality assurance requirements for the design, manufacture, construction, and operation of those structures, systems, and components. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those structures, systems, and components; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.
- “Quality Assurance,” as defined by Appendix B, “comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements.”

Applicable requirements in 10 CFR Part 53, “Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants,” include:

- § 53.460, “Safety categorization and special treatments” states that structures, systems, and components (SSCs) must be classified according to their safety significance. The SSC categories must include:
 - Safety-Related (SR)
 - Non-Safety-Related but Safety-Significant (NSRSS)
 - Non-Safety-Significant as defined in subpart A of 10 CFR 53.
- For SR and NSRSS SSCs, the conditions under which they must perform their safety function in § 53.230 must be identified. Special treatments must be established in accordance with this and other subparts to provide confidence that the SSCs will perform under the service conditions and with reliability consistent with the analysis performed under § 53.450 to demonstrate meeting the safety criteria in §§ 53.210 and 53.220.
- The special treatments for SR SSCs must include meeting the applicable QA requirements from appendix B of part 50 of this chapter.

- The special treatments for NSRSS SSCs and special treatments for SR SSCs beyond those required under “a.” may include meeting selected QA requirements from appendix B of part 50 of this chapter when such treatment is needed to address performance requirements, equipment reliability, or uncertainties.
- The identification of special treatments for SR and NSRSS SSCs must account for human actions needed to prevent or mitigate licensing basis events (LBEs), the need to perform such actions reliably under the postulated environmental conditions, and the role of programs established in accordance with subpart F of 10 CFR 53 to provide confidence that those actions will be performed as assumed in the analysis performed in accordance with § 53.450 to demonstrate meeting the applicable criteria in §§ 53.210, 53.220, and 53.450(e).

Proposed 10 CFR Part 57, “Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles” would include the following requirement in 10 CFR 57.60(a)(3) for the QA program to be included in the final safety analysis report:

- A description of the quality assurance program to be applied to the design, fabrication, manufacture (as applicable), construction, and testing of the safety-related SSCs of the facility.

A proposed rule under EO 14300, entitled “Modernizing Reactor Licensing, Safety Oversight, and Siting Practices,” includes the following proposed requirements:

- 10 CFR Part 50, Appendix T, “Streamlined Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.” The proposed Appendix T would provide a voluntary alternative to 10 CFR Part 50, Appendix B applicants under 10 CFR Parts 50, 52, and 53 meeting certain conditions could use in their respective applications for combined licenses (COLs), construction permits (CPs), and operating licenses (OLs) under certain eligibility requirements. The proposed rule supports implementation of E.O. 14300, section 5, through:
 - Establishment of performance-based QA criteria that provide explicit direction on the use of a graded approach for applying QA requirements to SSCs relative to their safety and risk contributions to the overall nuclear facility. The graded approach in the proposed rule is consistent with the DOE’s graded approach established in DOE O 414.1.
 - Establish “quality assurance” which are all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. QA includes quality control, which comprises those actions related to the physical characteristics of a material, structure, component, or system that provides a means to ensure the material, structure, component, or system meets predetermined requirements;
 - Incorporation of QA terminology and methodologies used across various industries;
 - Enhanced regulatory certainty during the application process; and

- Removal, when appropriate, of NRC oversight of suppliers and vendors of products and services related to nuclear power plant and fuel reprocessing plant SSCs subject to this proposed rule.

2. NRC Guidance and Related References

Content of License Application Requirements Related to QA

For a nuclear power plant application submitted under 10 CFR Part 50, the NRC requires the application to include the information specified in 10 CFR §§ 50.33, 50.34, and 50.36. The regulations require applicants include descriptions of the QA program to be applied to the plants design, fabrication, construction, and testing of the structures, systems, and components. 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” sets forth the requirements for QA programs for nuclear power plants and fuel reprocessing plants. The description of the QA program for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of Appendix B or an alternative such as the proposed Appendix T will be satisfied. A QAPD for a CP or ESP application must cover the design, fabrication, construction, postconstruction testing, and preoperational testing activities. A QAPD for an OL must describe the operational activities.

For a nuclear power plant COL application submitted under 10 CFR Part 52, the NRC requires the application to include a QAP that will vary depending on whether the application is for an early site permit (ESP), a COL, a design certification (DC), a design approval (DA), or a manufacturing license (ML), and the scope of activities covered by the application. For example, a QAPD for a DC must cover the design of the structures, systems, and components of the facility, and a QAPD for a COL under Part 52 must cover the full scope of both CP and OL plans in applications submitted under Part 50.

10 CFR 53 requires the content of applications for various permits and licenses to develop, implement, and maintain a QA program in accordance with 10 CFR 50, Appendix B or an alternative such as the proposed Appendix T. The following outlines 10 CFR 53 permits and licenses and QAP controls.

- CP applicants under 10 CFR 53.1309 are required to include in the Preliminary Safety Analysis Report (PSAR), a description of the QAP to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility.
- OLs under 10 CFR 53.1369 must include a Final Safety Analysis Report (FSAR), information pertaining to the managerial and administrative controls to be used to assure safe plant operation.
- COL applications subject to 10 CFR 53.1416 are required to include in the FSAR a description of the QA applied to the design, and to be applied to the fabrication, construction, and testing of the structures, systems, and components of the facility and to the managerial and administrative controls to be used to assure safe operation.
- Early site permit (ESP) applicants subject to §53.1146 are required to include in the site SAR a QAP that is applied to the site activities related to the future design, fabrication, construction, and testing of the structures, systems, and components of a facility or facilities that may be constructed on the site.

- DAs required by the provisions of §53.1209 shall include in the submittal a description of the QAP that is applied to the design of the structures, systems, and components of the facility.
- DC applications under § 53.1239 require in the FSAR a description of the QAP that applies to the design of the structures, systems, and components of the facility. All applicants must submit applications that contain design level information sufficient to enable the Commission to judge the applicant's proposed means assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted for a design certification must include performance requirements and design information of sufficient detail to permit the preparation of acceptance, and inspection requirements by the NRC. The Commission will require, before design certification, that information normally contained in engineering documents, such as analyses, drawings, procurement specifications, or construction and installation specifications, be completed and available for audit. If more detailed information is necessary, the Commission to verify the information in the application and make its safety determination.
- ML applications are required by the provisions of 53.1279 to include in the FSAR a description of the QA program applied to the design, and to be applied to the fabrication and testing of the structures, systems, and components of the manufactured reactor.

Quality Assurance Guidance

- Regulatory Guide (RG) 1.28, Revision 6 “Quality Assurance Program Criteria – Design and Construction” (ML2177A002) endorses the American Society of Mechanical Engineers (ASME) Nuclear Quality Assurance (NQA)-1-2022 standard as an acceptable method of meeting 10 CFR Part 50, Appendix B.
- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 17.5, “Quality Assurance Program Description—Design Certification, Early Site Permits, and New License Applicants” (ML15037A441). The NRC staff’s review of both light water reactor (LWR) and non-LWR applications using a traditional deterministic approach will be consistent with the guidance in NUREG-0800 because QAPDs for non-LWR applications are programmatic and not technology specific.

For QA associated with non-safety-related structures, systems, and components that are risk significant (i.e., non-safety-related with special treatment (NSRST) classification using the Licensing Modernization Project (LMP) process), additional staff review guidance in NUREG-0800, Section 17.4, “Reliability Assurance Program (RAP),” (ML063190018) may be appropriate.

- RG 1.253, “Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses Certifications, and Approvals for Non-Light-Water Reactors” (ML23269A222) endorses the methodology described in NEI 21-07, Revision 1 to support reviews of non-LWR applications using the LMP risk-informed approach. This RG specifically provides guidance on QA information that should be included in applications submitted following the LMP framework.

The NRC recommends that the QAPD be a standalone document that may be submitted by applicants as a topical report and incorporated by reference into the safety analysis report.

Additional References for Applicant and Staff Consideration

The following additional guidance documents may be useful to an applicant, the NRC staff, or both, but are not intended to be a comprehensive listing of relevant guidance documents. These documents are provided as background that an applicant may find useful in preparing the application:

- RG 1.33, Revision 3, “Quality Assurance Program Requirements (Operation),” issued June 2013. (ML13109A458).
- RG 1.164, Revision 1 “Dedication of Commercial-Grade Items for Use in Nuclear Power Plants,” issued April 2024. (ML24038A310).
- RG 2.5, “Quality Assurance Program Requirements for Research and Test Reactors.” (ML093520099).
- ASME NQA-1, “Quality Assurance Requirements for Nuclear Facility Applications.”
 - Subpart 2.2, “Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Facilities.”
 - Subpart 2.7, “Quality Assurance Requirements for Computer Software for Nuclear Facility Applications.”
 - Subpart 2.14, “Quality Assurance Requirements for Commercial Grade Items and Services.”
 - Subpart 2.17, “Quality Assurance Requirements for Electronic Quality Assurance Records Systems.”
 - Subpart 2.22, “Quality Assurance Requirements for Management Assessment and Quality Improvement for Compliance With 10 CFR 830 and Department of Energy (DOE) Order 414.1 for DOE Nuclear Facilities.”
 - Subpart 4.1.5, “Guidance to Modification of an ANSI/ANS-15.8–1995 (R2005; R2013) Quality Program to Meet NQA-1–2012 Requirements.”
- ANS-3.1, “Selection, Qualification, and Training of Personnel for Nuclear Power Plants.”
- ANS 3.2, “managerial, administrative, and quality assurance controls for the operational phase of nuclear power plants.”
- ANSI/ANS 15.8, “quality assurance program requirements for research reactors.”
- 10 CFR 50.55a “Codes and standards” incorporates by reference ASME NQA-1 into NRC regulations for boiling and pressurized water-cooled nuclear power reactors.

3. Summary of DOE Requirements, Order, and Standards

DOE requirements for the facility QA program typically consist of 10 CFR 830, Subpart A, “Quality Assurance Requirements” and Office of Nuclear Energy (NE) Order (O) 414.1, “Quality Assurance Order.”

- 10 CFR 830, “Nuclear Assurance Management” [§830.7](#), “Graded approach, states that a graded approach may be used to implement the requirements of this regulation in a manner commensurate with the associated risks, provided that the basis for the approach is documented, and the documentation is submitted to DOE. The graded approach must not be used to implement the Unreviewed Safety Question (USQ) process. The DOE graded approach allows for facilities with simple designs to require less rigor than a complex facility.
- 10 CFR 830.121, ‘Quality Assurance Program.” DOE requirements to ensure DOE contractors implement a QA program that provides confidence nuclear safety requirements are met for DOE nuclear facilities and activities including design, construction, operation, and decommissioning.
- DOE-STD-1271-2026, “Authorization Pathway for Nuclear Facilities” establishes DOE oversight of nuclear facilities, programs, and activities, ensuring compliance with safety, security, environmental, and legal obligations.
- DOE-HDBK 1221-2024, “Suspect/Counterfeit Items Resource Handbook” assists the DOE enterprise in preventing suspect/counterfeit items (S/CI) from entering the supply chain. Specifically, it provides examples and information that may be used to prevent, detect, document, report, and control these types of items.
- NE O 414.1, “Quality Assurance Order” establishes DOE requirements for an integrated QAP to ensure work is conducted under proper controls and meets requirements.
- NE O 226.1, “Authorization Pathway for Nuclear Facilities” establishes DOE oversight of nuclear facilities, programs, and activities, ensuring compliance with safety, security, environmental, and legal obligations.
- NE O 226.2, “Federal Oversight and Contractor Assurance Systems” contains DOE requirements for contractors to maintain an effective Contractor Assurance System (CAS) and for DOE to conduct risk-based oversight of its contractors.
- NE O 227.1, “Independent Oversight Program” establishes DOE’s independent oversight of its contractors, facilities, and programs to ensure safe, secure, and compliant operations.
- NE O 232.2, “Occurrence Reporting and Processing of Operations Information” discusses reporting to DOE personnel events under the responsibility of the Office of NE that could adversely affect the health and safety of the public or the workers, the environment, DOE missions, or the credibility of the Department.

4. Staff Considerations on NRC and DOE Requirements and Guidance

QA records generated under the DOE QA programs shall be retrievable, complete and accurate to provide objective evidence that enables the NRC to establish equivalency to nuclear safety requirements. These records prove that structures, systems, components, and programmatic elements were designed, fabricated, tested, and documented in accordance with applicable safety standards.

Leveraging Across NRC Licensing Frameworks

Applicants under 10 CFR Parts 50, 52, and 53 must meet the QA requirements of 10 CFR 50, Appendix B, unless an exemption can be justified or an alternative approach, such as the proposed

10 CFR 50 Appendix T, is applied. Proposed 10 CFR Part 50, Appendix T, if approved, provides a streamlined approach to QA. The graded approach in the proposed Appendix T is consistent with the graded approach and eleven QA criteria established in DOE O 414.1. Under proposed part 57, applicants would not have to meet the requirements in Appendix B of 10 CFR Part 50 but instead would choose an industry-approved QA program, subject to justification that the chosen program is appropriate for the design and operation of the facility.

10 CFR Part 50, Appendix B sets forth QA criteria for nuclear power plants and fuel reprocessing plants, while proposed 10 CFR Part 50, Appendix T and proposed 10 CFR Part 57 would provide more streamlined approaches justified based on the characteristics of the reactor facility.

Considerations of NRC and DOE Requirements

The DOE and NRC QA frameworks share a common foundation in ASME NQA-1 and cover the full SSC lifecycle. NRC has endorsed ASME NQA-1-2022 as an acceptable method to meet Appendix B. DOE implements QA programmatically under 10 CFR 830 and NE O 414.1 with a graded approach integrated into the safety basis, while NRC applies Appendix B to 10 CFR Part 50 within the licensing basis, reviewed using NUREG 0800, Section 17.5. Proposed Appendix T would align more closely with the approach in NE O 414.1.

While both frameworks converge on NQA-1 as noted above, NE O 414.1 does not formally invoke specific parts or subparts of NQA-1 in a similar prescriptive manner as the NRC does through RG 1.28. The main DOE-to-NRC translation challenge is not the existence of a requirement, but the degree of prescriptiveness and objective evidence expected by NRC review. NE O 414.1 is usually less prescriptive at the criterion text level, but DOE closes much of that gap through QAPD approval, Code of Record commitments, safety basis controls, occurrence reporting, contractor assurance, and independent oversight. Implementation of NRC requirements are the responsibility of the licensee in accordance with its NRC approved QAPD. The NRC provides oversight and enforcement of the implementation as an independent regulator, while DOE requirements are authorization/contract/oversight enforced.

Use of DOE standards for NRC licensing

For applicants considering the use of the proposed 10 CFR Part 50, Appendix T, the eleven QA criteria required of SSCs within the scope of the proposed rule are generally consistent with the ten QA criteria established in NE O 414.1. However, NE O 414.1 does not contain "Performance: Maintenance of structures, systems, and components." The QA criteria for design and analysis software and software used for digital items within the scope of the proposed rule are consistent with the safety software quality criteria established in DOE O 414.1. Therefore, the potential for leveraging from NE O 414.1 to use of proposed 10 CFR Part 50, Appendix T is higher than for applicants following 10 CFR Part 50, Appendix B.

Below is a high-level crosswalk between NE O 414.1 criterion and Appendix B to 10 CFR Part 50. Similarities in the regulatory framework are as follows:

- 10 CFR 50 Appendix B Criterion I and NE Criteria O 414.1 1(a) and 10(b) require defined organizational structure, responsibilities, authority, and independence of quality functions. NE O 414.1 explicitly requires independent assessment personnel to have freedom from line management. NRC enforces QA organization independence through the licensing basis, while DOE enforces through QAPD approval, contract requirements, and internal oversight.

- 10 CFR 50 Appendix B Criterion II and NE O 414.1 Criteria 1(b), 4, and 9 require a documented, implemented, and assessed QA program. DOE requires QAPD approval and graded application of requirements, while NRC controls QA program changes through the licensing basis. If used, NRC supplements QA with formal risk-informed frameworks, whereas DOE uses a graded approach tied to hazards and mission.
- 10 CFR 50 Appendix B Criterion III (Design Control) → NE Criterion 6: Appendix B Criterion III and NE Criterion 6 both require control of design inputs, outputs, interfaces, verification, and changes. DOE strengthens implementation through NE O 420.1 configuration management and Code of Record requirements. DOE-STD-1237 further aligns DOE reactor safety bases with NRC methodologies (e.g., RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,” and NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors”). NRC uniquely ties design changes to the licensing basis via 10 CFR 50.59.
- 10 CFR 50 Appendix B Criterion IV requires procurement documents to define technical and QA requirements and ensure flow down. NE O 414.1 Criterion 7 establishes procurement controls more broadly, including supplier qualification and performance. NRC adds mandatory 10 CFR Part 21 defect-reporting and flow down requirements whereas DOE implements comparable controls through QA, contracts, and occurrence reporting without an equivalent external reporting rule.
- 10 CFR 50 Appendix B Criterion V and NE O 414.1 Criterion 5(a) both require work affecting quality to be performed using approved procedures and controls. NRC Technical Specifications impose additional enforceable procedural requirements as license conditions. DOE implements comparable controls through TSRs, the safety basis, and conduct of operations.
- 10 CFR 50 Appendix B Criterion VI and NE O 414.1 Criterion 4(a) require control of documents through preparation, review, approval, issuance, and revision. DOE implements this programmatically, while NRC applies additional licensing-basis change controls for certain documents (e.g., QAPD, FSAR).
- 10 CFR 50 Appendix B Criterion VII and NE O 414.1 Criteria 7(b) and 7(c) require supplier evaluation, selection, and ongoing performance assurance. DOE adds explicit suspect/counterfeit item controls. NRC uniquely imposes 10 CFR Part 21 reporting and commercial-grade dedication requirements for safety-related items.
- 10 CFR 50 Appendix B Criterion VIII and NE O 414.1 Criterion 5(b) require identification and control of items to ensure proper use. Appendix B is more prescriptive regarding traceability and marking. DOE defines the requirement programmatically with implementation through procedures, safety basis controls, and applicable standards.
- 10 CFR 50 Appendix B Criterion IX and NE O 414.1 Criteria 5(a) and 2 require qualified procedures and personnel for special processes. DOE implements prescriptive requirements through approved codes and standards in the Code of Record (NE O 420.1). NRC incorporates applicable codes directly through regulation (e.g., 10 CFR 50.55a). Further discussion of codes and standards is provided in Appendix L of this report.
- 10 CFR 50 Appendix B Criterion X and NE O 414.1 Criterion 8(a) require inspections using defined acceptance criteria. Appendix B to 10 CFR Part 50 explicitly requires inspector independence from the work performed. NRC defines inspection scope and intervals through incorporated codes, while DOE implements inspections through QA

programs, and configuration management, with DOE-STD-1237, “Authorization Pathway for Nuclear Facilities,” reinforcing structured safety analysis and surveillance expectations.

- 10 CFR 50 Appendix B Criterion XI and NE O 414.1 Criterion 8(a) require testing to demonstrate performance against requirements. NRC supplements this with prescriptive test programs (e.g., Appendix J to 10 CFR Part 50, ASME Operations and Maintenance Code). DOE implements testing and surveillance through the QAP, safety basis, TSRs, and Code of Record, with DOE-STD-1237 reinforcing structured safety analysis and surveillance expectations.
- 10 CFR 50 Appendix B Criterion XII and NE O 414.1 Criteria 5(d) and 8(b) require control and calibration of measuring and test equipment. Appendix B specifies traceability and calibration controls explicitly. DOE establishes these requirements programmatically, with detailed implementation defined in the QAP and applicable standards.
- 10 CFR 50 Appendix B Criterion XIII and NE O 414.1 Criterion 5(c) require protection of items from damage, loss, or deterioration. Appendix B to 10 CFR Part 50 is more prescriptive regarding handling, packaging, and storage controls. DOE implements these requirements through procedures, safety basis controls, and approved QA program requirements.
- 10 CFR 50 Appendix B Criterion XIV and NE O 414.1 Criteria 8(a) and 5(b) require control of inspection/test status and operating status. NRC Technical Specifications impose structured limiting condition for operation (LCO) tracking and required actions as license conditions. DOE implements comparable status control through TSRs, safety basis controls, and configuration management, with DOE-STD-1237 reinforcing TSR structure.
- 10 CFR 50 Appendix B Criterion XV and NE O 414.1 Criterion 3(b) require identification, control, and disposition of nonconforming items. DOE supplements this with occurrence reporting (NE O 232.2), contractor assurance (NE O 226.1), and independent oversight (NE O 227.1). NRC uniquely requires external defect reporting under 10 CFR Part 21.
- 10 CFR 50 Appendix B Criterion XVI and NE O 414.1 Criteria 3(a), 3(c), and 3(d), with support from Criterion 9, require identification, correction, and prevention of conditions adverse to quality. DOE strengthens corrective action through occurrence reporting, contractor assurance systems, and independent oversight. NRC requires defined treatment of significant conditions adverse to quality within the licensing-basis QA program.
- 10 CFR 50 Appendix B Criterion XVII and NE O 414.1 Criterion 4(b) require records that provide objective evidence of quality. Appendix B specifies retention, storage, and protection requirements in detail. NRC requires retention tied to the licensing basis and facility life (10 CFR 50.71, “Maintenance of records, making of reports”). DOE implements records control through QA programs and federal records management requirements.
- 10 CFR 50 Appendix B Criterion XVIII and NE O 414.1 Criterion 10 require independent, periodic audits/assessments by qualified personnel with documented results and follow-up actions. DOE supplements internal assessments with federal oversight and independent assessments (NE O 226.1 and 227.1). Under NRC licensing, the licensee’s internal audit program must stand alone, with NRC inspections providing external oversight but not replacing internal audits.

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE QA information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency. In determining comparability, staff should evaluate alignment of standards and editions (e.g. ASME NQA-1 versions), method of program invocation, graded approach application to credited SSCs, supplier qualification and commercial grade dedication practices, software QA controls, corrective action and records control rigor, and audit/independence provisions.

The table below outlines three of the key factors in determining the extent to which DOE authorization information related to QA may be leveraged.

Table B.1 – Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	The objective of DOE NE O 414.1 is to establish an integrated QAP that assures: Controls for the conduct of work, The meeting of requirements, and; Achievement of organizational mission needs. The objective of NRC's 10 CFR 50, Appendix B establishes QA requirements for the design, manufacture, construction, and operation of those structures, systems, and components. Appendix B applies to all activities affecting the safety-related functions of those structures, systems, and components; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying. For QA program comparability, staff should verify criterion-level mapping to 10 CFR Part 50, Appendix B (organization independence; design control; procurement including dedication; inspection/testing; document control; records; audits; corrective action), supported by auditable objective evidence (procedures, supplier files, dedication packages, software V&V records, audit and design reports, corrective action logs, and records indices). The NRC is proposing two new rules that may impact QA requirements, the proposed 10 CFR Part 50 Appendix T and the proposed 10 CFR Part 57. For applicants considering the proposed part 57 or the proposed Appendix T, more direct leveraging of DOE information is expected. DOE QA is implemented programmatically under 10 CFR Part 830 and NE O 414.1 with a graded approach integrated into the safety basis; NRC QA is licensing basis driven under Appendix B to 10 CFR Part 50 and reviewed using SRP 17.5, with additional overlays (e.g., 10 CFR Part 21, 50.55a, 50.59, 50.71). These structural differences inform how far DOE QA can be leveraged in NRC applications.
Application quality and justification of relevance of DOE authorization to NRC review	To maximize potential leveraging, the application should map DOE safety basis documentation to NRC regulatory requirements, ensuring traceability and objective evidence at the criterion level. Where deviations exist, the application should provide justification and practical examples, supported

	by crosswalk matrices. Early engagement with NRC staff is recommended to clarify expectations and address any gaps.
Safety-significance of information leveraged	The safety-significance of information leveraged from DOE authorization is assessed based on its impact on public health and worker safety. Where DOE authorization aligns closely with NRC guidance, maximum leveraging is justified. For QA, the degree of leveraging may be limited to some degree by the cross-cutting nature of QA covering highly safety-significant aspects of the design.

NRC Staff Approach to Leveraging DOE QA Information

NRC staff should consider the similarity of QA-specific considerations. On this basis, NRC leveraging of DOE QA information can generally be organized into three categories.

For each category (especially the moderate and, most critically, the maximum) DOE QA records are essential regulatory evidence for NRC licensing. As objective, traceable, and verifiable documentation, they allow the NRC to determine whether DOE QA controls are sufficiently rigorous and whether DOE activities can be considered equivalent to NRC safety requirements. Record completeness, accuracy, and retrievability directly support NRC safety findings and equivalency determinations.

Maximum leveraging:

- Acceptance, with minimal additional review, of the full scope of QA related information and demonstration activities performed under DOE authorization as the primary basis for licensing decisions.
- NRC would recognize the DOE's authorization process including design, construction, and prototype testing predominately under 10 CFR Part 830 and DOE procedures (e.g. NE O 414.1) as providing sufficient objective evidence of QA program adequacy, provided the applicant demonstrates successful completion of DOE's requirements and testing validates the design and quality. NRC would focus its review on confirming that the DOE QA program (often NQA-1-based, with graded approaches) was properly implemented and that any gaps to NRC-specific requirements (e.g., unique regulatory guides or subparts of NQA-1) are transparently identified and justified. Efficiency is maximized by reducing duplicative reviews, relying on DOE's technical rigor and oversight, and only supplementing with NRC review where clear regulatory differences exist. This approach is most appropriate when there is strong overlap in QA criteria, mature DOE QA program implementation, and robust documentation of testing and performance outcomes.
- For applications under 10 CFR Parts 50, 52, and 53 using 10 CFR Part 50, Appendix B, NRC's ability to maximally leverage QA requires demonstrable alignment to an NRC-endorsed ASME NQA-1 edition (per RG 1.28), complete Appendix B criterion-level mapping with objective evidence, and explicit satisfaction of NRC overlays (Part 21, §50.55a, §50.59, §50.71).
 - NRC staff may accept DOE authorization QA program material as a primary technical basis when the following are all true:
 - The DOE QA program demonstrably implements ASME NQA-1 versions endorsed by NRC (per RG 1.28) or provides equivalent controls, and the program scope covers Safety Related (SR) SSCs or their DOE equivalents in a way that is traceable to NRC accident/safety functions.
 - QA criteria and implementing procedures show criterion level traceability to Appendix B topics (organization, design control, procurement, inspection/test,

- document/records control, audits, corrective action), including objective evidence and auditable records.
- Supplier oversight and dedication practices, software QA, and records retention meet NRC expectations or are shown equivalent; the applicant addresses NRC overlays such as Part 21, 50.55a, 50.59, and 50.71 where applicable.
- Maximum leveraging is strongest for program elements that are standards driven and methodology neutral (e.g., NQA-1 Part I governance, selected Part II subparts such as software or dedication). Where DOE relies on Specific Administrative Controls (SACs) or contractor assurance constructs without direct NRC equivalency, supplemental mapping is typically required.
- For applicants considering the proposed 10 CFR Part 57, applicants would choose an industry-approved QA program, subject to justification that the chosen program is appropriate for the design and operation of the facility. Prior DOE authorization information may be leveraged under this scenario.
- For applicants considering the proposed 10 CFR Part 50, Appendix T, more direct leveraging of DOE information may be possible. The eleven QA criteria required of SSCs within the scope of the proposed rule are consistent with the ten QA criteria established in NE O 414.1 with the exception of “Performance: Maintenance of structures, systems, and components.”
- Leveraging of DOE QA information is limited where NRC imposes unique licensing-basis overlays or external reporting obligations. Maximum leveraging is typically not appropriate unless the application explicitly demonstrates conformance to: (a) 10 CFR Part 21 defect reporting and flow-down, (b) §50.59 licensing-basis change control processes, (c) §50.71 records retention scope and duration, and (d) Technical Specifications-driven operating/inspection status controls. Where these overlays are not already met, additional program changes or license conditions are required.
- *Example:* DOE QA demonstrably implements an NRC-endorsed ASME NQA-1 edition, with criterion-level traceability to Appendix B, supplier oversight, software QA, and records retention meeting NRC expectations. NRC review focuses on overlays (Part 21, 50.55a, 50.59, 50.71) and licensing basis integration.

Moderate leveraging:

- DOE authorization is a substantial input to its licensing review, but with targeted supplemental NRC evaluation of QA program elements most critical to public health and safety or where regulatory requirements diverge.
- NRC would accept DOE QA documentation and test results for major portions of the licensing basis, such as design, construction, and prototype testing, while conducting focused reviews or audits of areas where NRC requirements (e.g., specific NQA-1 subparts, regulatory guides, or recordkeeping) are not fully addressed by DOE processes. Applicants would be required to provide a crosswalk or gap analysis between DOE and NRC QA requirements, and to implement compensatory measures or additional documentation where gaps are identified. This approach balances efficiency and regulatory assurance, leveraging DOE’s oversight and technical work while ensuring NRC’s unique statutory obligations are met. It is well-suited for cases where DOE and NRC QA frameworks are broadly compatible but not identical, and where NRC must maintain confidence in the traceability and independence of QA records and processes.
- Moderate leveraging should use DOE QA program documentation for common foundations such as standards invoked, procedures, supplier qualification, and audits with supplementation to meet NRC licensing basis specifics:

- Provide an Appendix B crosswalk that identifies gaps (e.g., explicit Part 21 reporting flows, Appendix B design control coverage of reactor physics/stress/thermal hydraulic analyses, surveillance intervals, independence of QA audits) and closes them via program changes or license conditions.
- Translate DOE graded approach into NRC applicability by showing that SR SSCs receive full Appendix B treatment, and tailoring controls for non-safety related items consistent with 50.69 or TI-RIPB frameworks when proposed.
- Explicitly identify and implement applicable NQA-1 Part II subparts (e.g., Subpart 2.7 software, Subpart 2.14 commercial grade dedication) and demonstrate invocation, application, and verification consistent with NRC expectations.
- *Example:* DOE QA aligns on foundations but lacks explicit Part 21 flows, §50.55a code linkage, §50.59 change control, or explicit invocation of NQA-1 Part II subparts. Provide an Appendix B (or proposed Appendix T) crosswalk, gap closure commitments, and auditable evidence; NRC leverages DOE content while verifying supplemental controls.

Limited leveraging:

- DOE authorization is used only as a reference or background, requiring applicants to fully demonstrate compliance with NRC QA requirements (e.g., 10 CFR Part 50, Appendix B, and NQA-1 or proposed Appendix T) through independent documentation, programmatic alignment, and direct NRC review.
- In this scenario, NRC would not accept DOE QA program implementation or test results as sufficient for licensing, instead requiring applicants to reconstitute or supplement their QA documentation and processes to meet NRC standards. DOE authorization would be considered informative but not determinative, and NRC would perform comprehensive reviews of all QA program elements, regardless of prior DOE acceptance. This approach is most appropriate when there are significant differences between DOE and NRC QA requirements, when DOE QA implementation is immature or insufficiently documented, or when NRC statutory obligations require a higher degree of independence and traceability than DOE processes provide. While maximizing NRC's regulatory assurance, minimal leveraging offers the least efficiency and greatest duplication of effort.
- Where designs, safety case structure, or QA governance differ materially, DOE QA information will be treated as background to inform program architecture, standard selection, and hazards driven controls. The applicant then submits a QAPD (or equivalent part 52, part 53, or proposed part 57 program description) that stands on its own, with DOE content cited as non-binding technical context.
- *Example:* Material differences in governance or safety case structures; treat DOE QA information as background only. Applicant submits a standalone QAPD or part 53 or proposed part 57 QA description that meets NRC requirements directly, with DOE references as nonbinding context.

6. Practical Expectations for Applicants Utilizing Appendix B

DOE QA records are essential regulatory evidence for NRC licensing. As objective, traceable, and verifiable documentation, they allow the Commission to determine whether DOE QA controls are sufficiently rigorous and whether DOE activities can be considered equivalent to NRC safety

requirements. Record completeness, accuracy, and retrievability directly support NRC safety findings and equivalency determinations.

NRC screening to determine whether DOE QA program information is leverageable in an NRC QA review under Appendix B include:

- Standards alignment: DOE QA invokes NQA-1 editions endorsed by NRC or provides equivalent controls, with documented exceptions/clarifications.
- Appendix B mapping: Clear, criterion level mapping shows organization, program, design control, procurement, inspection/test, document control, records, corrective action, audits equivalence, with objective evidence.
- Supplier/Dedication: Commercial grade dedication and defect reporting are addressed to NRC expectations (Part 21) or justified via alternative controls.
- Software QA: Software classified and controlled per NQA-1 Part II subparts with verification/validation evidence suitable for NRC auditing.
- Graded approach translation: DOE graded QA is translated into NRC applicability: SR SSCs receive full Appendix B, NSRST/NST treatments justified if TI-RIPB is used.
- Change control and records: Program changes controlled to NRC licensing basis expectations (e.g., QAPD change processes) and records meet 50.71 style retention.

Applicants Under Appendix B leveraging DOE Quality Assurance should:

- Identify the applicable DOE and NRC requirements, standards, and editions; explain their relationship and programmatic mapping; and document criterion level traceability to NRC Appendix B.
- Provide a gap analysis and remediation plan for NRC overlays (Part 21 reporting, 50.55a code compliance, 50.59 change control, 50.71 records), plus any technology inclusive adjustments (e.g., RG 1.233 / NEI 18-04 framing for non-LWRs).
- Submit objective evidence (procedures, records, supplier files, audit reports, corrective action logs) demonstrating that QA activities affecting quality have been performed per approved requirements and are auditable to NRC standards.
- Recognize that a prior DOE authorization does not define the scope or extent of NRC review; NRC retains independent determination of adequacy for licensing.
- Explicitly invoke ASME NQA-1 (edition endorsed by NRC) and list applicable Part II subparts in the QAPD, with objective evidence of implementation (e.g., Subpart 2.7 for software; Subpart 2.14 for commercial grade dedication).
- Provide an Appendix B crosswalk showing criterion level traceability from DOE objective evidence to Appendix B topics; identify gaps and close them with program changes or license conditions.
- Address NRC overlays explicitly: Part 21 defect reporting, §50.55a code compliance, §50.59 licensing basis change control, §50.71 records retention; show how DOE processes are integrated or supplemented.
- Translate DOE graded QA into NRC applicability: demonstrate full Appendix B treatment for SR SSCs and justify tailored controls for other SSC categories under Part 53 or LMP, as applicable.

THIS NRC STAFF WHITE PAPER HAS BEEN PREPARED AND IS BEING RELEASED TO SUPPORT INTERACTIONS WITH THE PUBLIC. THIS PAPER HAS NOT BEEN SUBJECT TO NRC MANAGEMENT AND LEGAL REVIEWS AND APPROVALS, AND ITS CONTENTS SHOULD NOT BE INTERPRETED AS OFFICIAL AGENCY POSITIONS. A PROSPECTIVE APPLICANT SHOULD NOT USE THE CONTENTS OF THIS PAPER OR RELY ON ITS CONTENTS IN PREPARING AN APPLICATION.

- Package objective evidence: procedures, records, supplier files, audit reports, and corrective action documentation demonstrating auditable compliance for activities affecting quality.

APPENDIX C—Licensing Event Selection and Classification of Structures, Systems, and Components

1. NRC Regulatory Requirements

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34 and 50.69; 10 CFR 52.47, 52.79, 52.137, and 52.157 related requirements under subparts B, C, and H of 10 CFR Part 53 and proposed 10 CFR 57.30, and 57.60, an applicant must provide a description and analyses of the SSCs of the proposed facility, including performance requirements and their bases, and the associated evaluations required to show that safety functions will be accomplished.

10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” provides regulations for licensing production and utilization facilities, including these excerpts associated with licensing event (LE) selection and SSC classification:

- 10 CFR 50.2 provides a definition for safety-related (SR) structures, systems, and components (SSCs) based on those SSCs needed to remain functional during and following design basis events to support specific safety functions.
 - A proposed rulemaking in response to EO 14300, “Modernizing Reactor Licensing, Safety Oversight, and Siting Practices,” proposes to add definitions to 10 CFR 50.2 for design basis events (DBEs) and beyond design basis events (BDBE). This rule change, if approved, would better focus safety assessments needed to demonstrate the capabilities of safety-related SSCs on credible and realistic risks. The proposed rule changes would provide definitions for the following terms: (1) DBEs based on those initiating events with a frequency greater than a determinate and data backed threshold that provides an appropriate level of safety; and (2) beyond design basis events (BDBEs) based on initiating events with a frequency below the DBE threshold but greater than a threshold for credibility that provides an appropriate level of safety. Draft guidance for determining thresholds for DBEs and BDBEs was provided for public comment in DG-1454, “Implementation of Determinate and Data Backed Thresholds for Reactor Safety Assessments.”
- 10 CFR 50.34 contains the requirements for the content of a Part 50 application including the preliminary design information and the safety analysis report.
- 10 CFR 50.69 includes risk-informed categorization and treatment requirements for SSCs.
- Appendix A to 10 CFR Part 50 provides general design criteria (GDC) for nuclear power plants.

10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” governs the issuance of standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities:

- 10 CFR 52.47 contains the requirements for the content of a Part 52 application, including a description and analysis of the SSCs.
- 10 CFR 52.79 contains the requirements for the final safety analysis report of a COL.
- 10 CFR 52.137 contains the content of application requirements for the Standard Design approvals, including a description and analysis of the SSCs of the facility.
- 10 CFR 52.157 contains the content of application requirements for a Manufacturing License.

10 CFR Part 53, “Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants,” establishes a risk-informed, performance-based, and technology-inclusive regulatory framework as an alternative for the licensing and regulation of commercial nuclear plants.

- 10 CFR 53.020 provides definitions. For licensing event identification, definitions are provided for anticipated event sequences, unlikely event sequences, very unlikely event sequences, design-basis accidents (DBAs), and licensing-basis events (LBEs). For SSC classification, definitions are provided for safety-related (SR) SSCs, non-safety-related but safety significant (NSRSS) SSCs, and non-safety-significant SSCs.
 - SR SSCs are defined as those SSCs relied upon to demonstrate compliance with the safety criteria in 10 CFR 53.210.
 - NSRSS SSCs are defined as SSCs that are not SR but are relied on to achieve adequate defense in depth or to perform risk significant functions and warrant special treatment.
- 10 CFR 53.240 requires LBEs to be analyzed under 10 CFR 53.450 and provides requirements concerning the LBEs that must be identified and their associated analyses.
- 10 CFR 53.410 requires that functional design criteria be defined for design features classified as SR in terms of their role in demonstrating compliance with the safety criteria in 10 CFR 53.210.
- 10 CFR 53.420 requires that functional design criteria be defined for design features that play a significant role in demonstrating that the safety criteria for LBEs other than DBAs in 10 CFR 53.220 are satisfied (which could include SR and NSRSS SSCs).
- 10 CFR 53.450 establishes analysis requirements and centers upon the use of a PRA, other SREs, or a combination thereof with other generally accepted approaches for systematically evaluating engineered systems.
- 10 CFR 53.460 provides requirements concerning safety categorization of SSCs and identification of special treatments, including quality assurance requirements.
- Content of application requirements:
 - 10 CFR 53.1239(a)(17) and (19) provide requirements for the FSAR of a standard design certification (DC) related to SSC classification and analysis related to LBEs, respectively. 10 CFR 53.1209, 53.1279, 53.1309, 53.1369, 53.1416 address content of application requirements for a standard design approval, manufacturing license, construction permit, operating license, and a combined license, respectively, that invoke the same requirements for SSC classification and analysis related to LBEs as the DC.

Proposed 10 CFR Part 57, “Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles,” if approved, would 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.

- Proposed 10 CFR 57.3 defines “safety related SSCs” as those relied upon to remain functional during and following design basis accidents to ensure: (1) capability to adequately control thermodynamic conditions and reactivity and to retain radioactive material; (2) capability to shut down the reactor and maintain it in a safe shutdown condition; and (3) capability to prevent or mitigate the consequences of accidents analyzed to meet the entry criteria in Subpart B of [proposed 10 CFR Part 57].
- Proposed 10 CFR 57.25 establishes a dose-based entry criterion requiring applicants to demonstrate that any individual in the unrestricted area would not exceed 1 rem (0.01 Sv) TEDE for the duration of a postulated accident that bounds a broad range of design basis accidents.

- Proposed 10 CFR 57.30 identifies design criteria attributes that applicants must address—such as reactivity control, heat removal, fission product retention, shielding, effluent control, and security by design—to ensure fundamental safety functions are maintained. To support these attributes, source term would be developed by the applicant when performing the maximum hypothetical accident (MHA) or maximum credible accident (MCA) methodology.
- Proposed 10 CFR 57.60 provides technical information for the content of applications including establishing the design, fabrication, construction, testing, and performance requirements for safety-related SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.
 - Proposed 10 CFR 57.60(a)(2) requires applicants to provide principal design criteria (PDC) and describe their relation to the design criteria attributes in proposed 10 CFR 57.30.
 - Proposed 10 CFR 57.60(a)(3) further requires applicants to describe the quality assurance program applied to the design, fabrication, manufacture, construction, and testing of safety related SSCs.
 - Proposed 10 CFR 57.60(a)(6) requires the applicant to provide an evaluation of the safety-related SSCs of the facility. The evaluation must clearly explain the SSC performance requirements and technical basis supporting the requirements to show that safety functions will be accomplished.
 - Proposed 10 CFR 57.60(a)(12) requires that the applicant provide a description of the design-specific risk analysis methods used to demonstrate adequate defense-in-depth and safety margin, including the results of the analysis.

2. NRC Guidance and Related References

The sections below summarize acceptable options for the identification of LEs and the related classification of SSCs. In many cases, applicants may be proposing to use a combination of approaches to address specific systems, hazards, and related uncertainties.

Light-Water Reactor Technologies Using a Predominantly Deterministic Approach

Newer or differing light-water reactor (LWR) designs may consider the draft guidance in RG 1.254, “Technology-Inclusive Identification of Licensing Events for Commercial Nuclear Plants” (ML25232A005), when establishing a process for the comprehensive identification of LEs. This RG provides guidance for systematically identifying licensing events that support the design, safety analysis, and licensing basis of commercial nuclear plants under 10 CFR Parts 50, 52, and 53. It emphasizes a technology-inclusive and “blank-sheet-of-paper” approach that avoids reliance on predefined accident lists and instead recommends the use of established hazard analysis and probabilistic risk assessment methods to consider all plant operating states, radiological sources, internal and external hazards, and human-induced events. RG 1.254 also addresses the use of relevant operating experience and independent reviews to ensure completeness.

Another source of LE identification and SSC classification guidance for LWR technology designs is NUREG-0800 (ML052340514) Chapter 15 addresses transient and accident analysis focused on events associated with specific operating commercial LWR designs. This approach may not completely or adequately reflect event types to be considered in the “blank paper” approach to the comprehensive assessment of new LWR designs described in RG 1.254. NUREG-0800 Sections 3.2.1 and 3.2.2 (ML052340514) provide SSC classification guidance for LWRs on the topics of seismic classification and system quality groups for mechanical systems.

Applicants implementing the alternative regulatory framework reflected in 10 CFR 50.69 to establish SSC special treatments can refer to RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," (ML061090627), for SSC categorization guidance. Additional insights and considerations based on experience and lessons learned from implementing this trial-use regulatory guide are summarized in the NRC's periodic review of RG 1.201.

Non-Light-Water Reactor Technologies Using a Predominantly Deterministic Approach

Within this category, various regulatory approaches may be taken to establish a comprehensive set of LEs for the proposed reactor facility. The considerations and guidance in RG 1.254 may offer useful information in establishing such LEs. This RG provides technology-inclusive guidance for identifying initiating events, delineating event sequences, and identifying licensing events that can be used to inform the design and licensing bases, as well as the content of applications for commercial nuclear plants licensed under 10 CFR Parts 50, 52, and 53.

Applicants pursuing a maximum hypothetical accident (MHA) approach in safety analysis may consider the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," (ML12251A353). This NUREG focuses on establishing the technical and licensing bases for proposed research or test reactor facilities to be licensed as class 104c research and development facilities and commercial non-power reactors, such as medical isotope production facilities, licensed as class 103 commercial facilities. However, it may provide useful insights and considerations for applicants proposing to use an MHA approach when pursuing commercial licenses or permits for nuclear power reactors. Note this guidance is not directly applicable as an acceptable method for LE identification and SSC classification for class 103 commercial power reactors.

Non-Light-Water Reactor Technologies Using a Risk-Informed and Performance-Based (Licensing Modernization Project) Approach

The NRC has endorsed, with clarifications and points of emphasis, a risk-informed and performance-based approach for informing the licensing basis and determining the appropriate scope and level of detail for parts of applications for licenses, certifications, and approvals for non-LWRs. That approach, also referred to as the Licensing Modernization Project (LMP), is reflected in Nuclear Energy Institute (NEI) 18-04, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light-Water Reactor Licensing Basis Development," (ML19241A472). RG 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," (ML20091L698), presents the results of the NRC's review of the approach. LMP provides a holistic, integrated methodology for licensing event identification and SSC safety classification. Under LMP, initiating events are identified and categorized based on their frequency, and SSCs are classified based on their role in preventing and mitigating licensing events or providing adequate defense-in-depth.

RG 1.253, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors" (ML23269A222), contains guidance for the development of associated NRC application content for applicants implementing the LMP approach. RG 1.253 guidance is based on the industry's content development approach for LMP-based applications, reflected in NEI 21-07, "Technology-Inclusive Guidance for Non-Light-Water Reactors: Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology," (ML22060A190). The NRC has also

issued companion guidance addressing other parts of applications that follow this approach. DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap,” (ML23277A139), presents that companion guidance.

Advanced Reactor Technologies Using the Part 53 Risk-Informed and Performance-Based Approach

The 10 CFR Part 53 framework provides an optional, technology-inclusive, risk-informed, and performance-based regulatory structure for licensing commercial advanced reactor facilities. Within this framework, applicants identify LBEs and classify the associated SSCs using methods that integrate deterministic engineering judgment, insights from probabilistic risk assessments (PRAs) or other systematic risk evaluations (SREs), and performance-based criteria.

The comprehensive identification of LBEs under Part 53 should be conducted objectively and without reliance on predefined lists, consistent with technology-inclusive guidance such as RG 1.254 (ML25232A005) and the LMP approach described in NEI 18-04, Revision 1 (ML19241A472), and endorsed in RG 1.233 (ML20091L698). In Part 53, SSCs are classified based on their role in preventing and mitigating accidents and providing adequate defense-in-depth.

Applicants applying under Part 53 may consider the guidance in RG 1.247 (ML21232A524) regarding PRA technical acceptability and RG 1.253 (ML23269A222) and NEI 21-07 (ML22060A190) to structure application content documenting LBEs, SSC classifications, functional design criteria, and performance requirements. The NRC staff is currently in the process of developing more comprehensive implementation and review guidance for 10 CFR Part 53.

Microreactor Designs Using the Proposed Part 57 Consequence-Based Approach

The proposed 10 CFR Part 57 rule does not use the exact term “Licensing Event Selection,” but it establishes a consequence-based, risk-informed approach for event selection and safety analysis that serves the same function for microreactors and certain compact designs.

Under proposed 10 CFR Part 57, applicants may consider the draft NUREG-2271, “Guidance for Preparing and Reviewing Applications Under 10 CFR Part 57.” Section 12.2 of draft NUREG-2271 provides guidance for selecting and categorizing design basis events. The guidance describes different types of risk methods for the applicant to develop a spectrum of credible design-basis events. Section 12.3 of draft NUREG-2271 provides guidance for selecting and categorizing DBEs, including both qualitative and quantitative criteria for inclusion in the licensing basis, and lists the event categories that must be considered. These event categories include insertion of excess reactivity, loss of coolant, loss of coolant flow, mishandling or malfunction of fuel, loss of normal electrical power, external events, and others. Appendix B of draft NUREG-2271 provides a deterministic SSC classification example describing how reactor plant conditions and event categories are linked to safety classifications.

3. Summary of DOE Requirements, Orders, and Standards

10 CFR Part 830 requires a DOE safety basis for Hazard Category 1, 2, and 3 nuclear facilities. Subpart B, “Safety Basis Requirements,” requires that the Documented Safety Analysis (DSA) identify hazards, evaluate normal, abnormal, and accident conditions, derive hazard controls, and demonstrate that those controls provide adequate protection of workers, the public, and the

environment. SSC classification is developed from this process by identifying which SSCs are credited to perform preventive or mitigative safety functions. The following bullets summarize the DOE orders and standards related to this topic:

- DOE-STD-1237, “Documented Safety Analysis for DOE Reactor Facilities” and DOE-STD-3009, “Preparation of Nonreactor Nuclear Facility Documented Safety Analysis” provides accepted DOE DSA methodologies that may be used by applicants.
- DOE-STD-1271-2026, “Authorization Pathway for Nuclear Facilities,” notes that the approach for format and content of the DSA may include the application of existing standards or development of an individualized approach.
- The DOE standards are supplemented with guidance from NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors”, RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants”, and ANS-15.21, “Format and Content for Safety Analysis Reports for Research Reactors” based on the reactor power and selected approach. These standards support identification and classification of credited controls, including:
 - Safety-Class SSCs, which are credited to protect the public, generally by preventing or mitigating accidents that could exceed DOE public dose evaluation guidelines (EG).
 - Safety-Significant SSCs, which make major contributions to defense-in-depth, worker safety, or significant accident prevention or mitigation.
- NE O 420.1, “Facility Safety,” integrates SSC classification into the design process by requiring safety analyses to identify Safety-Class and Safety-Significant SSCs, define their safety functions and functional requirements, and ensure that those SSCs are designed to perform their credited safety functions under applicable conditions, including natural phenomena hazards and other design basis challenges.

4. Staff Considerations on NRC and DOE Requirements and Guidance

This section contains high-level NRC staff observations on NRC and DOE requirements and guidance and how DOE requirements and standards may be used for various NRC licensing frameworks with respect to licensing event identification and SSC classification.

Leveraging Across NRC Licensing Frameworks

As discussed above, the NRC licensing frameworks contain a variety of requirements and acceptable approaches with respect to licensing event identification and SSC classification based on use of risk information as well as reactor technology (LWR or non-LWR) and hazard. Applicants and NRC staff reviewers should carefully consider the NRC framework being applied to a specific application to ensure DOE information is leveraged to the maximum extent practical while meeting appropriate NRC requirements.

Considerations on NRC and DOE Requirements and Guidance

The DOE and NRC regulatory frameworks share several foundational concepts in accident analysis, licensing event identification, and SSC classification, even though they rely on different terminology, regulatory structures, and implementation mechanisms. However, important differences must be

considered to ensure information or methodologies developed under DOE standards are appropriately adapted to NRC regulatory expectations.

DOE hazard analysis requirements, such as those in DOE-STD-3009, call for identifying initiating events, evaluating unmitigated accident consequences, and determining hazard controls necessary to prevent or mitigate those consequences. NRC frameworks likewise require identification of licensing events, including anticipated operational occurrences, design-basis accidents/events, and beyond-design-basis events. In both frameworks, accident analysis serves as the technical foundation for defining necessary safety functions and identifying SSCs essential for protecting public health and safety.

SSC Classification

NRC and DOE regulatory frameworks trace SSC classification directly from accident analysis results. DOE Safety-Class SSCs, which protect the public from radiological consequences, generally correspond to NRC Safety-Related SSCs credited during design-basis events. In NRC's risk-informed frameworks, such as LMP or Part 53, SSCs performing key safety functions are classified as safety-significant SSCs, which aligns conceptually with DOE Safety-Significant SSCs that contribute to defense-in-depth and worker protection. In both DOE and NRC frameworks, classification establishes the level of rigor applied to SSC design, qualification, performance, and reliability.

Several notable differences must be addressed when transitioning DOE information into an NRC licensing basis. One significant divergence lies in terminology and underlying regulatory constructs. DOE relies on design basis accidents derived through hazard analysis, often based on bounding unmitigated scenarios tied to threshold dose criteria. NRC uses multiple event categories—design-basis accidents, design-basis events, and licensing-basis events—depending on the licensing framework. Under 10 CFR Part 53, events must be categorized as anticipated event sequences, unlikely event sequences, or very unlikely event sequences, based on a combination of deterministic engineering judgment, PRA insights, defense-in-depth, and mechanistic source term considerations; design basis accidents are also identified based on these events and analyzed deterministically. DOE accident scenarios must be reassessed under NRC licensing constructs to ensure that credited safety functions and SSCs remain appropriate. SSC classification from DOE authorizations will not directly map to NRC SSC classification, and a systematic crosswalk will be required. Additionally, SSC classification under NRC licensing carries quality assurance requirements for Safety-Related SSCs that may exceed or otherwise differ from DOE requirements.

DOE hazard analysis also plays a more explicit role in worker safety, which is a primary driver of DOE Safety-Significant SSC designation. NRC licensing focuses on protection of public health and safety, with worker safety addressed through occupational dose requirements rather than SSC classification. As a result, DOE Safety-Significant SSCs may not correspond directly to NRC safety-related or non-safety-related but safety-significant SSCs, necessitating a careful assessment when preparing NRC applications.

Analysis Methodology

Other divergences arise from differences in analysis methodology. NRC's deterministic frameworks (Parts 50 and 52) do not require PRA for identification of LEs and the related classification of SSCs, but PRA becomes more critical under risk-informed approaches like LMP and Part 53. PRA informs licensing-basis event categorization, safety-significance determinations, and defense-in-depth

evaluations under Part 53. DOE analyses must be augmented with PRA when applicants pursue NRC’s risk-informed pathways.

Proposed 10 CFR Part 57 would adopt a consequence-based licensing model centered on a bounding accident that meets a 1-rem TEDE dose criterion in the unrestricted area, selected through maximum hypothetical accident or maximum credible accident methodologies. Whichever methodology the applicant selects must define a bounding accident that covers a broad range of design-basis accidents for the facility. DOE-developed MHAs or MCAs may provide valuable inputs for analyses that would be required under proposed part 57, but SSC classification must be reassessed considering the design criteria attributes in proposed 10 CFR 57.30 and principal design criteria established by the applicant under proposed 10 CFR 57.60.

Use of DOE standards for NRC licensing

RG 1.70 is recognized under DOE’s 10 CFR 830 safety basis requirements as an acceptable format for presenting accident analyses and design bases for LWRs. Once the relevant initiating events and accident sequences have been identified, RG 1.70 can serve as a common organizing structure for both DOE authorization and NRC licensing, with DOE requiring additional worker safety analyses beyond those needed for NRC licensing. Similarly, DOE-STD-1237 explicitly allows use of NUREG-1537 as a basis for DBA identification and SAR content for non-power reactors, supporting a consistent approach for both DOE and NRC reviews of non-LWR designs.

For advanced reactors that use the Licensing Modernization Project (LMP) approach, DOE-STD-1237 allows use of NEI 18-04 for identifying DBAs, which aligns with NRC’s endorsement of NEI 18-04 through RG 1.233.

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE licensing event selection and SSC classification information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

Consistent with the discussion in the main body of this white paper, this table outlines three of the key factors in determining the extent to which DOE authorization information related to licensing event identification and SSC classification may be leveraged.

Table C.1: Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	The DOE and NRC regulatory frameworks share several foundational concepts in accident analysis, licensing event identification, and SSC classification. In some cases, both frameworks may utilize the same guidance documents. There is a potential for leveraging DOE authorization information, but differences must be considered to ensure information or methodologies developed under DOE standards are appropriately adapted to NRC regulatory expectations. The specific NRC licensing pathway chosen will influence the amount

	of leveraging possible. Applicants pursuing risk-based NRC licensing approaches may require significant PRA development effort. SSC classification from DOE authorizations will not directly map to NRC SSC classification, and a systematic crosswalk will be required. Differences in related aspects of SSC classification, such as requirements for quality assurance for different classes of SSCs, must also be considered.
Application quality and justification of relevance of DOE authorization to NRC review	Since DOE SSC classification will not directly map to NRC requirements, applicants should consider NRC event selection and SSC classification early in their design process and provide a thorough and complete justification for their SSC classification against NRC requirements. Considering these differences early in the design and authorization process may help in maximizing the potential to leverage information from the DOE authorization process.
Safety-significance of information leveraged	Event selection and SSC classification are highly significant aspects of safe plant design and operational safety under all of NRC's licensing frameworks and DOE's authorization process. As a result, this factor will generally limit the degree of NRC leveraging with respect to event selection and SSC classification that can be achieved.

NRC Staff Approach to Leveraging DOE Licensing Event Selection and SSC Classification Information

The extent to which DOE licensing event selection and SSC classification information may be leveraged in an NRC review should be based on the extent to which the information provides a technically adequate basis for addressing the applicable NRC regulatory requirement. NRC staff should consider the similarities of licensing event selection and SSC classification schemes in NRC's licensing frameworks and the DOE authorization process.

On this basis, NRC leveraging of DOE licensing event selection and SSC classification information can generally be organized into three categories:

Maximum leveraging:

- Under maximum leveraging, DOE information and supporting engineering documentation can be incorporated into a license application under 10 CFR Part 50, 52, or 53, or proposed 10 CFR Part 57 with minimal modification and can serve as the foundation for NRC review. To achieve this, the applicant needs to consider the differences between DOE and NRC requirements and provided a clear technical justification that the bases used in DOE application are adequate for demonstration of compliance with the NRC requirements.
- *Example:* An example of maximum leveraging of DOE information is when an applicant has developed DOE safety basis documents (e.g., DSA, hazard analysis, TSRs) using guidance and format already compatible with NRC frameworks. This situation arises when a DOE-authorized reactor design follows NRC guidance such as RG 1.70 for format and content, uses initiating event lists and accident evaluation approaches consistent with NRC guidance, and considers differences in SSC classification early during DOE authorization.

Moderate leveraging:

- Under moderate leveraging, DOE documentation may be used to establish initial licensing event selection and SSC classification, but NRC may require additional justification, PRA integration, SSC reclassification, and licensing basis restructuring before acceptance. DOE information provides a strong technical starting point but cannot serve as the licensing foundation without refinement.
- *Example:* An example of moderate leveraging is when DOE information is technically justified and broadly compatible in structure but requires moderate reinterpretation or augmentation to meet NRC requirements, especially under risk-informed frameworks such as Part 53. In this scenario, DOE hazard analyses identify initiating events and select bounding DBAs, and SSC classification follows Safety Class/Safety Significant categories based on consequence thresholds. This information aligns conceptually with NRC approaches but must be restructured to match NRC's licensing event categories (AOOs, DBEs, BDBEs under Part 53; DBAs under Part 50/52; MHA/MCA under proposed 10 CFR Part 57) and expanded to include PRA insights and defense in depth evaluations, where appropriate. DOE SSC classifications can support identification of safety-significant SSCs, but they must be re-evaluated to ensure correct mapping to NRC requirements.

Limited leveraging:

- Under limited leveraging, DOE information may serve only as background material for understanding design evolution or identifying hazards, rather than forming the basis of licensing event selection or SSC classification under NRC frameworks. Accordingly, the DOE information would inform the NRC's review but would not constitute the primary technical basis for the NRC's safety determination.
- *Example:* An example of minimum leveraging is when DOE documentation exists, but methodological differences and regulatory requirements limit its direct applicability to NRC licensing. This may occur if DOE analyses do not align with NRC's LBE categories or if DOE's SSC classifications do not map to NRC safety related or safety-significant SSCs.

APPENDIX D—Design Criteria

1. NRC Regulatory Requirements

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, 10 CFR 52.47, 52.79, 52.137, and 52.157, 10 CFR 53.410, 53.420, 53.425, and 53.430; and proposed 10 CFR 57.30 and 57.60, an applicant must provide either principal design criteria (PDC) or functional design criteria in its NRC application for a commercial reactor.

10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” provides regulations for licensing production and utilization facilities, including these excerpts associated with PDC:

- 10 CFR 50.34(a)(3) requires that an application for a construction permit include the PDC for a proposed facility and describe how the design bases relate to the PDC.
- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” contains the general design criteria (GDC) that establish the minimum requirements for the PDC for water-cooled nuclear power plants. Appendix A also establishes that the GDC are generally applicable to other types of nuclear power units and are intended to provide guidance in determining the PDC for such other units.

10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities.

- 10 CFR 52.47(a)(3)(i) requires that an application for a design certification include the PDC for a proposed facility.
- 10 CFR 52.79(a)(4)(i) requires that an application for a combined license include the PDC for a proposed facility.
- 10 CFR 52.137(a)(3)(i) requires that an application for a standard design approval include the PDC for a proposed facility.
- 10 CFR 52.157(a) requires that an application for a manufacturing license include the PDC for a proposed facility.

Appendix A to 10 CFR Part 50 states that the PDC “establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) that are important to safety” and describes important to safety SSCs as providing “reasonable assurance that the nuclear power plant can be operated without undue risk to the health and safety of the public.”

Rather than PDC, 10 CFR Part 53, “Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants,” uses functional design criteria, which are defined as metrics for the performance of specific SSCs that are determined from the role of the SSC in meeting the safety criteria.

- 10 CFR 53.410 requires that functional design criteria be defined for safety-related design features. These safety-related design features are those relied upon to demonstrate that the consequences from DBAs would be below the criteria in § 53.210 based on analyses

performed under § 53.450(f), which include insights from deterministic analyses and probabilistic risk assessments, other systematic risk evaluations, or a combination thereof.

- 10 CFR 53.420 requires that functional design criteria be defined for design features that play a significant role in demonstrating that the safety criteria for licensing basis events (LBEs) other than design basis accidents (DBAs) are satisfied. The safety criteria are provided in § 53.220, and include:
 - (a) ensuring that SSCs, personnel, and programs provide capability and reliability to address LBEs other than DBAs and provide measures for defense in depth, and
 - (b) providing an appropriate level of safety relative to risk performance objectives based on comprehensive risk metrics.
- 10 CFR 53.425 establishes requirements for design features and related functional design criteria limiting doses to members of the public during normal operations to satisfy the criteria in 10 CFR Part 20, “Standards for Protection Against Radiation.”
- 10 CFR 53.430 provides similar requirements for design features and related functional design criteria for protection of plant workers to meet the safety criteria in 10 CFR Part 20.

Proposed 10 CFR Part 57, “Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles,” establishes 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. Proposed 10 CFR Part 57 would require applicants to establish PDC, which form the foundation for identifying safety-related SSCs and demonstrating adequate protection using a postulated accident consequence-based approach that bounds a broad range of design basis accidents.

- Proposed 10 CFR 57.30 would require an applicant to provide information that demonstrates that the design has design criteria attributes related to reactivity control, heat removal, fission product retention, shielding, radioactive effluent control, and security by design.
- Proposed 10 CFR 57.60(a)(2) would require an applicant to provide in the application the design basis of the facility, including the PDC, the relation of the design bases to the design criteria, and the relation of the PDC to the design criteria attributes described in proposed 10 CFR 57.30.

2. NRC Guidance and Related References

Light-Water Reactor Technologies Using a Predominantly Deterministic Approach:

As noted in 10 CFR Part 50, Appendix A, the GDC provide the scope and minimum requirements for PDC for water-cooled reactors similar in design and location to plants for which construction permits have been issued by the Commission. Historically, light-water reactors (LWRs) adopted the GDC as their PDC. However, for additional guidance on meeting the requirements of 10 CFR Part 50 and 10 CFR Part 52 stated above, applicants may consult the following:

- Section 3.1 of Regulatory Guide (RG) 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition” provides guidance on the safety analysis report format and content associated with PDC for applications submitted pursuant to 10 CFR Part 50. (ML011340072).
- NRC RG 1.206, “Applications for Nuclear Power Plants” presents guidance on the application and safety analysis report format and content for applications submitted pursuant to 10 CFR Part 52. (ML18131A181).

Non-Light-Water Reactor Technologies:

- One acceptable means of identifying PDC for a proposed non-LWR facility is contained in RG 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors” (ML17325A611).

Using a Risk-Informed and Performance-Based (Licensing Modernization Project) Approach:

- RG 1.253, “Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors” (ML23269A222) endorses, with clarifications and additions, Nuclear Energy Institute (NEI) 21-07, Revision 1, “Technology Inclusive Guidance for Non-Light Water Reactors: Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology” (ML22060A190) NEI 21-07 includes guidance for developing proposed PDC for those aspects of the facility design that focus on design functions and features that are safety related (SR) and those that are non-safety-related with special treatment (NSRST). NEI 21-07 describes an approach for developing a two-tiered structure for PDC, including functional design criteria and complementary design criteria, as defined in NEI 21-07, which also correlates with design functions classified as SR and NSRST, respectively.
- As discussed in DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap” (ML23277A139) the regulations require PDC to be developed considering aspects of the design not informed by LMP (that is, normal operations). DANU-ISG-2022-01 states that the two-tiered RG 1.253 approach is acceptable for developing proposed PDC for those design functions and features of the facility used in normal operations that are not informed by LMP but determined to be important to the protection of public health and safety. Accordingly, each applicant is responsible for identifying the need for additional PDC for facility design functions and features not informed by the LMP process that, because of the technology, design, or site, are necessary to protect public health and safety.

Microreactors Using the Proposed 10 CFR Part 57

- Section 3.1 of draft NUREG-2271, “Guidance for Preparing and Reviewing Applications Under 10 CFR Part 57” (ML26111A077), provides application and staff review guidance related to PDC for applications that would be submitted under proposed 10 CFR Part 57.

3. Summary of DOE Requirements, Orders, and Standards

DOE implements a hazard-based nuclear safety design criteria framework established through the safety basis process under 10 CFR Part 830 and supporting DOE NE Orders (Os), standards, and guides. Key aspects of this framework include:

- 10 CFR 830.206, “Preliminary Documented Safety Analysis,” requires that, prior to construction of a new or modified Hazard Category 1, 2, or 3 DOE nuclear facility, the contractor obtain DOE approval of the “nuclear safety design criteria.”
- NE-STD-1027-2025, “Hazard Categorization of DOE Nuclear Facilities,” establishes the methodology for categorizing DOE nuclear facilities as Hazard Category 1, 2, or 3 which determines the scope, rigor, and functional classification of safety SSCs and corresponding design criteria.

- NE O 420.1, "Facility Safety," establishes DOE nuclear safety design criteria, design criteria for safety SSCs, definition of associated safety functions and functional requirements, and implementation of defense-in-depth principles.
- DOE-STD-1237-2021, "Documented Safety Analysis for DOE Reactor Facilities," provides accepted methodology for development of design criteria for DOE reactor facilities.
- DOE-STD-1271-2026, "Authorization Pathway for Nuclear Facilities," establishes the authorization framework for DOE reactor facilities, including use of the Nuclear Safety Design Agreement (NSDA). The NSDA is used early in the project lifecycle to establish safety strategies, design criteria structure, and Code of Record prior to completion of the Documented Safety Analysis.
- NE-STD-1020-2025, Change 1, "Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities," establishes seismic, wind, flood, and other NPH design requirements applicable to safety SSCs. These criteria are mandatory inputs to facility design criteria development for new DOE nuclear facilities and major modifications.

4. Staff Considerations on NRC and DOE Requirements and Guidance

This section contains high-level NRC staff observations on NRC and DOE requirements and guidance and how DOE requirements and standards may be used for various NRC licensing frameworks with respect to design criteria.

Leveraging Across NRC Licensing Frameworks

The NRC staff notes that 10 CFR Parts 50 and 52, and proposed 10 CFR Part 57 all require PDC, while Part 53 uses functional design criteria tied to safety criteria in its risk-informed, performance-based framework. While there are some differences between how PDC and functional design criteria are developed, they serve substantially the same purpose across the NRC licensing frameworks. Therefore, the NRC staff notes that the NRC licensing frameworks contain substantially similar requirements and acceptable approaches with respect to design criteria, such that there are not significant differences between NRC frameworks in how DOE information is leveraged.

Considerations on NRC and DOE Requirements and Guidance

DOE requirements in 10 CFR 830 and DOE NE orders and standards describe a hazard-based nuclear safety design criteria framework tied to the safety basis of the facility. DOE requirements and guidance do not contain a design criteria framework equivalent to the general design criteria in 10 CFR Part 50 Appendix A or the advanced reactor design criteria in RG 1.232 for non-LWRs. However, DOE regulations do allow for the use of alternative guidance, where appropriate, and applicants may elect to utilize NRC guidance documents related to PDC or functional design criteria.

There are general similarities between the NRC and DOE approaches in use of design criteria to connect safety analysis to design basis and SSC requirements. One important distinction is NRC requires explicit documentation of the design criteria in the preliminary or final safety analysis report (PSAR or FSAR, respectively) and the relationship between the design criteria and design bases.

Given the cross-cutting nature of design criteria across the licensing basis, NRC staff recommends early pre-application engagement in this area with a focus on appropriate traceability and clarity in the documentation to meet NRC requirements.

Use of DOE standards for NRC licensing

Given the general similarities between the NRC and DOE approaches, use of DOE standards in the area of design criteria may be a starting point for developing an NRC licensing application. However, the applicant would need to address the key difference between the NRC and DOE frameworks and ensure that explicit documentation of the design criteria and the relationship between the design criteria and design bases is provided in the PSAR or FSAR.

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE design criteria information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

Consistent with the discussion in the main body of this white paper, this table outlines three of the key factors in determining the extent to which DOE authorization information related to design criteria may be leveraged.

Table D.1: Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	General similarities between the NRC and DOE approaches to design criteria create greater opportunities for leveraging. Applicants should provide explicit documentation of the design criteria in the PSAR or FSAR and the relationship between the design criteria and design bases and seek pre-application engagement as appropriate.
Application quality and justification of relevance of DOE authorization to NRC review	To maximize potential NRC leveraging of DOE information, application quality should be high, with design criteria identified consistent with NRC guidance and clear, explicit documentation of design criteria and how they relate to the design bases and are satisfied.
Safety-significance of information leveraged	Design criteria establish the necessary design, fabrication, construction, testing, and performance requirements SSCs that are important to safety. Therefore, they are fundamental and integral to the safety basis for the design. If DOE design criteria have not considered NRC frameworks and guidance this factor will generally limit the amount of NRC leveraging with respect to design criteria that can be achieved.

NRC Staff Approach to Leveraging DOE Design Criteria Information

NRC staff should consider the aspects of the application that were reviewed by DOE as a part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the application. The extent of any NRC leveraging opportunities, including the scope and depth of the review, would be dependent on how the applicant demonstrates compliance with applicable NRC regulatory requirements and whether the design criteria were developed utilizing the NRC guidance reflected above or are based on a proposed alternate or hybrid approach. On this basis, NRC leveraging of DOE design criteria information can generally be organized into three categories:

Maximum leveraging:

- Under maximum leveraging, NRC staff can focus their review on verifying that the design criteria and their relation to the design bases is clear and complete throughout the safety analysis. Given the cross-cutting nature of the design criteria in supporting the licensing basis, the NRC staff will review various aspects of the design criteria and how they are satisfied but can achieve significant review efficiencies, depending on the scope and depth of the DOE review and documentation.
- *Example:* An example of maximum leveraging would be when DOE authorization information provides a complete, traceable, and NRC-compatible basis for the design criteria to be brought into a PSAR or FSAR. In this example, the DOE authorization uses design criteria structured consistent with NRC design criteria guidance (e.g., Appendix A GDC, RG 1.232, or functional design criteria), with only limited deviations. The design criteria would be directly traceable to design bases, safety functions, and SSC requirements.

Moderate leveraging:

- Under moderate leveraging, NRC staff can use DOE information as a foundation, but meaningful additional work is required to reconcile differences and ensure that the resulting design criteria fully satisfy NRC requirements. Detailed NRC evaluation is needed to establish traceability between DOE criteria and NRC PDC or functional design criteria, including clarifying design bases, acceptance criteria, or bounding assumptions. The DOE analyses may also require additional NRC confirmatory analyses, sensitivity evaluations, or gap resolution to support PDC adequacy.
- *Example:* An example of moderate leveraging would be a DOE authorization that uses design criteria (or PDC) following select portions of NRC guidance and with multiple deviations and varied levels of documentation of how design criteria relate to design basis. In this example, the DOE authorization uses a different hazard-categorization or graded-approach justification that must be translated to NRC deterministic or risk-informed, performance-based, requirements. Some DOE criteria rely on DOE-specific standards, exemptions, or equivalencies that require technical justification or supplementation to align with NRC regulations.

Limited leveraging:

- Under limited leveraging, NRC staff treats DOE authorization documentation primarily as relevant background material, and the applicant must provide a substantially new NRC-aligned set of PDC and supporting technical bases.
- *Example:* An example of limited leveraging is one where DOE design criteria do not align with the structure, scope, or intent of NRC design criteria requirements (e.g., criteria are embedded in PDSA hazard analyses without clear criterion-by-criterion articulation) or the DOE authorization relies heavily on DOE-specific frameworks that do not map to NRC expectations and the applicant does not provide a crosswalk.

APPENDIX E—Source Term

1. NRC Regulatory Requirements

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, 52.17, 52.47, 52.79, 52.137, 52.157, 53.210, 53.530, and proposed 10 CFR 57.25(a), an applicant must provide source term information in its application.

10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” provides regulations for licensing commercial and non-power production and utilization facilities through a two-part construction permit (CP) and operating license (OL) process.

- 10 CFR 50.2 defines *Source term* as the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release.
- 10 CFR 50.34(a)(1)(ii)(D) requires applicants to provide an analysis of the safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be paid to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release from the core into the containment assuming that the facility is operated at the ultimate power level contemplated.
 - Footnote 3 states that the fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.
 - Proposed changes to Part 50 in the Modernizing Licensing, Safety Oversight, and Siting Practices rulemaking (Docket ID: NRC-2025-0975), if approved, would revise footnote 3 to state: The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events to bound a broad range of design basis accidents. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.
- 10 CFR 50.34(a)(4) requires, in part, that each applicant must provide an analysis and evaluation of design and performance of structures, systems, and components (SSCs) of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility.

10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” governs the issuance of early site permits (ESPs), standard design certifications (DCs), combined licenses (COLs), standard design approvals (SDAs), and manufacturing licenses (MLs) for nuclear power facilities.

- 10 CFR 52.17(a)(1)(ix) requires an early site permit applicant to provide a site safety analysis report (SAR) that contains an analysis and evaluation of the major SSCs of the facility that bear significantly on the acceptability of the site and to evaluate the offsite radiological consequences from accidents.
- 10 CFR 52.47(a)(2)(iv), 52.79(a)(1)(vi), 52.137(a)(2)(iv), and 52.157(d) require an applicant for a standard design certification, combined license, standard design approval, and manufacturing license, respectively, to perform an evaluation and analysis of the postulated

fission product release to demonstrate that the reactor facility meets the dose criteria established for safety system design as documented in the final SAR.

- Each of these regulations have a footnote stating that the fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. These accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.
- Proposed changes to Part 52 in the Modernizing Licensing, Safety Oversight, and Siting Practices rulemaking (Docket ID: NRC-2025-0975), if approved, would revise the footnotes to state: The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events to bound a broad range of design basis accidents. These accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

10 CFR Part 53, "Risk-Informed, Technology Inclusive Regulatory Framework for Commercial Nuclear Plants," provides a technology-inclusive, performance-based framework for ESPs, CPs, COLs, DCs, SDAs, and MLs for commercial nuclear plants.

- 10 CFR 53.020 defines that *Fission product release* means the amount and composition of radioactive material released to the environment, after accounting for any retention of radionuclides provided by reactor design features.
- 10 CFR 53.210 requires, in part, that an applicant provide an analysis of the design features and programmatic controls for each commercial nuclear plant that demonstrate for design basis accidents (DBAs) that the offsite consequences do not exceed 25 rem total effective dose equivalent (TEDE) for any 2-hour period to an individual at the exclusion area boundary (EAB) and for the entire duration of the postulated fission product release to an individual at the outer boundary of the low population zone (LPZ).
- 10 CFR 53.530 includes requirements related to the development of a source term to evaluate population related considerations.

Proposed 10 CFR Part 57, "Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles," includes proposed provisions for CPs, OLs, MLs, SDAs, and for affording regulatory finality to nuclear plant designs and essentially complete standardized operational programs. Proposed 10 CFR Part 57 would include the following provisions:

- Proposed 10 CFR 57.3 would define that:
 - *Source term* means the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release.
 - *Fission product release* means the amount and composition of radioactive material released to the environment, after accounting for any retention of radionuclides provided by reactor design features.
- Proposed 10 CFR 57.25(a) would require, to be eligible for a construction permit and operating license or a manufacturing license under proposed 10 CFR Part 57, that an applicant demonstrate, in part, that an evaluation of the applicable radiological consequences shows with reasonable assurance that any individual located in the unrestricted area following the onset of a postulated accident that bounds a broad range of DBAs would not exceed 1 rem (0.01 Sv) TEDE for the duration of the accident. Guidance for the analysis for the dose-based entry criterion describes that the applicant may use either a maximum hypothetical accident (MHA) or maximum credible accident (MCA) approach (see "Microreactors Using the Proposed 10 CFR Part 57 Approach" section below).

- Proposed 10 CFR 57.60(a)(1)(iii) would require that the applicant provide an evaluation that meets the dose-based entry criterion of 10 CFR 57.25(a). In performing this evaluation, an applicant must assume a fission product release utilizing a postulated accident source term that represents the most limiting fission product inventory during the lifetime of the nuclear reactor while assuming that the reactor is operated at the ultimate power level contemplated.

10 CFR Part 100, "Reactor Site Criteria," establishes criteria that must be met to ensure that the site is suitable from a radiological safety perspective for applications for power and testing reactors subject to 10 CFR Part 50 or Part 52.

2. NRC Guidance and Related References

Light-Water Reactor (LWR) Technologies Using a Predominantly Deterministic Approach:

To meet the requirements of 10 CFR Part 50 and 10 CFR Part 52 stated above, applicants may consult the following:

- NRC Regulatory Guide (RG) 1.183, Revision 1, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (ML23082A305), presents representative MHA loss-of-coolant accident (LOCA) source terms for a boiling-water reactor and for a pressurized-water reactor derived from physically based evaluations of LWR severe accidents. Regulatory Position 2 of RG 1.183 provides attributes of an acceptable accident source term.
 - An applicant may propose an accident source term that differs in source term parameters (e.g., timing, release magnitude, and chemical form) from those provided, based on and justified by design-specific features. For example, developers of light-water small modular reactors may use the accident source term (as applicable to the design) or propose a design-specific accident source term.
 - While RG 1.183, Revision 1, is the latest revision, RG 1.183, Revision 0 (ML003716792), is acceptable if the plant fits within its applicability, as noted in the background section of RG 1.183, Revision 1.
- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (ML052340514) Chapter 15, "Transient and Accident Analysis."
 - Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," (ML070230012) cites the use of the guidance contained in RG 1.183.
- NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" (ML12251A353).
 - Chapter 13, "Accident Analysis," presents a representative MHA source term for analyses of non-power reactor credible accidents.

Non-Light-Water Reactor Technologies Using a Predominantly Deterministic Approach:

- While RG 1.183 is focused on LWRs and is not appropriate for direct application to non-LWRs, RG 1.183 includes useful information that may be adapted to aid in design-specific accident source term development for non-LWR technologies. Specifically, Regulatory Position 2 of RG 1.183 states attributes of an acceptable accident source term.
- Applicants developing deterministic accident source terms may, in general, use conservative assumptions to bound the expected plant response to credible accidents, such as for the MHA concept discussed in NUREG-1537, Part 1, Chapter 13.

- When developing accident source terms, applicants may also use more detailed mechanistic analyses of the plant response for a range of accidents, including severe accidents, as a technical basis for the conservatively biased modeling and assumptions such as in the development of the LWR source terms described in RG 1.183.
 - In Staff Requirements Memorandum (SRM)-SECY-93-092, “SECY-93-092—Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements” (ML003760774), the Commission approved the NRC staff’s recommendation that source terms for non-LWRs be based on a mechanistic analysis. The acceptability of the applicant’s analysis should be based on the NRC staff’s assurance that the following conditions are met:
 - The performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist on the reactor and fuel performance through the research, development, and testing programs to provide adequate confidence in the mechanistic approach.
 - The transport of fission products can be adequately modeled for all barriers and pathways to the environs, including the specific consideration of the containment design. The calculations should be as realistic as possible so that the values and limitations of any mechanism or barrier are not obscured.
 - The events considered in the analyses to develop the set of source terms for each design are selected to bound severe accidents and design-dependent uncertainties.
 - The design-specific source terms for each accident category would constitute one component for evaluating the acceptability of the design.

Reactors Using a Risk-Informed and Performance-Based (Licensing Modernization Project) Approach:

- RG 1.253, “Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors” (ML23269A222) indicates that the applicant should provide information on event sequence source terms specific to its design used in the licensing-basis event consequence analyses. The guidance states that the source term information should cover all radioactive material inventories and include the type, quantity, and timing of the release of radioactive material from the facility during licensing-basis events. The application should include the use of a mechanistic source term (MST), consistent with the advanced non-LWR probabilistic risk assessment (PRA) standard definition (see appendix A to NEI 21-07, “Technology Inclusive Guidance for Non-Light Water Reactors: Safety Analysis Report Content for Applicants Using the NEI 18-04 Approach”).
- RG 1.247 (for Trial Use), “Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities” (ML21235A008), endorses, with exceptions for trial use, American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-S-1.4-2021, “Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants,” as one acceptable approach to a PRA for use in regulatory decision-making for non-LWRs. RG 1.247 Regulatory Position 2.C.1.3.16 describes the MST analysis PRA element, and table A-17 indicates that the NRC staff had no objections to the positions in the non-LWR PRA standard. Applicants may find that the PRA standard provides helpful information pertaining to source term phenomena.

- RG 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors” (ML20091L698), endorses the Licensing Modernization Project (LMP) methodology in NEI 18-04, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development”. With respect to MST this RG provides that the established methods for addressing radiological source terms for LWRs have limited applicability to non-LWR designs and more mechanistic approaches have been proposed. This document references SRM-SECY-93-092, where the Commission approved the NRC staff’s recommendation that source terms for non-LWRs be based upon a mechanistic analysis. This document also provides information related to non-LWR SSCs having specific barrier functions. The barrier functions are important in the development and assessment of mechanistic source terms that help determine the offsite doses for non-LWR designs for each licensing basis event. Performance criteria for a barrier, or set of barriers taken together, to the transport of radionuclides are discussed in SECY-18-0096 “Functional Containment Performance Criteria For Non-Light-Water-Reactors” (ML18114A546).
- Part 53 applicants may be able to leverage LMP guidance in NEI 18-04, NEI 21-07, and NEI 24-05, “An Approach for Risk-Informed Performance-Based Emergency Planning.”

Microreactors Using the Proposed 10 CFR Part 57 Approach:

- For applications under proposed Part 57, NUREG-2271 (draft), “Guidelines for Preparing and Reviewing Applications Under 10 CFR Part 57” (ML26111A077), Section 1.1.12 and Chapter 12, describe the MHA or MCA consequence analysis, with more specific guidance for developing advanced reactor source terms in Section 12.5 and Appendix C. NUREG-2271, Section 2.2 and Appendix F, describe the unmitigated consequence analysis to determine nuclear siting category (NSC) for a graded siting approach. Provisions of proposed 10 CFR Part 57 are not final and subject to change at the final rule stage.

Other Resources:

- The NRC’s public website includes a web page that provides resources related to nuclear reactor source term: <https://www.nrc.gov/facilities-safety/new-reactors/advanced-reactors/references-and-training/nuclear-power-reactor-source-term>

3. Summary of DOE Requirements, Orders, and Standards

10 CFR Part 830, “Nuclear Safety Management,” and 10 CFR Part 835, “Occupational Radiation Protection,” set the regulatory requirements for radiation protection. 10 CFR 830.204(b)(3) states that the documented safety analysis for a DOE nuclear facility must evaluate normal, abnormal, and accident conditions, including consideration of natural and manmade external events, identification of energy sources or processes that might contribute to the generation or uncontrolled release of radioactive and other hazardous materials, and consideration of the need for analysis of accidents which may be beyond the design basis of the facility. The relevant DOE documents include DOE-STD-1237-2021, “Documented Safety Analysis for DOE Reactor Facilities,” which refers to DOE-STD-3009-2014, “Preparation of Nonreactor Nuclear Facility Documented Safety Analysis,” DOE-HDBK-3010-94, “Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities,” and DOE-STD-1271-2026, “Authorization Pathway for Nuclear Facilities.”

DOE-STD-3009-2014 and DOE-HDBK-3010-94 provide information on the five-factor formula method for determining a radiological source term. DOE-HDBK-3010-94 provides information for two of the five factors for nonreactor nuclear facilities.

4. Staff Considerations on NRC and DOE Requirements and Guidance

This section contains high-level NRC staff observations on NRC and DOE requirements and guidance and how DOE requirements and standards may be used for various NRC licensing frameworks with respect to source term development.

Leveraging Across NRC Licensing Frameworks

The analyses and evaluations of postulated fission product releases to be provided in applications for permits, certifications, and licenses under 10 CFR Parts 50 and 52 are similar. LWR guidance under 10 CFR Parts 50 and 52 suggests the use of an MHA and leverages conservative approaches to source term. The NRC-endorsed NEI 18-04 methodology for non-LWRs licensing under 10 CFR Parts 50 or 52 uses mechanistic source terms to evaluate licensing basis events and includes conservative evaluation of DBAs. 10 CFR Part 53 and its supporting guidance suggest the use of a mechanistic approach to source term for source term development for licensing basis events other than DBAs, and use of deterministic conservative methods for evaluation of DBAs. Proposed 10 CFR Part 57, if approved, could also leverage approaches similar to the 10 CFR Parts 50, 52, and 53 frameworks but requires use of an MHA or MCA approach to clarify the use of conservative best estimate or conservative but realistic calculations. The NRC licensing frameworks contain substantially similar requirements and acceptable approaches with respect to source term, such that there are not significant differences between NRC frameworks in how DOE information is leveraged.

Considerations on NRC and DOE Requirements and Guidance

There are general similarities between the NRC and DOE approaches in the calculation of reactor source term. The proposed use of an MHA or MCA should be clearly stated so that the NRC staff can verify that justifications made are consistent with the licensing framework being used to meet NRC regulations. The DOE method of solving for source term uses the five-factor formula as described in DOE-STD-3009-2014 and DOE-HDBK-3010-94. The NRC staff note that the five-factor formula has similarities to calculating source terms using an MST development approach (consistent with SRM-SECY-93-092), which may be applicable, if the factors used are justified. DOE guidance for the five-factor formula does not provide airborne release fractions (ARF) and respirable fractions (RF) for reactor fuels but describing and justifying an MST could leverage phenomena to accomplish similar approaches in calculating source term.

The NRC staff notes that other topical areas addressed in this white paper may influence source term, and applicants should consider underlying assumptions associated with those topical areas. For example, Appendix L of this white paper provides additional insight into the fuel qualification topical area. This topical area will help bound some of the assumptions that provide insight into assumed manufacturing defects for fuel which will inform the technical basis for source term.

Use of DOE Standards for NRC Licensing

As previously mentioned, high level similarities exist between the approaches in the calculation of reactor source terms for NRC and DOE applications. For example, if an applicant uses RG 1.183 for a DOE authorization, then there would be an opportunity to leverage information developed during

the DOE authorization within the application to the NRC. However, for non-LWR designs, sufficient documentation is needed to explain MST approaches. NRC staff would expect uncertainty to be quantified for non-conservative approaches to source term development, which may differ from provisions in DOE-STD-3009-2014. The NRC staff recommends early pre-application engagement in this area given the cross-cutting nature of source term across the design, and to facilitate a well-documented and justified approach to source term development.

5. Leveraging DOE Authorization Information in NRC License Reviews

The extent to which DOE source term information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

The information below describes how NRC staff will assess key factors for NRC leveraging of DOE authorization information related to source term and provide examples of scenarios that might lead to maximum, moderate, or limited leveraging.

Table E.1: Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	General similarities between NRC and DOE approaches to source term. Approaches would require documentation and justifications to ensure a level of understanding to compare approaches. The guidance in RG 1.183 can be used for LWR reactor accident source term calculations. Non-LWR guidance for NRC reviews indicates the use of a mechanistic approach for source term. DOE-STD-3009-2014 and DOE-HDBK-3010-94 discusses an approach for developing source terms for DOE applications. Some similarities could be drawn to an MST approach in which the identification of relevant phenomena is provided along with models used to represent phenomena impact on source term. NRC staff should review the details for determining source term based on the technical basis provided in support of licensing activities following DOE-STD-3009-2014.
Application quality and justification of relevance of DOE authorization to NRC review	Use of DOE-STD-3009-2014, along with detailed explanations of the approach used, may be appropriate for use in NRC applications given that the source term provided is supported by a first principles approach to determining source term, where the models and assumptions are clearly described. ARF and RF will need specific justifications given that DOE-HDBK-3010-94 does not provide fractions for use at reactor facilities. Other methods for determining source terms that are acceptable to NRC staff are dependent on fuel type and the regulatory framework being applied for reactor design. The NRC staff encourages pre-application engagement to ensure source term methodologies used in DOE authorizations align with the planned regulatory framework for NRC license application. At the

	NRC application phase, NRC staff review should ensure that sufficient justification for source term development is provided if it departs from RG 1.183, or if it presents an MST approach.
Safety-significance of information leveraged	Source term information is necessary to support consequence analysis that justifies sufficient systems are incorporated by the reactor design to prevent and mitigate radiological releases from the reactor facility. A well-documented source term gives confidence that appropriate credit of safety systems is established by the design.

NRC Staff Approach to Leveraging DOE Source Term Information

NRC staff should consider the aspects of the application that were reviewed by either DOE as a part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the application. The extent of NRC leveraging opportunities, including to inform review scope and depth, would be dependent on how the applicant demonstrates compliance with applicable NRC regulatory requirements and whether the source term was developed using the NRC guidance reflected above or was based on a proposed alternate or hybrid approach. On this basis, NRC leveraging of DOE source term information can generally be organized into three categories:

Maximum leveraging:

- Under a maximum leveraging scenario, technical information developed or evaluated as part of a DOE authorization may provide a direct basis for an NRC application when the underlying information remains valid and the DOE authorization provides a clear method for determining reactor source term following RG 1.183 or a well-documented MST. NRC staff can focus their review on verifying that the method and justification for source term development as well as description and justification of any cross-cutting underlying assumptions (e.g., fuel degradation) are clear and complete throughout the application.

Moderate leveraging:

- Under a moderate leveraging scenario, source term information from a DOE authorization may form part of the basis for an NRC application, but additional work is required to reconcile or justify departures from RG 1.183, or to enhance justification for use of an MST approach. For example, justification may be needed within the application to support the level of conservatism stated or have source term uncertainties quantified. In this case, NRC staff review is needed to verify that the approach and supplemental information meet NRC requirements, and the DOE authorization information may require NRC staff confirmatory analyses, sensitivity evaluations, or gap resolution to support source term adequacy.

Limited leveraging:

- Under a minimal-leveraging scenario, the DOE authorization does not follow RG 1.183 and does not present an MST methodology. High level assumptions are stated with little or no documentation available for review within the DOE authorization, or the DOE authorization has a significantly different scope (i.e., DOE authorized reactor is for experimental application whereas an application to the NRC is for power generation), limiting applicability of the source term developed for the DOE authorization to the NRC application. In this case, NRC staff should treat DOE authorization information primarily as relevant background material, and the applicant should provide a substantially new NRC-aligned source term.

APPENDIX F—Consequence Analysis

1. NRC Regulatory Requirements

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, 52.47, 52.79, 52.137, 52.157, 53.210, and 53.530, and proposed 10 CFR 57.25(a), an applicant must provide consequence analysis information in its application.

10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” provides regulations for licensing commercial and non-power production and utilization facilities.

- 10 CFR 50.34(a)(1)(ii)(D) requires applicants to provide description and safety assessment of the site and the facility. The evaluation must determine that the radiological consequences are such that:
 - (1) an individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem (0.25 Sv) total effective dose equivalent (TEDE) and,
 - (2) an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem (0.25 Sv) TEDE.
- 10 CFR 50.34(a)(4) requires, in part, that each applicant must provide an analysis and evaluation of design and performance of structures, systems, and components (SSCs) of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility
- 10 CFR 50.34(f) provides additional Three Mile Island (TMI)-related requirements and states that each applicant for a combined license shall demonstrate compliance with the technically relevant portions of the requirements, except for paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). The TMI-related requirements that refer to accident source term releases and may rely on radiological consequence analysis, if technically relevant, and the related topic areas are as follows:
 - 10 CFR 50.34(f)(2)(vii)—postaccident access shielding
 - 10 CFR 50.34(f)(2)(viii)—postaccident sampling capability
 - 10 CFR 50.34(f)(2)(xvii)—accident monitoring instrumentation
 - 10 CFR 50.34(f)(2)(xxvi)—leakage control
 - 10 CFR 50.34(f)(2)(xxviii)—control room habitability
- 10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants,” relates to qualification of safety-related equipment with regard to integrated radiation dose during normal and accident conditions.
- 10 CFR 50.160, “Emergency preparedness for small modular reactors, non-light water reactors, and non-power production or utilization facilities,” is an alternative to emergency planning for small modular reactors, non-LWRs, and non-power production or utilization facilities. To comply with 10 CFR 50.160, 10 CFR 50.33(g)(2) describes sizing criteria for a plume exposure pathway emergency planning zone (EPZ), which requires a consequence analysis to aid in the EPZ determination. Applicants otherwise comply with 10 CFR 50.47(b) and 10 CFR Part 50, appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” with consequence analyses as necessary to demonstrate that habitability criteria for emergency response facilities are met as described in RG 1.183 and NUREG-0696, “Functional Criteria for Emergency Response Facilities.” Proposed changes to the emergency preparedness requirements in the Modernizing Licensing, Safety

Oversight, and Siting Practices rulemaking (Docket ID: NRC-2025-0975) would expand the applicability of 10 CFR 50.160 and the performance-based, technology-inclusive, risk-informed, and consequence-oriented alternative emergency preparedness framework to large light-water reactors (LWRs).

- 10 CFR 50.34(a)(3)(i) requires that the preliminary safety analysis report (SAR) includes the principal design criteria (PDC) for the facility. 10 CFR Part 50, appendix A, “General Design Criteria for Nuclear Power Plants,” contains the general design criteria (GDC) that establish the minimum requirements for the PDC for water-cooled nuclear power plants. Appendix A to 10 CFR Part 50 also establishes that the GDC are generally applicable to other types of nuclear power units and are intended to provide guidance in determining the PDC for such other units. GDC 19, “Control room,” establishes a control room design radiological habitability dose criterion of 5 rem (0.05 Sv) TEDE. Proposed changes to GDC 19 in the Modernizing Licensing, Safety Oversight, and Siting Practices rulemaking (Docket ID: NRC-2025-0975) would increase the numerical value of the control room design dose criterion from 5 rem to 10 rem (0.05 to 0.10 Sv); the value may range from 10 rem up to 25 rem (0.10 Sv up to 0.25 Sv) TEDE with consideration of the plant-specific risk profile or risk information.

10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities.

- 10 CFR 52.17(a)(1)(ix) requires an early site permit applicant to provide a site SAR that contains an analysis and evaluation of the major SSCs of the facility that bear significantly on the acceptability of the site and to evaluate the offsite radiological consequences from accidents.
- 10 CFR 52.47(a)(2)(iv), 52.79(a)(1)(vi), 52.137(a)(2)(iv), and 52.157(d) require an applicant for a standard design certification, combined license, standard design approval, and manufacturing license, respectively, to perform an evaluation and analysis of the postulated fission product release to demonstrate that the reactor facility meets the dose criteria established for safety system design as documented in the final SAR.
- For each of these requirements, the accident analysis offsite dose reference values are the same as given in 10 CFR 50.34(a)(1)(ii)(D).

10 CFR Part 53, “Risk-Informed, Technology Inclusive Regulatory Framework for Commercial Nuclear Plants,” provides a technology-inclusive, performance-based framework for the issuance, amendment, renewal, and termination of licenses, permits, certifications, and approvals for commercial nuclear plants.

- 10 CFR 53.210, “Safety criteria for design-basis accidents,” requires, in part, that an applicant provide an analysis of the design features and programmatic controls for each commercial nuclear plant that demonstrate for design basis accidents that the offsite consequences do not exceed 25 rem (0.25 Sv) TEDE for any 2-hour period to an individual at the exclusion area boundary (EAB) and do not exceed 25 rem (0.25 Sv) TEDE for the entire duration of the plume passage to an individual at the outer boundary of the low population zone (LPZ).
- 10 CFR 53.530, “Population-related considerations,” also requires that the analysis of design basis accidents (DBAs) must be used to confirm that the radiological consequences do not exceed 25 rem (0.25 Sv) TEDE for any 2-hour period to an individual at the EAB and do not exceed 25 rem (0.25 Sv) TEDE for the entire duration of the plume passage to an individual at the outer boundary of the LPZ.

Proposed 10 CFR Part 57, “Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles,” would provide 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.

- Proposed 10 CFR 57.25(a) describes a dose-based entry criterion using a maximum hypothetical accident (MHA) or maximum credible accident (MCA) approach. An applicant would be required to demonstrate that an evaluation of the applicable radiological consequences shows with reasonable assurance that any individual located in the unrestricted area following the onset of a postulated accident that bounds a broad range of design basis accidents would not exceed 1 rem (0.01 Sv) TEDE for the duration of the accident.
- Proposed 10 CFR 57.60(a)(1)(iii) would require that the applicant provide an evaluation that meets the dose-based entry criterion of proposed 10 CFR 57.25(a). In performing this evaluation, an applicant would be required to assume a fission product release utilizing a postulated accident source term that represents the most limiting fission product inventory during the lifetime of the nuclear reactor while assuming that the reactor is operated at the ultimate power level contemplated.

10 CFR Part 100, “Reactor Site Criteria,” establishes criteria that must be met to ensure that the site is suitable from a radiological safety perspective.

- In 10 CFR 100.21, “Non-seismic siting criteria,” the NRC requires that the site characteristics, including population density, meteorology, and proximity to population centers, be evaluated in conjunction with the potential radiological consequences of postulated accidents. The analysis must demonstrate that the site can support the implementation of emergency plans and that radiological consequences of postulated accidents meet the values given in 10 CFR 50.34(a)(1) at the EAB and outer boundary of the LPZ.
- Proposed changes to Part 100 in the Modernizing Licensing, Safety Oversight, and Siting Practices rulemaking (Docket ID: NRC-2025-0975) would, in part:
 - Make Subpart A of Part 100 applicable to testing facilities and nuclear power reactor facilities with unmitigated consequences < 25 rem TEDE at EAB and revise the 10 CFR 100.11 dose reference values to be 25 rem TEDE for any 2-hour period to an individual at the EAB and for the entire duration of plume passage to an individual at the outer boundary of the LPZ
 - Make Subpart B of Part 100 applicable to power reactor facilities with unmitigated consequences at EAB ≥ 25 rem TEDE.

2. NRC Guidance and Related References

LWR Technologies Using a Predominantly Deterministic Approach:

To meet the requirements of 10 CFR Part 50 and 10 CFR Part 52 stated above, applicants may consult the following:

- NRC Regulatory Guide (RG) 1.183, Revision 0, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” issued July 2000 (ML003716792), presents guidance and assumptions to use for consequence analysis.
 - For the methodology and assumptions for calculating the radiological consequences of postulated accidents, the NRC staff will use, where applicable, the regulatory positions stated in RG 1.183 (specifically, section C.4, which is technology inclusive). The reviewer should confirm that the applicant’s approach is clearly documented;

should review the associated codes and methods that were applied, including input and outputs; and should consider performing scoping calculations.

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (ML070810350) describes the review of accident analyses as historically used in the licensing of LWR power reactors. The discussion in chapter 15, "Transient and Accident Analysis," includes the DBA consequence analyses, with some additional support for evaluation of control room habitability in section 6.4, "Control Room Habitability System." In particular, section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," (ML070230012) describes the review of consequence analyses that use the guidance in RG 1.183, Revision 0, for LWR source terms for a new LWR application under 10 CFR Part 50 or 10 CFR Part 52.

Non-Light-Water Reactor Technologies Using a Predominantly Deterministic Approach:

- RG 1.183 may provide useful information for satisfying the radiological dose analysis requirements in 10 CFR Part 50 and 10 CFR Part 52 for new non-LWR applicants. Although the source term information is specific to LWR designs similar to those currently operating, this guide could provide useful information on radiological consequence analysis for non-LWR designs. The consequence analysis portions (section C.4) are not technology specific.
- NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," (ML042430048) chapter 13, "Accident Analyses," describes the postulated MHA approach and provides guidance for research and test reactor accident and consequence analyses that may be useful to applicants seeking class 103 commercial licenses. The MHA refers to a postulated accident scenario whose potential consequences are shown to exceed and bound all credible accidents. The specific dose locations and dose acceptance criteria discussed in NUREG-1537 are not applicable to power reactors.

Using a Risk-Informed and Performance-Based (Licensing Modernization Project) Approach:

- RG 1.253, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses Certifications, and Approvals for Non-Light-Water Reactors," (ML23269A222) endorses, with clarifications and additions, NEI 21-07, "Technology Inclusive Guidance for Non-Light Water Reactors Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology,"
 - RG 1.253, section C.2 of NEI 21-07 provides acceptable guidance on the discussion of the software and analytical tools used to (1) perform the event sequence modeling and quantification, (2) determine the mechanistic source terms, and (3) perform radiological consequence evaluations for the non-DBA licensing basis events and DBAs listed in section C.3 of NEI 21-07, and the cumulative dose and risk calculations in section C.4.1 of NEI 21-07. Applicants should discuss the analysis methods and assumptions for the total calculated radiological consequence dose at the EAB, the outer boundary of the LPZ, and the control room, if required (e.g., if operator actions are relied on for safety-significant functions) to demonstrate that the facility meets the requirements of 10 CFR 50.34(a)(1)(ii)(D) or 10 CFR 52.79(a)(1)(vi) and the PDC for the control room (if applicable). An applicant is free to propose different approaches. The LMP methodology requires that:
 - (1) each identified anticipated operational occurrence, design-basis event, and beyond-design-basis event that releases radioactive material to the environment be realistically analyzed to determine its 30-day cumulative

- TEDE, plus uncertainties, at the EAB to evaluate against a frequency-consequence target curve,
- (2) DBAs derived from the licensing basis events (LBEs) be evaluated deterministically with conservative assumptions for comparison with the offsite dose criteria at the EAB and LPZ,
 - (3) the integrated risk from all events considered in the probabilistic risk assessment (PRA) be evaluated against integrated risk targets based on the Commission's Quantitative Health Objectives (i.e., the annual average individual risk of early and latent fatalities), reported as mean risk without uncertainty, and
 - (4) the cumulative frequency of exceeding 100 mrem per plant-year at the EAB be less than 1.0/plant-year, reported as mean risk without uncertainty.
- RG 1.253, section C.3, discusses the LMP approach for consequence analysis. In addition, RG 1.183 may provide generally useful information on radiological consequence analysis for non-LWR designs. RG 1.183, section C.4 is not technology specific and therefore provides guidance for non-LWR dose calculations.
 - RG 1.247 "Acceptability of Probabilistic Risk Assessment Results for Non-Light Water Reactor Risk-Informed Activities," (ML21235A008), which endorses, with conditions, the non-LWR PRA standard American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants". RG 1.247, section C.1.3.17, provides guidance on the radiological consequence analysis PRA element, while table A-18 provides the specific staff position on the ASME/ANS RA-S-1.4-2021 technical requirements for radiological consequence analysis.
 - RG 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology To Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors," (ML20091L698) endorses, with clarifications, the methodology described in NEI 18-04.
 - NUREG-0800, section 15.0.3, describes the review of DBA consequence analyses for new LWR reactor applications. Although the guidance was developed for LWRs and refers to RG 1.183, it provides information that may be useful for non-LWR applications. Although accident source term guidance is applicable to LWR designs, the regulatory requirements, concepts of the deterministic evaluation, and the radiological consequence methods are not technology specific.
 - Part 53 applicants may be able to leverage LMP guidance in NEI 18-04, NEI 21-07, and NEI 24-05. Staff is developing guidance for review of applications under Part 53.

Proposed 10 CFR Part 57 Approach:

- For applications under proposed Part 57, NUREG-2271 (draft), "Guidelines for Preparing and Reviewing Applications Under 10 CFR Part 57" (ML26111A077), chapter 12 describes the consequence analysis for the MHA or MCA. NUREG-2271 section 9.10 describes the consequence analysis for transportable reactors. NUREG-2271 section 2.2 and appendix F describe the unmitigated consequence analysis to determine nuclear siting category (NSC) for a graded siting approach.

3. Summary of DOE Requirements, Orders, and Standards

10 CFR Part 830, "Nuclear Safety Management," governs conduct that may affect the safety of DOE nuclear facilities. 10 CFR 830.204(a) states that DOE must approve the methodology used to

prepare the documented safety analysis (DSA) for the facility unless the contractor uses a methodology set forth in table 1 of appendix A to Part 830. 10 CFR 830.204(b)(3) states that the DSA for a DOE nuclear facility must evaluate normal, abnormal, and accident conditions, including consideration of natural and manmade external events, identification of energy sources or processes that might contribute to the generation or uncontrolled release of radioactive and other hazardous materials, and consideration of the need for analysis of accidents which may be beyond the design basis of the facility. 10 CFR 835 "Occupational Radiation Protection", implemented via DOE Order NE O 458.1, "Radiation Protection of the Public and the Environment," establishes requirements for protecting the public and the environment. Order DOE O 420.1D, "Facility Safety," establishes facility and programmatic safety requirements for nuclear facilities authorized by the DOE. DOE P 420.3, "Nuclear Safety Policy for Offsite Nuclear Facilities Authorized by the Office of Nuclear Energy," establishes and maintains an authorization pathway for offsite nuclear facilities that are safely operational in a manner that provides reasonable assurance of adequate protection of workers, the public, and the environment. DOE P 420.2, "Nuclear Safety Policy for Onsite Nuclear Facilities Authorized by the Office of Nuclear Energy," is the same for onsite nuclear facilities. The relevant DOE standards include DOE-STD-1237-2021, "Documented Safety Analysis for DOE Reactor Facilities," and DOE-STD-3009-2014, "Preparation of Nonreactor Nuclear Facility Documented Safety Analysis."

DOE O 420.1D and DOE-STD-1237-2021 provide reference to DOE-STD-3009-2014 as an applicable standard based on the types of reactors. DOE-STD-3009-2014 section A.10 describes the concept of an evaluation guideline to help determine the rigor of controls (including defense-in-depth) needed to avoid the potential dose from an accident, the level of planning necessary to respond to given accidents, or the training needed for individuals that may be placed in situations where such doses might occur. The evaluation guideline is established for SSC classification. A design criterion of 25 rem is used to identify and define what measures and controls are necessary.

DOE Order 458.1 provides the public dose limits of 100 mrem TED in a year, 1500 mrem lens of the eye in a year, 5000 mrem skin or extremity dose in a year, from all sources of ionizing radiation and exposure pathways that could contribute significantly to the total dose with exceptions described in the order.

DOE Policy 420.3 states that adequate protection Evaluation Guideline remains at 25 rem. The risk to an average individual member of the public in the vicinity of a nuclear facility for prompt fatalities that might result from accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which the public is generally exposed. For evaluation purposes, individuals are assumed to be located within one mile of the site boundary. In addition, the design goal for new nuclear facilities is that the worst-case DBA (greater than 10E-6 events/year) will be 5 rem at the facility's public access boundary.

4. Staff Considerations on NRC and DOE Requirements and Guidance

This section contains high-level NRC staff observations on NRC and DOE requirements and guidance and how DOE requirements and standards may be used for various NRC licensing frameworks with respect to consequence analysis.

Leveraging Across NRC Licensing Frameworks

The analyses and evaluations of postulated fission product releases to be provided in applications for permits, certifications, and licenses under 10 CFR Parts 50 and 52 are similar. LWR guidance

under 10 CFR Parts 50 and 52 suggests the use of an MHA and leverages conservative approaches to consequence analysis leveraging the guidance contained in RG 1.183. The NRC-endorsed NEI 18-04 methodology for non-LWR applications under 10 CFR Parts 50 and 52 uses consequence analysis information but requires the frequency and consequence uncertainty information needed for LMP which requires 5th, mean, and 95th uncertainties for each evaluated non-DBA LBE, in addition to the conservative deterministic analysis of DBAs. 10 CFR Part 53 requires evaluation of the offsite radiological consequences of DBAs using conservative approaches, as well as analysis of risks to public health and safety resulting from LBEs other than DBAs to evaluate against comprehensive risk metrics that satisfy associated risk performance objectives, which may include evaluation of radiological consequences. Proposed 10 CFR Part 57 could also leverage approaches similar to the 10 CFR Parts 50, 52, and 53 frameworks, but would require use of an MHA or MCA approach to clarify the use of conservative best estimate or conservative but realistic calculations. The NRC staff notes that the NRC licensing frameworks contain substantially similar requirements and acceptable approaches with respect to consequence analysis. As highlighted above, there may be differences in an MHA or MCA approach and additional consequence information may be required to support other frameworks such as LMP or to support more realistic consequence analysis when using an MCA approach.

Considerations on NRC and DOE Requirements and Guidance

There are general similarities between the NRC and DOE approaches to consequence analyses. If an applicant uses RG 1.183 as guidance in developing their analyses for a DOE authorization, then there would be significant alignments between the DOE and NRC approach to consequence analysis. Departures from the consequence assumptions described in RG 1.183 section C.4 should be justified. The DOE approach in calculating TED is sufficient for comparison to NRC dose criteria determined using TEDE.

Use of DOE Standards for NRC Licensing

DOE Order NE O 458.1A requires that the general public would not receive dose exceeding 100 mrem TED in a year, 1500 mrem lens of the eye in a year, 5000 mrem skin or extremity dose in a year, from all sources of ionizing radiation and exposure pathways that could contribute significantly to the total dose with exceptions described in the order. The criteria described in this order appears consistent with, or more restrictive than requirements in 10 CFR Part 20, "Standards for Protection Against Radiation," with the acknowledgments that NRC used TEDE in place of TED as mentioned above.

DOE-STD-3009-2014 discusses consequence calculations in section 3.2.4, which refers to NUREG-1140, "A regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees," as basis for most cases only evaluating the airborne pathway since inhalation pathway will dominate the (overall) dose. This is consistent with the NRC approach, however, there are cases where direct shine and ground shine are evaluated for control room habitability. DOE-STD-3009-2014 also refers to consequence levels in table 1, which provides public and co-located worker numerical dose criteria. The dose criteria presented for public exposure is the same or lower than the dose criteria in NRC regulations. The facility worker dose criteria provided in DOE guidance would need to be adjusted to suit NRC regulations regarding control room habitability. DOE-STD-3009-2014 refers to release and exposure durations that account for 2 hours of exposure and the duration of plume passage which appears consistent with NRC approaches in quantifying dose. DOE-STD-3009-2014 also refers to RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," which is an NRC-approved

method for determining atmospheric dispersion for potential accident consequence assessment. The NRC staff note that RG 1.249, "Use of ARCON Methodology for Calculation of Accident-Related Offsite Atmospheric Dispersion Factors," is an appropriate alternative for calculating atmospheric dispersion for potential accidents and RG 1.194 "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," for the control room location for atmospheric dispersion. DOE-STD-3009-2014 section A.10 describes the importance of the consequence analysis in determining SSC classifications similar to NRC staff processes.

DOE P 420.3 reinforces and improves on the evaluation guidelines discussed by DOE-STD-3009-2014. The dose criteria leveraged by DOE-STD-3009-2014 and DOE P 420.3 appear consistent with NRC requirements in evaluating sufficient systems are in place to mitigate radiological releases from the reactor facility.

For consequence analysis under an LMP framework (using guidance contained in RGs 1.233 and 1.253), additional information would be required to address the frequency of the LBEs alongside the corresponding dose calculated for the LBE. In addition, dose consequences for 30-day TEDE to an individual at the EAB would be needed for comparison to the frequency-consequence curve required under the LMP methodology. The frequency and consequence information needed for LMP requires 5th, mean, and 95th uncertainties for each postulated LBE.

DOE P 420.2 and DOE P 420.3 provide the quantitative risk objective where the risk to an average individual member of the public in the vicinity of a nuclear facility for prompt fatalities that might result from accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which the public is generally exposed. For evaluation purposes, individuals are assumed to be located within one mile of the site boundary. NRC's risk assessment under LMP requires the following:

- (1) Each identified anticipated operational occurrence, design-basis event, and beyond-design-basis event that releases radioactive material to the environment be realistically analyzed to determine its 30-day cumulative TEDE, plus uncertainties, at the EAB to evaluate against a frequency-consequence target curve,
- (2) DBAs derived from the LBEs be evaluated deterministically with conservative assumptions for comparison with the offsite dose criteria at the EAB and LPZ,
- (3) The integrated risk from all events considered in the probabilistic risk assessment be evaluated against integrated risk targets based on the Commission's Quantitative Health Objectives (i.e., the annual average individual risk of early and latent fatalities), reported as mean risk without uncertainty, and
- (4) The cumulative frequency of exceeding 100 mrem per plant-year at the EAB be less than 1.0/plant-year, reported as mean risk without uncertainty.

For an application under proposed 10 CFR Part 57, the entry criteria would need to be met which is that using MHA or MCA approach that the DBA cannot exceed 1 rem TEDE anywhere in the unrestricted area for the duration of the accident.

5. Leveraging DOE Authorization Information in NRC License Reviews

The extent to which DOE consequence analysis information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

The information below describes how NRC staff will assess key factors for NRC leveraging of DOE authorization information related to consequence analysis and provide examples of scenarios that might lead to maximum, moderate, or limited leveraging.

Table F.1: Factors for NRC Leveraging of DOE Information

Factor	Description						
Similarity of DOE and NRC requirements and guidance	DOE-STD-3009-2014 discusses consequence calculations in section 3.2.4, which refers to NUREG-1140 as the basis for most cases only evaluating the airborne pathway, since inhalation pathway will dominate the (overall) dose. This is consistent with the general NRC approach for calculating dose consequences, however, there are cases where direct shine and ground shine are evaluated for control room habitability. There are some similarities in numerical dose criteria, but applicants will need to confirm the final analysis provided for NRC review conforms to our numerical dose criteria for the public and workers. DOE-STD-3009-2014 guidance on radiological dose consequence refers to applicable NRC guidance either as an acceptable method for DOE applications or as option for an acceptable method for these topics: <table border="1"> <tr> <td>Atmospheric dispersion and determination of offsite X/Qs</td><td>RG 1.145</td></tr> <tr> <td>Meteorological data</td><td>RG 1.233 (as used to support RG 1.145 methodology)</td></tr> <tr> <td>Release and exposure duration</td><td>RG 1.145, and additional modeling considerations for exposure duration</td></tr> </table> DOE-STD-3009-2014 guidance on dose coefficients is based on international standards and differs from NRC guidance. The guidance on breathing rate differs from NRC guidance but is based on a similar basis (light activity breathing rate for adults).	Atmospheric dispersion and determination of offsite X/Qs	RG 1.145	Meteorological data	RG 1.233 (as used to support RG 1.145 methodology)	Release and exposure duration	RG 1.145, and additional modeling considerations for exposure duration
Atmospheric dispersion and determination of offsite X/Qs	RG 1.145						
Meteorological data	RG 1.233 (as used to support RG 1.145 methodology)						
Release and exposure duration	RG 1.145, and additional modeling considerations for exposure duration						
Application quality and justification of relevance of DOE authorization to NRC review	Between the DOE authorization and NRC reviews there are many of commonalities in the way that consequence analysis is performed. In review of the consequence analysis, the information that is reviewed during the audit process is not always available in the docketed application. If sufficient audit documentation is provided with a DOE authorization that is applying to meet NRC regulations, then the NRC staff can review the methods and assumptions within the audited documentation to verify that the analysis performed for the DOE authorization satisfies applicable NRC regulations.						
Safety-significance of information leveraged	Consequence analysis provides numerical dose estimates to compare to criteria that show that sufficient systems are incorporated into the reactor design to mitigate radiological releases from the reactor facility. A well-documented						

	consequence analysis gives confidence that appropriate credit of dose factors, environmental conditions, and other supporting assumptions are established by the licensee.
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NRC Staff Approach to Leveraging DOE Consequence Analysis Information

NRC staff should consider the aspects of the application that were reviewed by either DOE as a part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the application. The extent of NRC leveraging opportunities, including to inform review scope and depth, would be dependent on how the applicant demonstrates compliance with applicable NRC regulatory requirements and whether the consequence analysis was developed using the NRC guidance reflected above or was based on a proposed alternate or hybrid approach. On this basis, NRC leveraging of DOE consequence analysis information can generally be organized into three categories:

Maximum leveraging:

- Under a maximum leveraging scenario, technical information developed or evaluated as part of a DOE authorization may provide a direct basis for an NRC review when the underlying information remains valid and the DOE authorization provides a clear method for determining consequence analysis following RG 1.183 or provides a well-documented consequence analysis (e.g., based on DOE-STD-3009-2014) that result in consequence estimates that address the applicable NRC regulatory requirements. NRC staff can focus their review on verifying that the method and justification for consequence analysis development as well as description and justification of any cross-cutting underlying assumptions are clear and complete throughout the application.

Moderate leveraging:

- Under a moderate leveraging scenario, consequence analysis information from a DOE authorization may form part of the basis of an NRC application, but additional work is required to reconcile or justify departures from RG 1.183, or to enhance justification for departures from assumptions that the NRC finds do not meet requirements of the regulatory framework being used (e.g., 10 CFR Parts 50, 52, or 53, or proposed 10 CFR part 57). The documented justifications are needed to support the level of conservatism stated or have clearly quantified uncertainties for accident dose consequences. In this case, NRC staff review is needed to verify that the approach and supplemental information meet NRC requirements, and the DOE authorization information may require NRC staff confirmatory analyses, sensitivity evaluations, or gap resolution to support consequence analysis adequacy.

Limited leveraging:

- Under a minimal-leveraging scenario, the DOE authorization does not follow RG 1.183 and does not provide any detail regarding the consequence analysis. High level assumptions are stated with little or no documentation available for review within the DOE authorization, or the DOE authorization has a significantly different scope (e.g., DOE authorized reactor is for short-term experimental activities whereas an application to the NRC is for long-term power generation), limiting applicability of the consequence analysis developed for DOE authorization to the NRC application. In this case, NRC staff should treat DOE authorization information primarily as relevant background material, and the applicant should provide a substantially new NRC-aligned consequence analysis.

APPENDIX G—Emergency Preparedness

1. NRC Regulatory Requirements

The NRC emergency preparedness (EP) regulatory framework is intended to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The applicable requirements depend on the licensing framework, reactor type, licensing stage, and EP approach selected by the applicant.

Pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50, 52, and 53 Frameworks:

- 10 CFR 50.47, “Emergency plans,” together with Appendix E to 10 CFR Part 50, establishes the traditional NRC EP framework. For power reactors, the planning standards in § 50.47(b) and the requirements of Appendix E provide the principal basis for emergency-plan content and the NRC reasonable-assurance finding.
- Under the § 50.47 framework, § 50.33(g)(1) generally provides for a plume exposure pathway EPZ of about 10 miles and an ingestion pathway EPZ of about 50 miles for nuclear power reactors, while also providing for case-by-case EPZ sizing for gas-cooled reactors and reactors with authorized power levels below 250 MWt.
- 10 CFR 50.160, “Emergency preparedness for small modular reactors, non-light-water reactors, and non-power production or utilization facilities,” establishes a performance-based EP framework currently available to eligible small modular reactors (SMRs), non-light-water reactors (non-LWRs), and non-power production or utilization facilities. Applicants using this framework may apply the scalable plume exposure pathway emergency planning zone (EPZ) provisions in § 50.33(g)(2).
- For the § 50.160 framework, the plume exposure pathway EPZ is established using § 50.33(g)(2). The analysis considers the area within which public dose is projected to exceed 1 rem TEDE over 96 hours, considering accident likelihood, source term, accident timing, and meteorology, and whether predetermined, prompt protective measures are necessary. Both criteria inform the EPZ determination.
- 10 CFR Part 52 incorporates the applicable Part 50 EP requirements for reactor licensing. For early site permits (ESPs), an applicant may elect to submit no emergency plan, major features of emergency plans, or complete and integrated emergency plans, as permitted by Part 52. Combined license (COL) applications include the emergency-planning information required for the selected framework and licensing finding.
- 10 CFR Part 53, “Risk-Informed Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants,” provides corresponding EP requirements for commercial nuclear plants and makes the NRC EP frameworks available, as applicable, through the Part 53 licensing structure.

Under the Part 50 construction-permit requirements, § 50.34(a)(10) and applicable provisions of Appendix E address preliminary plans for coping with emergencies in the preliminary safety analysis report (PSAR), as applicable to the proposed facility and selected EP framework.

Proposed 10 CFR Part 57, “Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles” (91 FR 23628, Docket ID NRC-2025-0379), would include the following emergency preparedness provisions:

- Proposed § 57.60(a)(8)(iv) would require an applicant to provide an emergency plan for responding to events that could lead to an accidental release offsite or loss of control of radioactive material, and associated hazards directly incident thereto. Applicants would coordinate response needs with local emergency planning and offsite response organizations, provide expected offsite responders an opportunity to comment on the emergency plan before submittal, and include received input in the application.

Draft NUREG-2271, “Guidelines for Preparing and Reviewing Applications Under 10 CFR Part 57,” (ML26111A077) Section 11.3, provides the proposed Part 57 EP implementation framework to be considered. The draft guidance describes a graded EP approach commensurate with the potential radiological risks and hazards of the facility. Rather than relying on a conventional fixed EPZ, the guidance describes a “planning distance” for arrangements to monitor and assess offsite radiological conditions and identify areas where protective or other response actions may be warranted following a significant release. The draft guidance expressly distinguishes the planning distance from an EPZ.

The proposed Part 57 classification approach also differs from the traditional power-reactor structure. Draft NUREG-2271 describes Notification of Unusual Event and Alert classifications, with consideration of a Site Area Emergency where appropriate. A General Emergency classification is not used as the mechanism for initiating predetermined offsite protective actions; instead, monitoring is used to determine whether offsite protective actions are warranted as conditions develop.

2. NRC Guidance and Related References

NRC guidance depends on the EP framework selected and the type of application. Relevant guidance and related references include the following:

- NUREG-0800, Section 13.3, “Emergency Planning,” (ML070810350) provides staff review guidance for evaluating EP information for nuclear power plant applications.
- RG 1.242, “Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities,” (ML23226A036) provides an acceptable method for demonstrating compliance with § 50.160 and for developing the scalable EPZ basis under § 50.33(g)(2).
- For the § 50.47/Appendix E framework, RG 1.101, “Emergency Response Planning and Preparedness for Nuclear Power Reactors,” (ML25065A240), RG 1.219, “Guidance on Making Changes to Emergency Plans for Nuclear Power Reactors,” (ML16061A104), and NUREG-0654/FEMA-REP-1 “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” (ML19347D139) provide established emergency-planning guidance, including methods for addressing planning standards, emergency-plan maintenance, and NRC/FEMA preparedness interfaces.
- For non-power production or utilization facilities licensed under Part 50, relevant guidance may include RG 2.6, “Emergency Planning for Research and Test Reactors,” (ML17263A472), NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” (ML12251A353) and NUREG-0849, “Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors,” (ML062190191) as applicable to the facility and licensing stage.
- RG 1.206, “Application for Nuclear Power Plants,” (ML18131A181) provides application-preparation and content guidance for reactor license applications, including EP information. DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor

Applications – Roadmap,” (ML23277A139) provides a roadmap for review of risk-informed, technology-inclusive advanced reactor applications.

- For proposed Part 57, Draft NUREG-2271, Section 11.3, provides application and staff-review guidance. It identifies RG 3.67, “Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities,” (ML103360487), RG 2.6, and NUREG-0654/FEMA-REP-1 as potential sources for implementing planning elements proposed in 10 CFR Part 57, as appropriate to the facility under consideration.

3. Summary of DOE Requirements, Orders, and Standards

DOE Office of Nuclear Energy emergency-management requirements are established through 10 CFR Part 830 and NE O 151.1, “Comprehensive Emergency Management System.” NE O 151.1 establishes an all-hazards emergency-management framework for covered DOE-NE facilities and activities. The framework uses an All-Hazards Survey and hazardous-material screening to determine whether an Emergency Planning Hazards Assessment (EPHA) is necessary. Where required, the EPHA provides the technical basis for emergency classification, protective actions, emergency planning zone (EPZ) determination, and associated response capabilities.

NE O 151.1 uses the EPHA to define an EPZ that encompasses the area requiring predetermined protective-action planning. The DOE EPHA includes credible natural, technological, and human-caused hazards; severe/beyond-design-basis scenarios; and consideration of malevolent acts. DOE consequence analyses use total effective dose (TED) as an applicable dose measure.

The DOE emergency-management framework addresses emergency classification and response capabilities, protective actions, emergency organizations, communications, emergency facilities and equipment, training, exercises, medical response, consequence assessment, and coordination with offsite organizations. DOE uses a three-level emergency-classification structure consisting of Alert, Site Area Emergency, and General Emergency.

4. Staff Considerations on NRC and DOE Requirements and Guidance

This section contains high-level NRC staff observations on NRC and DOE requirements and guidance and how DOE requirements and standards may be used for various NRC licensing frameworks with respect to emergency preparedness.

Leveraging Across NRC Licensing Frameworks

The NRC licensing Part 50, 52 and 53 frameworks contain similar requirements and acceptable approaches. However, the degree of leveraging may differ depending on the NRC EP framework selected and the licensing stage.

Proposed 10 CFR Part 57 would provide distinct differences. Proposed 10 CFR Part 57 would adopt a graded emergency-preparedness approach tailored to the facility’s radiological hazards, as described in Draft NUREG-2271. Instead of a conventional fixed emergency planning zone (EPZ), the guidance introduces a “planning distance,” used solely to inform offsite monitoring, assessment, and potential response actions—and explicitly not as an EPZ. Proposed Part 57’s classification framework would depart from traditional power reactor structures by using Notification of Unusual Event and Alert levels, with Site Area Emergency considered where appropriate, and relying on real-time monitoring rather than predetermined General Emergency protective actions. Because the regulations in proposed 10 CFR Part 57 would not use a fixed EPZ, the most meaningful leveraging from DOE is not the DOE EPZ itself but rather DOE EPHA and consequence information, which can support hazard assessments, planning-distance determinations, protective-action bases, classification/EAL development, site characterization, offsite coordination, and emergency-response

capability evaluations. DOE EALs likewise should not be directly imported; instead, applicants should map underlying conditions and thresholds into the proposed Part 57 structure to ensure appropriate notification, protective-action, and coordination functions.

Considerations on NRC and DOE Requirements and Guidance

Both the NRC and DOE frameworks for emergency preparedness use facility accident or hazard information to establish the technical basis for emergency planning; establish emergency classification and response capabilities; and address protective actions, emergency organizations, communications, emergency facilities and equipment, training, exercises, medical response, consequence assessment, and coordination with offsite organizations.

Several similarities may provide opportunities for leveraging DOE-developed information. For example, the DOE EPHA and the NRC § 50.160 EPZ analysis both use consequence-informed planning concepts and a 1-rem early-phase protective-action threshold as an important planning consideration. DOE-developed hazard, source-term, consequence, meteorological, protective-action, classification, and emergency-response information may therefore provide useful technical inputs to an NRC EP review where the information remains applicable to the NRC licensing basis.

The NRC notes that important differences remain, including that the DOE consequence analyses use total effective dose (TED), whereas § 50.33(g)(2) uses total effective dose equivalent (TEDE) and includes NRC-specific considerations associated with event selection, source term, accident timing, meteorology, and the need for predetermined prompt protective measures. These differences do not necessarily preclude leveraging DOE-developed consequence analyses or their supporting technical information; however, their applicability and any necessary NRC-specific evaluation or supplementation must be addressed before the information can support an NRC licensing finding. The DOE framework is also an all-hazards framework encompassing radiological, chemical, biological, toxin, and other hazardous materials, whereas NRC reactor EP requirements focus principally on radiological emergencies and other hazards to the extent they affect implementation of the radiological emergency plan.

Differences also exist in emergency-classification structures. DOE uses Alert, Site Area Emergency, and General Emergency classifications. The traditional NRC framework commonly uses Notification of Unusual Event, Alert, Site Area Emergency, and General Emergency, while § 50.160 allows a graded classification structure based on facility characteristics. These differences do not preclude leveraging DOE information but may require mapping, confirmation, or supplemental analysis before the information can support an NRC licensing finding.

The degree to which DOE information may be leveraged should consider whether the underlying DOE requirement, criterion, method, or authorization information addresses the substance of the applicable NRC requirement and whether the information remains technically applicable to the reactor design, source term, site, configuration, and licensing stage proposed for NRC licensing. For DOE leveraging, NRC staff should understand what DOE reviewed, the criteria used for that review, the design/site/configuration covered by the authorization information, and how the applicant proposes to use that information to satisfy the applicable NRC requirement. Pre-application engagement can be used to identify DOE-developed or DOE-reviewed EP information and supporting documentation that may be leveraged for NRC licensing, and any NRC-specific supplementation expected in the application.

Use of DOE standards for NRC licensing

DOE emergency-management requirements, standards, criteria, and authorization information may be appropriate to reference in NRC licensing applications where they provide technically adequate

and applicable information that addresses the substance of an NRC emergency preparedness requirement or review criterion.

DOE information that may provide opportunities for leveraging, subject to its technical adequacy and applicability to the NRC licensing basis, includes the EPHA and supporting hazard identification; source-term, atmospheric-dispersion, and consequence analyses; site, meteorological, and receptor information as applicable to the site and configuration proposed for NRC licensing; protective-action analyses; technical bases for emergency classification and EALs; emergency-response organization and staffing information; communications, facilities, and equipment information; training and exercise records; offsite-response coordination information; and after-action and corrective-action records.

Technically adequate DOE information in these areas, as justified by an applicant, may provide input to, or part of the technical basis for, an NRC licensing finding without requiring the applicant or NRC staff to recreate information.

Where DOE-developed source-term, consequence-analysis, or risk information is proposed by an applicant for use as an input to an NRC emergency preparedness review, the EP review should continue to rely, as appropriate, on the applicable NRC technical review and finding for those underlying inputs rather than independently duplicating reviews assigned to other technical disciplines.

In particular, elements of NE O 151.1 may provide useful technical support for reviews conducted under the performance-based § 50.160 framework because both approaches use consequence-informed planning concepts, scalable emergency planning, emergency classification, protective actions, response capabilities, and a 1-rem early-phase protective-action threshold. Under the proposed Part 57 framework, leveraging would be expected to be maximized when the DOE hazard and consequence information substantially supports the hazard assessment, planning-distance basis, protective actions, classification, and offsite-coordination findings, with only limited NRC-specific supplementation.

Under the § 50.47(b)/Appendix E framework, DOE information may continue to provide substantial technical and implementation support, but direct leveraging may be more limited where NRC requirements establish specific planning standards, classification or notification expectations, NRC/FEMA interfaces, or other regulatory findings that are not addressed by the DOE authorization basis. DOE analyses, site data, demonstrated response capabilities, training and exercise records, and offsite-coordination information may reduce duplicative fact-finding, while NRC-specific requirements would continue to be addressed in the application and evaluated by NRC staff.

Under proposed 10 CFR Part 57, the principal leveraging opportunity would not necessarily be the DOE EPZ determination itself because the proposed 10 CFR Part 57 guidance in draft NUREG-2271 does not rely on a conventional fixed EPZ. Instead, DOE EPHA and consequence information may support the proposed 10 CFR Part 57 hazard assessment, planning-distance basis, protective-action basis, classification/EAL development, site characterization, offsite coordination, emergency-response capabilities, and demonstrated performance. The more appropriate transition is from the DOE EPHA, consequence analysis, and EPZ technical basis to the proposed 10 CFR Part 57 hazard assessment, planning-distance determination, and protective-action basis, rather than treating the DOE EPZ as the NRC proposed 10 CFR Part 57 defined EPZ.

Similarly, DOE EALs should not automatically be carried into a proposed 10 CFR Part 57 emergency plan. The underlying initiating conditions, indications, consequence thresholds, and response logic may be reusable, but the applicant should map them to the proposed 10 CFR Part 57 classification and response structure and demonstrate that the required notification, protective-action, and offsite-coordination functions are addressed.

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE emergency preparedness information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

Consistent with the discussion in the main body of this white paper, this table outlines three of the key factors in determining the extent to which DOE authorization information related to emergency preparedness may be leveraged.

Table G.1: Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	Leveraging is expected to be greatest where DOE and NRC use similar technical inputs and planning concepts. Examples include hazard/accident identification, source-term and consequence analyses, site and meteorological information, protective-action considerations, emergency classification bases, communications, response capabilities, and exercises. Alignment is particularly strong between many elements of NE O 151.1 and § 50.160. For proposed 10 CFR Part 57, the underlying DOE hazard and consequence information may be highly reusable even though proposed 10 CFR Part 57 would include a planning distance rather than a conventional fixed EPZ.
Application quality and justification of relevance of DOE authorization to NRC review	Maximum leveraging depends on a clear applicant justification on how information from the DOE authorization is relevant and sufficient to meet NRC requirements. This may include the DOE requirement or criterion applied, what DOE reviewed or approved, the design/site/configuration covered, and the corresponding NRC requirement. Analyses should be traceable, technically adequate, current, and supported by sufficient documentation to permit an NRC finding without reconstructing the DOE review, as appropriate.
Safety-significance of information leveraged	EP information supports public-protection functions and may be safety significant. Greater NRC review scope and depth may be necessary where information establishes the radiological planning basis, EPZ or planning-distance basis, protective-action strategy, classification/EAL scheme, staffing, or key response functions. More routine factual information, demonstrated capabilities, site data, and implementation records may offer greater opportunity for focused confirmation rather than duplicative review.

NRC Staff Approach to Leveraging DOE Emergency Preparedness Information

NRC staff should determine the appropriate degree of leveraging based on the technical adequacy, applicability, traceability, and completeness of the DOE information; the relationship between the

DOE review basis and the corresponding NRC requirement or review criterion; and the nature and significance of any remaining NRC-specific gap.

This information provides the basis for NRC staff to determine the appropriate degree of leveraging and whether focused confirmation, supplemental evaluation, or a standard NRC review is warranted.

On this basis, NRC leveraging of DOE emergency preparedness authorization information can generally be organized into three categories:

Maximum leveraging

- Maximum leveraging may be appropriate where DOE information was developed and reviewed for the same or substantially unchanged reactor design, source term, site, and configuration proposed for NRC licensing; the applicable DOE criteria and review basis are documented and traceable; and the information substantially addresses the applicable NRC requirement or review criterion.
- In these circumstances, NRC staff may directly leverage technically adequate DOE information and focus the NRC review on confirming continued applicability and addressing any specific supplemental information necessary to support the NRC finding.
- *Example:* An example of maximum leveraging includes a DOE EPHA, supporting source-term and consequence analyses, meteorological information, and DOE EPZ analysis may substantially support the technical basis for an NRC § 50.33(g)(2) EPZ determination. However, the applicant would still need to demonstrate that the DOE information adequately addresses applicable NRC-specific elements, including the dose metric, event-selection methodology, source term, release timing, meteorology, and the need for predetermined prompt protective measures.

Moderate leveraging

- Moderate leveraging may be appropriate where DOE information provides a technically sound and relevant foundation but material differences remain in areas such as dose methodology, event selection, source term, meteorology, classification structure, site configuration, offsite interfaces, staffing, or NRC-specific requirements or guidance.
- NRC staff may leverage the applicable DOE technical information while performing targeted review, mapping, confirmation, or supplemental evaluation necessary to address the identified differences and support the NRC finding.
- *Example:* For example, a DOE EPZ analysis may provide relevant supporting technical information but require an NRC-specific § 50.33(g)(2) determination or additional evaluation to support the proposed 10 CFR Part 57 planning-distance approach. Similarly, DOE emergency-classification technical bases may be reusable but may need information to clearly map to the selected NRC framework.

Limited leveraging

- Limited leveraging may be appropriate where DOE information is high-level or primarily site-wide; is not traceable to the reactor design or configuration proposed for NRC licensing; relies on materially different hazards, source terms, accident scenarios, site conditions, or analytical methods; depends on DOE site-wide response resources that will not remain

available under the NRC licensing basis; or otherwise, does not provide sufficient information to support the applicable NRC finding.

- In these circumstances, DOE information may still be useful for limited factual, historical, site-characterization, or implementation purposes, but NRC staff would perform substantially the normal review necessary to determine compliance with applicable NRC requirements.
- *Example:* Limited leveraging is appropriate where DOE response capability depends heavily on site-wide DOE resources that will not be available under the NRC licensing basis.

6. Additional Review Considerations

NRC staff should not equate prior DOE approval with NRC acceptance. Prior DOE authorization should inform the scope and depth of NRC review where the applicant demonstrates that DOE information is technically adequate, applicable, and sufficient to address all or part of an NRC requirement. Site-specific applicant commitments, offsite interfaces, and implementation demonstrations remain subject to NRC review as required by the selected licensing framework.

Where DOE information has been successfully demonstrated or exercised, exercise after-action reports, corrective-action records, equipment and communications test results, training records, and evidence of offsite coordination may provide additional confidence in implementation capability and may support a more focused NRC review.

APPENDIX H—Security

1. NRC Regulatory Requirements

Pursuant to Title 10 of the *Code of Federal Regulations* 10 CFR 50.34(c)–(d), 10 CFR 52.79(a)(35)–(36), 10 CFR 53.860, and proposed 10 CFR 57.60(a)(8)(v), an applicant must have a physical protection program that provides reasonable assurance of adequate protection against the Design Basis Threat (DBT) of radiological sabotage.

- 10 CFR 50.34(c), “Physical security plan,” and (d), “Safeguards contingency plan,” and the corresponding Part 52 requirements, 10 CFR 52.79(a)(35)–(36), establish the security plan content that must be included as part of a complete application for an operating license under Part 50 and COL under Part 52.
- 10 CFR 73.55, “Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage,” specifies the requirements for physical protection programs that apply to nuclear power plant applicants and licensees and is applicable to Parts 50, 52 and 53.
- 10 CFR Part 53, “Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants,” requires applicants to submit a physical security plan as part of the required licensing basis information. 10 CFR 53.860, “Security programs,” requires each nuclear power reactor applicant or licensee to establish, maintain, and implement a physical protection program that provides reasonable assurance that licensed activities involving special nuclear material are not inimical to the common defense and security. As with the Part 50 and Part 52 frameworks, this regulatory provision establishes the security plan content that must be included as part of a complete license application under Part 53.
- 10 CFR 73.100, “Technology-Inclusive Requirements for Physical Protection of Licensed Activities at Commercial Nuclear Plants Against Radiological Sabotage,” provides a performance-based regulatory framework for the design, implementation, and maintenance of a physical protection program and security organization for certain commercial nuclear plants licensed under 10 CFR Part 53. Note that applicants under Parts 50, 52, and 53 have the option to choose between 10 CFR 73.55 or 10 CFR 73.100 as their security framework.
- 10 CFR Part 37, “Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material.”
- 10 CFR 73.67, “Licensee fixed site and in-transit requirements for the physical protection of special nuclear material of moderate and low strategic significance.”

Proposed 10 CFR Part 57, “Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles,” would require applicants to include a physical security plan as part of the licensing basis. Proposed §57.60(a)(8)(v) identifies the applicable dose-based acceptance criterion used to assess system performance, including security-initiated scenarios, and therefore establishes the level of protection that applicants must demonstrate. Proposed 10 CFR Part 57 would also include 10 CFR 57.325 in Subpart J, “Physical Security Requirements.”

Collectively, the requirements in proposed 10 CFR 57.60(a)(8)(v) for applicants under proposed 10 CFR Part 57 and 10 CFR 53.860 for applicants under 10 CFR Part 53 parallel the long-standing requirements in 10 CFR Parts 50 and 52 by ensuring that each applicant submits a physical security plan demonstrating how the facility will meet NRC security requirements. These provisions maintain the NRC's consistent regulatory expectation that physical protection programs are fully defined within all commercial reactor licensing applications, including microreactors and technology-inclusive licensing pathways, thereby ensuring security measures remain commensurate with potential risk posed by such facilities.

2. NRC Guidance and Related References

The following regulatory guides (RG's) describe an approach that is acceptable to the NRC staff to meet the regulatory requirements for physical security, addressing various requirements in 10 CFR Part 73, "Physical Protection of Plants and Materials."

- RG 1.260, "License Applications for Commercial Nuclear Power Plants," (Part 53). (Not yet publicly available).
- RG 5.54, "Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants." (ML003739247).
- RG 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance." (ML100341301).
- RG 5.66, "Access Authorization Program for Nuclear Power Plants." (ML21145A433).
- RG 5.69, "Guidance for the Application of Radiological Sabotage Design-Basis Threat in the Design, Development, and Implementation of a Physical Security Program that Meets 10 CFR 73.55 Requirements." (SGI) (Not publicly available)
- RG 5.75, "Training and Qualification of Security Personnel at Nuclear Power Reactor Facilities." (ML17111A682).
- RG 5.76, "Physical Protection Programs at Nuclear Power Reactors." (Not publicly available).
- RG 5.77, "Insider Mitigation Program." (ML16342B024).
- RG 5.79, "Protection of Safeguards Information." (ML103270656).
- RG 5.81, "Target Set Identification and Development for Nuclear Power Reactors." Revision 2. (OUO-SRI). (not publicly available).
- RG 5.97, "Guidance for Technology-Inclusive Requirements for Physical Protection of Licensed Activities at Commercial Nuclear Plants." (ML25232A009).
- NUREG 0800, Chapter 13, "Physical Security - Combined License and Operating Reactors." (ML052340514).

3. Summary of DOE Requirements, Orders, and Standards

DOE Order NE O 470.1 establishes a comprehensive safeguards and security framework for facilities and activities under the Department of Energy's (DOE) Office of Nuclear Energy (NE). The order requires a graded, performance based physical protection program that accounts for site-

specific risks, consequence thresholds, and threat considerations which are documented in a Local Threat Assessment (LTA).

Each nuclear facility subject to the order must have a security plan that addresses site-specific security requirements and procedures. Core program elements include maintaining an approved security plan and organization, establishing layered physical barriers, implementing access control and search programs, deploying intrusion detection and assessment systems, and establishing and maintaining a response strategy. The order further requires response capabilities commensurate with potential radiological consequences, insider mitigation measures, cybersecurity integration, and established processes for testing, evaluation, maintenance, and compensatory actions to ensure sustained system effectiveness.

NE O 470.1, “Office of Nuclear Energy Safeguards and Security” discusses that:

- This Order establishes a comprehensive framework of programs and requirements within DOE NE to ensure the protection, control, and accountability of DOE NE assets. It details specific security measures tailored to the needs of DOE NE, emphasizing a performance-based strategy that grants significant decision-making authority to NE’s Safety Basis Approval Authority (SBAA) or Field Element Manager (FEM), as applicable. The approach prioritizes achieving desired outcomes and results over merely fulfilling requirements, offering flexibility in how goals are accomplished and emphasizing continuous improvement, proactive risk management, demonstrated performance, and data-driven decision-making to ensure optimal results.

4. Staff Considerations on NRC and DOE Requirements and Guidance

This section contains high-level NRC staff observations on NRC and DOE requirements and guidance and how DOE requirements and standards may be used for various NRC licensing frameworks with respect to physical security.

Leveraging Across NRC Licensing Frameworks

Across these licensing pathways, 10 CFR 73.55, “Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage,” 10 CFR 73.100, “Technology-inclusive requirements for physical protection of licensed activities at commercial nuclear plants against radiological sabotage,” and proposed 10 CFR Part 57, Subpart J, “Physical Security Requirements,” continue to specify the mandatory elements of physical protection programs applicable to nuclear power plant applicants and licensees. For applicants under Part 50, 52, and 53, and proposed 10 CFR Part 57, these requirements form the foundation for the design, documentation, and implementation of site security programs.

NRC licensing frameworks for physical security contain substantially similar requirements and regulatory approaches relevant to the use of DOE-generated technical information. Accordingly, although the manner and extent to which DOE information is applied may vary among licensing frameworks, the differences are not expected to fundamentally alter their potential use in supporting NRC licensing reviews.

Considerations on NRC and DOE Requirements and Guidance

Potential leveraging of DOE physical security requirements centers on the conceptual alignment between DOE NE O 470.1 safeguards and security framework and the NRC’s regulatory

requirements in 10 CFR 73.100.⁸ The NRC staff notes that there are aspects of DOE's performance-based approach, such as documented security organizations, risk-informed threat assessments, and defense-in-depth protective strategies, align closely with NRC's physical security objectives for commercial reactors. These DOE requirements demonstrate alignment with NRC physical security objectives, providing a technically comparable basis that supports NRC licensing reviews and can be leveraged where the demonstrated DOE capabilities correspond to the performance objectives defined in the NRC framework.

While certain DOE physical security objectives can be leveraged, other areas will require further analyses or NRC review. For example, radiological consequence thresholds in NE O 470.1 differ numerically from NRC's reference values in 10 CFR 53.210, necessitating reconciliation. DOE's firearms authorization and background-check provisions do not fully match the requirements of 10 CFR 73.17, meaning that NRC applicants must still address these NRC-specific elements independently. Additionally, NE O 470.1 authorizes the use of an LTA in lieu of a standardized Design Basis Threat (DBT). Despite these differences, many of NE O 470.1's security program components, such as search programs, record keeping, security reviews, and maintenance and testing requirements, closely mirror the structure and intent of 10 CFR 73.100's programmatic requirements.

Additionally, although both the DOE and NRC maintain DBT frameworks, the two agencies define and apply these threat bases differently. DOE's physical protection programs rely on LTA's tailored to the specific site, mission, and regional conditions, whereas the NRC employs a standardized, nationally applied DBT that does not incorporate such localized assumptions. Because DOE's protection strategies and performance demonstrations may be based on site-specific threat environments that are not considered under the NRC's regulatory framework, this difference presents a significant consideration when evaluating the extent to which DOE authorization information can be leveraged in a commercial application. As a result, reliance on DOE security analyses must consider how the DOE's localized threat assumptions align with the security posture required to meet NRC's DBT and associated performance objectives for licensed facilities.

Use of DOE Standards for NRC Licensing

The NRC staff notes that DOE Order DOE NE O 470.1 contains physical security elements generally similar to the NRC, in particular to the security requirements of 10 CFR 73.100; however, the NRC's regulations for physical protection may not fully be met by this DOE Order, or other guidance documents. For example, as discussed, DOE's firearms authorization and background-check provisions do not fully match the requirements in §73.17. Additionally, although DOE and NRC maintain DBT frameworks, the two agencies define and apply these threat bases differently. In instances where there is not justifiable overlap or complete applicability of the information from an authorization, an applicant will need to provide the supplemental information necessary to meet NRC requirements.

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE physical security information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting

⁸ The NRC's focus on 10 CFR 73.100 reflects its ability to provide maximum regulatory leverage, given its risk-informed, performance-based, and technology-inclusive framework for advanced reactor physical security requirements.

the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

Consistent with the discussion in the main body of this white paper, this table outlines three of the key factors in determining the extent to which DOE authorization information related to physical security may be leveraged.

Table H.1: Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	The objective of both NRC and DOE physical security requirements is to ensure robust protection of nuclear facilities by minimizing the likelihood and consequences of unauthorized access or radiological sabotage through a defense-in-depth, performance-based security posture. Both agencies employ risk-informed approaches that rely on threat assessments, engineered and administrative protective measures, performance-based analyses, and facility-specific evaluations to achieve effective detection, assessment, delay, and response capabilities consistent with their respective regulatory frameworks. The two organizations share substantial conceptual alignment across core security functions, including intrusion detection, surveillance and assessment, access control, physical barriers, security response, insider mitigation, cybersecurity integration, and performance evaluation programs, which creates a strong basis for leveraging DOE security information in NRC licensing reviews. Given this similarity in underlying security philosophy and technical expectations, there is potential for NRC applicants to leverage DOE security analyses, testing, and performance evidence, except in areas where NRC requirements diverge meaningfully, such as those involving DBT-specific assumptions, target-set methodology, or NRC-unique statutory provisions that require supplemental justification.
Application quality and justification of relevance of DOE authorization to NRC review	Before incorporating DOE authorization information into an NRC physical security review, NRC staff should evaluate the information's quality, technical adequacy, underlying security basis, changes to facility design, and the application of DOE's graded security approach. This evaluation should confirm that the information is complete and traceable, that the supporting security analyses and performance evaluations are technically sound, that applicable threat scenarios and target sets are accurately characterized, and that the referenced protective measures and methodologies have been applied appropriately to help inform the degree to which DOE information can be leveraged.
Safety-significance of information leveraged	The NRC staff should pay close attention to information that supports essential physical-security functions, including

	detection, assessment, delay, response, protection of target sets, and measures that contribute to maintaining the reliability and availability of security-related structures, systems, and components. When the information pertains to security functions or includes security-sensitive content, the NRC staff should also ensure that all applicable physical-security requirements, safeguards controls, and protective-strategy elements are properly implemented and consistent with NRC regulations.
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NRC Staff Approach to Leveraging DOE Physical Security Information

In making this determination, NRC staff should consider the degree of alignment between DOE and NRC security objectives, the applicability of DOE analyses to the NRC DBT, and any differences in protective strategies, target-set methodologies, or security-related assumptions. On this basis, NRC leveraging of DOE physical security information can generally be organized into three categories: information that can be leveraged directly, information that may be leveraged with supplemental evaluation, and information that can only serve as supporting context but still requires an NRC-specific review.

Maximum leveraging

- Technical information and analyses developed or evaluated as part of a DOE authorization may provide a direct basis for an NRC review when the underlying information remains valid and the relevant facility conditions are sufficiently comparable to those required by the NRC. Under maximum leveraging, NRC staff can focus their review on verifying that the security related information and its relation to the design bases is clear and complete. Given the cross-cutting nature of the design criteria in supporting the licensing basis, the NRC staff will review various aspects of the design criteria and how they are satisfied but can achieve significant review efficiencies, depending on the scope and depth of the DOE review and documentation. Significant overlap is possible across security organizations, risk-informed threat assessments, and defense-in-depth protective strategies.
- *Example:* An example of a maximum leveraging scenario is expected when applicants have the end in mind for foundational elements such as maintaining a comprehensive physical protection program, conducting site-specific threat and vulnerability assessments, and implementing robust access authorization and continuous improvement processes. Programmatically, maximum alignment is expected in the development and approval of security plans, establishing and staffing a qualified security organization, enforcing entry and search procedures, maintaining rigorous training and qualification programs for protective forces, conducting independent security reviews, carrying out performance evaluations, and ensuring thorough maintenance, documentation, and recordkeeping practices. These shared structures reflect similar operational philosophies and create substantial opportunities to leverage DOE processes directly in meeting NRC regulatory requirements.

Moderate leveraging:

- Certain DOE information may be technically applicable but require additional NRC evaluation before it can support an NRC finding. In this case, additional work is required to reconcile differences and ensure that the resulting security program fully satisfies NRC requirements. Detailed NRC evaluation is needed to establish traceability between how an applicant meets DOE requirements and the justification of applicability to NRC security requirements.

- *Example:* Examples of a moderate leveraging scenario exist in the areas of performance-based requirements such as intrusion detection, assessment, communication, response capabilities, vehicle bomb protection, and access control. In these cases, supplemental information would likely be required to address NRC requirements. Moderately aligned program elements also include dose-related design criteria, cybersecurity protocols, insider threat mitigation, and coordination of safety-security interfaces, all of which share conceptual foundations but differ in numerical thresholds, analytical approaches, or implementation specificity.

Limited leveraging

- Under limited leveraging, NRC staff treats information from a DOE authorization primarily as relevant background material, and the applicant must provide new or significant supplemented information to meet NRC requirements. In this case, DOE information could inform the NRC's review but would not constitute the primary technical basis for the NRC's safety determination.
- *Example:* An example of limited leveraging is associated with the NRC's design basis threat and the DOE's local threat assessment. The NRC staff anticipate limited leveraging in this area as these approaches differ in adversary characteristics, target set identification, and performance evaluation. In this case, an applicant would need to provide a significant amount of additional or new information to ensure that all NRC physical security requirements are met.

6. Additional Review Considerations

The NRC can leverage a prior DOE authorization as a validated security basis to streamline the physical-security review for a reactor design seeking commercial licensing, provided the DOE evaluation contains sound technical analyses, demonstrations, and operational evidence relevant to NRC requirements. DOE authorization would not function as an NRC approval; rather, it would serve as a source of objective evidence that the NRC may leverage when it demonstrates alignment with NRC performance objectives under 10 CFR 73.100. This includes DOE-developed security analyses, threat and consequence evaluations, and testing results that remain applicable to the reactor's proposed commercial configuration.

The NRC staff should perform a review to the level of scope and depth necessary to verify the findings and basis from the prior DOE authorization; however, the review should focus on differences, or "deltas", such as variations in design, siting, operational assumptions, threat basis, or security-related systems. Where DOE evidence directly supports compliance with NRC objectives, the NRC could leverage that information rather than an applicant repeat the underlying analysis or testing, thereby reducing regulatory duplication while maintaining reasonable assurance of adequate protection against the DBT of radiological sabotage. To support this approach, an applicant may provide a DOE-to-NRC physical-security crosswalk identifying which aspects of the DOE authorization correspond to NRC requirements and which areas require additional evaluation.

APPENDIX I—Fire Protection

1. NRC Regulatory Requirements

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.48, 10 CFR 52.47, 10 CFR 52.79, 10 CFR 53.875, and proposed 10 CFR 57.60(a)(8)(ix), an applicant must have a fire protection program.

10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” provides regulations for licensing production and utilization facilities, including these excerpts associated with fire protection:

- 10 CFR Part 50.48, “Fire protection,” established the requirements to maintain a fire protection plan that satisfies Criterion 3, Fire protection,” of Appendix A to 10 CFR Part 50.
- National Fire Protection Association (NFPA) 805, “Fire protection for Light Water Reactor Electric Generating Plants,” as provided in §50.48.
- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” contains the General Design Criteria (GDC) that establish the minimum requirements for light-water reactor (LWR) designs. Criterion 3 establishes the requirement for structures, systems, and components (SSCs) important to safety be designed and located to minimize the probability and effects of fires and explosions.
- 10 CFR Part 50, Appendix R, “Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979”

10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants” fire protection requirements for design certification and COL applicants reference a description and analysis of the fire protection features necessary to comply with 10 CFR Part 50, Appendix A, Criterion 3, and §50.48. This is also reflected in 10 CFR 52.157 and 10 CFR 52.137 for manufacturing licenses and standard design approvals, respectively.

10 CFR Part 53, “Risk-Informed Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants” fire protection requirements include these excerpts:

- 10 CFR 53.875, “Fire protection,” must have a fire protection plan that describes the program, organization responsibilities, and outline the fire protection features necessary to limit fire damage. This part also requires a fire protection program that demonstrates compliance with the safety criteria outlined in §§53.210 and 53.220, safety functions outlined in §53.230 and defense in depth outlined in §53.250.

Proposed 10 CFR Part 57, “Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles” would include the following fire protection excerpts:

- Proposed 10 CFR 57.60(a)(8)(ix) would require the applicant to provide a fire protection program and would be similar to 10 CFR 50.48(a).
- Proposed 10 CFR 57.60(a)(8)(ix)(A)-(C) would require the applicant to describe the fire protection program for the facility, any specific features necessary to implement the program,

and an analysis to demonstrate that a fire or explosion in any area of the plant would not prevent a safety-related SSC from performing its safety function.

- Proposed 10 CFR 57.60(a)(8)(ix)(D)-(H) would establish specific requirements for the fire protection program.

2. NRC Guidance and Related References

Fire Protection Guidance

- Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routs Near Nuclear Power Plants, November 2021, [ML21260A242](#).
- Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants," October 2023, [ML23214A287](#).
- Regulatory Guide 1.191, "Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown," January 2021, [ML20287A199](#).
- Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," May 2021, [ML21048A448](#).
- DANU-ISG-2022-09, "Risk-Informed, Performance-Based Fire Protection Program," (for Operations), March 2024, [ML23277A147](#).
- Regulatory Guide 1.253, Revision 1, "Guidance for a Technology-Inclusive Content of Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," [ML25260A526](#).
- Draft NUREG-2271, "Guidelines for Preparing and Reviewing Applications Under 10 CFR Part 57," [ML26111A077](#).
- NUREG-0800, Standard Review Plan, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.5.1.1, "Fire Protection Program," Revision 0, February 2009, [ML090510170](#).
- NUREG-0800, Standard Review Plan, Section 9.5.1.2, Revision 0, "Risk-Informed, Performance-Based Fire Protection Program," December 2009, [ML092590527](#).
- Principal Design Criteria (PDC) 3, "Fire protection," for non-LWR designs. Regulatory Guide 1.232, Revision 0, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," issued in April 2018 ([ML17325A611](#)) provides guidance for developing fire protection PDC. This approach is similar to that established under 10 CFR Part 50, Appendix A, Criterion 3, which sets forth the general requirements for fire protection in LWR designs.

DANU-ISG-2022-09 provides guidance for developing an advanced reactor application for an operating license under 10 CFR Part 50 or a combined license under 10 CFR Part 52 that incorporates a proposed risk-informed, performance-based fire protection program for operations. (ML23277A105).

Although 10 CFR 50.48(c), "National Fire Protection Association Standard NFPA 805," does not apply to non-LWRs, elements and concepts of NFPA 805, as well as the guidance in Regulatory Guide 1.205, and Regulatory Guide 1.189, may be applied to non-LWR applications, provided that any exceptions or deviations are appropriately justified.

Although the scope of DANU-ISG-2022-09, is limited to risk-informed, performance-based applications, the NRC staff encourages applicants to leverage existing guidance in developing their fire protection programs. Where applicants depart from applicable guidance, they should identify the exceptions in their applications and provide the associated technical justifications. Relevant guidance and industry standards include, but are not limited to NFPA 804, “Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants”); and NFPA 806, “Performance-Based Standard for Fire Protection for Advanced Nuclear Reactor Electric Generating Plants Change Process”, as discussed in DANU-ISG-2022-09.

In addition, the guidance on administrative programs for operations and the associated NRC staff acceptance criteria in DANU-ISG-2022-09 are generally applicable to non-LWR applications that use a deterministic approach. Applicants are encouraged to discuss the applicability of this guidance to their specific design and licensing approach during preapplication engagements. For applications that are not expected to include a probabilistic risk assessment (PRA), these discussions should also address aspects of the fire protection program that may incorporate risk information considerations, such as the use of risk insights to inform the fire protection program change process and surveillance requirements.

3. Summary of DOE Requirements, Orders, and Standards

For Hazard Category 1, 2, or 3 nuclear facilities, [DOE-STD-1066-2023](#), “Fire Protection,” establishes fire protection requirements consistent with the Highly Protected Risk (HPR) criteria, which represent a stringent fire-protection standard intended to minimize the likelihood of a significant fire and its potential consequences. HPR defines a facility in which both the frequency and severity of fire-related losses (i.e., fire risk) are addressed through fire protection features. These features may include, but are not limited to noncombustible construction, fire detection and suppression systems, redundant water supplies, fire alarm systems, and specific measures and procedures for addressing special hazards.

DOE requirements for the facility fire protection program typically consist of [NE O 420.1](#), “Facility Safety,” with detailed criteria and guidance provided in DOE-STD-1066-2023.

- NE O 420.1 establishes requirements for comprehensive fire protection programs for DOE nuclear facilities and emergency response organizations. The objective being to minimize the likelihood and consequences of fire-related events.
- NE O 420.1 (Attachment 2, Chapter II, “Fire Protection”) organizes fire protection program requirements into several categories, including:
 - General requirements (codes and standards)
 - Administration
 - Design
 - Fire protection and life safety
 - Operations
 - Emergency response
 - Fire hazards analysis
 - Wildland fires

Section 4, “Fire Protection Design,” of DOE-STD-1066-2023 provides criteria and guidance for meeting the requirements of [DOE O 420.1C](#), “Facility Safety.” It covers administrative controls, operational requirements, emergency response, and wildland fire management. It also contains

detailed criteria for facility fire protection systems used in safety classes and safety-significant applications.

4. Staff Considerations on NRC and DOE Requirements and Guidance

This section contains high-level NRC staff observations on NRC and DOE requirements and guidance and how DOE requirements and standards may be used for various NRC licensing frameworks with respect to fire protection.

Leveraging Across NRC Licensing Frameworks

Parts 50, 52, 53, and proposed 10 CFR Part 57 share the fundamental requirement that applicants establish a fire protection plan and fire protection program that ensures fire protection defense-in-depth, protects safety-related SSCs, and meets the intent of 10 CFR Part 50, Appendix A, Criterion 3, including preventing, detecting, controlling, and mitigating fires so they do not impede safety functions. NRC licensing frameworks for fire protection contain substantially similar requirements and regulatory approaches relevant to the use of DOE-generated technical information. Accordingly, although the manner and extent to which DOE information is applied may vary among licensing frameworks, the differences are not expected to fundamentally alter its potential use in supporting NRC licensing reviews.

The NRC encourages preapplication engagement to identify where aspects of a DOE authorization align with the selected NRC licensing framework and to clarify where additional information may be needed to support an efficient review.

Considerations on NRC and DOE Requirements

Both DOE and NRC programs incorporate a defense-in-depth design philosophy that is applied to fire protection. DOE's [defense-in-depth](#) approach is described in DOE-STD-3009, "Preparation of Nonreactor Nuclear Facility Documented Safety Analysis" ([ML18019A922](#)). The NRC has a fire-specific [defense-in-depth](#) approach described in RG 1.189. Both frameworks are designed to ensure that fire hazards do not prevent important nuclear safety functions. A fundamental principle common to both approaches is the use of layered defense-in-depth objectives to prevent, detect, control, and mitigate fires and thereby minimize the potential for fire-induced loss of nuclear safety functions.

DOE regulations ([10 CFR 830.3](#)) specify any SSCs that are not designated as safety class, but whose preventive or mitigative function is a major contributor to defense-in-depth and/or worker safety as determined by the safety analyses, be classified as a safety significant SSC.

The NRC fire protection approach is a licensing-based regulatory framework in which fire protection requirements are established through CFR and incorporated into the nuclear power plant's fire protection licensing basis, supplemented by NRC guidance and applicable industry consensus codes and standards and heavily emphasizes maintaining the capability for safe reactor shutdown. As previously discussed, the NRC regulations require specific fire protection defense-in-depth measures, including redundant and physically separated trains, fire barriers, and dedicated on-site fire brigades capable of responding to design-basis fire events. In order to leverage fire-protection information from a DOE authorization, an applicant should consider that the manner in which a DOE authorization applies the defense-in-depth principles from Appendix A of DOE-STD-3009 is clearly aligned with, and demonstrably supports, the fire-protection-specific defense-in-depth principles outlined in Regulatory Guide 1.189.

In addition to the defense in depth approach discussed above, DOE requirements focus on nuclear safety, worker protection, and asset preservation across a wide range of reactor and non-reactor facilities. DOE guidance relies on national consensus standards for industrial fire protection with limits placed on the level of protection. The DOE uses a requirement and standards-based framework that relies on DOE directives, technical standards, the facility's approved safety basis, engineering analyses, and facility-specific criteria, supplemented by relevant industry consensus codes and standards. This distinction is of note when NRC requirements are used as benchmarks for DOE facilities, in that their applicability and equivalence to DOE requirements must be evaluated and demonstrated rather than assumed. DOE also explicitly requires the development of a fire hazard analysis or facility assessment early on in the authorization process (NE O 420.1, DOE-STD-1271, DOE-STD-1189, [DOE-STD-1166](#), "Integration of Multiple Hazard Analysis Requirements and Activities"). While the core principles of fire protection engineering are similar, the applicable requirements, scope, compliance basis, and methods of implementation may vary based on facility type, hazards, safety classification, and the facility-specific safety basis.

A key area of difference between NRC and DOE requirements exists for the area of post-fire safe shutdown analysis. NE O 420.1 Attachment 3, "Design Criteria for Safety Structures, Systems, and Components," contains a design criteria for the protection of Safety Class systems against the effects of fire such that these safety class and safety significant systems are designed with redundancy or other means to ensure safety functions are maintained. DOE-STD-1066-2023 provides high-level guidance on meeting this requirement, whereas NRC requirements and guidance provide extensive detail on how to meet the requirements to ensure post-fire safe shutdown.

Use of DOE Standards for NRC Licensing

The NRC staff notes that DOE requirements and standards, such as NE O 420.1, DOE-STD-1271, DOE-STD-1189, and DOE-STD-1166, contain fire protection program elements and hazard-analysis methodologies generally similar to the NRC; however, the NRC's regulations for fire protection may not fully be met by these DOE standards, for example, in the areas of specific defense in depth analyses, post fire safe shutdown capability and SSC classification.

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE fire protection information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

This table outlines three of the key factors in determining the extent to which DOE authorization information related to fire protection may be leveraged.

Table I.1 - Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	The overall objective of both NRC and DOE fire protection requirements aims to minimize the likelihood and consequences of fire and employ defense-in-depth concepts. Guidance from both organizations relies on industry consensus codes and standards, agency-specific requirements and guidance, engineering analyses, and facility-specific evaluations to prevent, detect, control, and mitigate fires and protect nuclear safety functions. Given the similarities between NRC and DOE requirements, there is a potential for significant leveraging of DOE information except for post-fire safe shutdown where NRC requirements are significantly more extensive.
Application quality and justification of relevance of DOE authorization to NRC review	Before leveraging DOE authorization information provided in an application in an NRC fire protection safety review, NRC staff should evaluate the applicant's justification of the information's quality, technical adequacy, underlying technical basis, changes to facility design, and application of DOE graded approach. The NRC staff notes that to maximize leveraging of DOE information, an applicant should provide a clear, well-documented, and traceable justification for the information described above. This justification should demonstrate that fire hazard analyses and engineering evaluations are technically sound, applicable fire hazards are accurately characterized, and industry codes and standards have been appropriately applied.
Safety-significance of information leveraged	To the extent that the fire protection program identifies protection of SSCs that are more significant to plant safety, additional focus may be needed to verify that the program adequately protects those more risk significant systems.

NRC Staff Approach to Leveraging DOE Fire Protection Information

NRC staff should consider the similarity of the facility and configuration, fire hazards and fire modeling, assumptions and analytical methods, credited safety functions, performance criteria, and applicable acceptance criteria. The application of DOE's graded approach should also be considered to determine if design changes require additional rigor to demonstrate safety to meet NRC requirements. On this basis, use of DOE fire protection information can generally be organized into three categories:

Maximum leveraging

- Technical information developed or evaluated as part of a DOE authorization may provide a direct basis for an NRC review when the underlying information remains valid and the relevant facility conditions, fire hazards, configurations, assumptions, analytical methods, and credited nuclear safety functions are sufficiently comparable to those required by the NRC. Examples include fire hazard analyses, fire hazard identification, fire modeling calculations and analysis methods, fire detection and alarm systems, fire suppression systems, fire protection equipment and system characteristics, fire barrier and fire-resistive construction performance, fire protection engineering analyses, and supporting test or qualification information. The potential for direct leveraging is strongest when the information addresses a technical attribute or analysis that is not dependent on DOE-specific regulatory requirements

or acceptance criteria. The potential for maximum leveraging is limited in the area of the post-fire safe shutdown analysis where NRC regulations require more specific details.

- *Example:* A DOE nuclear facility and a proposed NRC-regulated facility have substantially comparable fire compartmentation, combustible loading, fire barriers, safety-related equipment, fire detection, alarm, and suppression systems, and credited safety functions and analysis for post-fire safe shutdown.

Moderate leveraging

- Certain DOE information may be technically applicable but require additional NRC evaluation before it can support an NRC finding. NRC staff should evaluate whether the DOE methodology, assumptions, fire scenarios, plant configuration, credited safety functions, and performance or acceptance criteria are sufficiently comparable to those applicable to the NRC-regulated nuclear facility. Where differences exist, the applicant should identify those differences and provide additional analysis or justification as necessary to establish applicability to the NRC licensing basis.
- *Example:* DOE and NRC nuclear facilities use essentially the same fire barrier design and have comparable combustible loading, fire detection, alarm, and suppression systems but the NRC facility has a different arrangement of safety-related electrical equipment, NRC staff could leverage the DOE fire barrier analysis and fire detection, alarm, and suppression systems analysis and fire growth calculations for the common configuration. The applicant would then perform a supplemental fire hazards analysis to address the NRC facility-specific electrical equipment arrangement and any associated differences in the fire-safe shutdown analysis.

Limited leveraging

- Under a scenario with significantly different facility designs, NRC staff could use the DOE information as background and supplemental technical information to identify relevant fire hazards, fire scenarios, fire-protection features, analytical methods, and applicable industry codes and standards. However, given the significant differences between the facilities, the applicant would perform the fire hazards analysis and support engineering evaluations necessary to demonstrate compliance with applicable NRC requirements. Such information may nevertheless be leveraged to the extent that it demonstrates relevant technical characteristics or provides a sound technical basis for subsequent NRC evaluation. Where DOE requirements or determinations are relied upon, the applicant should identify the applicable DOE and NRC requirements, explain the relationship between them, and provide sufficient information to demonstrate that the underlying assumptions, hazards, methodologies, credited safety functions, and acceptance criteria support the applicable NRC finding.
- *Example:* DOE facility and a proposed NRC-regulated facility have some common active and passive fire protection systems and features, such as similar fire barriers, fire detection, alarm, and suppression systems, but differ significantly in facility configuration, combustible loading, safety-related equipment, operating conditions, and credited safety functions. The DOE information could inform the NRC's review but would not constitute the primary technical basis for the NRC's safety determination.

APPENDIX J—Technical Specifications

1. NRC Regulatory Requirements

Section 182(a) of the Atomic Energy Act of 1954, as amended, requires applicants for production and utilization facility operating licenses to provide the following:

[S]uch technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization ... of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

This requirement is implemented in 10 CFR in Parts 50 and 52 focused on large light water reactors (LWRs), in a technology neutral approach in Part 53 and in proposed 10 CFR Part 57 for microreactors and other reactors with comparable risk profiles.

10 CFR Parts 50 and 52

10 CFR 50.36, “Technical Specifications,” sections (a)(1) and (2) require, in part, that an applicant for license authorizing operation shall include in his application proposed technical specifications, and for applicants for a design certification or manufacturing license under part 52 of this chapter shall include in its application proposed generic technical specifications. Specifically, 10 CFR 50.36(c) states that the TS must include the following sections:

- (1) Safety Limits, Limiting Safety System Settings, and limiting control settings
- (2) Limiting Conditions for Operation
- (3) Surveillance Requirements
- (4) Design Features
- (5) Administrative Controls

For applicants pursuing licensing under Part 52, 10 CFR 52.47(a)(11) and 52.79(a)(30) require design certification (DC) and combined operating license (COL) applicants, respectively, to propose TS prepared in accordance with 50.36.

10 CFR Part 53

10 CFR 53.710, “Maintaining capabilities and availability of structures, systems, and components,” section (a) requires that Technical specifications must be developed, implemented, and maintained that define conditions or limitations on plant operations that are necessary to ensure that safety-related SSCs can fulfill their safety functions

The requirements in 10 CFR 53.710 describe the measures that must be provided for each commercial nuclear plant licensed under 10 CFR Part 53 such that the capabilities, availability, and reliability of plant structures, systems, and components (SSCs), when combined with corresponding programmatic controls and human actions, provide that the safety criteria defined in 10 CFR 53.210 and 10 CFR 53.220 will be met.

The basic structure of 10 CFR 53.710 is that measures for safety-related (SR) SSCs are established through TS, while measures for non-safety-related but safety-significant (NSRSS) SSCs are

required to be addressed in licensee-controlled documents and procedures. The requirements in 10 CFR 53.710(a) specify that TS must be developed, implemented, and maintained to ensure that SR SSCs can fulfill the safety functions identified under 10 CFR 53.230 and support meeting the safety criteria of 10 CFR 53.210.

Proposed 10 CFR Part 57

Proposed 10 CFR 57.60(a)(8)(vi) would require each applicant for a license to operate a nuclear reactor or nuclear plant to include proposed TS prepared in accordance with 10 CFR 50.36.

2. NRC Guidance and Related References

- Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Chapter 16, "Technical Specifications." (ML011340122).
- Draft Guide (DG) 1460 "Applications for Nuclear Power Plants," rev 2, Section C.1.4 (ML25255A088).
- NUREG-1430-1434, Standard Technical Specifications Series (ML21272A363, ML21259A155, ML21258A421, ML21272A357, ML21271A582).
- Draft NUREG-2271, "Guidelines for Preparing and Reviewing Applications Under 10 CFR Part 57," Chapter 13, "Technical Specifications." (ML25259A34).

3. Summary of DOE Requirements, Orders, and Standards

10 CFR 830.205(a) is the regulation for DOE technical safety requirements (TSR). It states:

(a) A contractor responsible for a Hazard Category 1, 2, or 3 DOE nuclear facility must:

(1) Develop technical safety requirements that are derived from the documented safety analysis;

(2) Prior to use, obtain DOE approval of technical safety requirements and any change to technical safety requirements; and

(3) Notify DOE of any violation of a technical safety requirement.

DOE G 423.1-1B, "Implementation Guide for Use in Developing Technical Safety Requirements," provides information to assist in the implementation of 10 CFR 830.205, Technical Safety Requirements. The guide provides methods acceptable to DOE for contractors to develop technical safety requirements (TSRs) for Hazard Category 1, 2, and 3 nuclear facilities. Section 2 notes that DOE TSR format guidance is based of NRC TS Improvement Program, but further notes TSR guidance does not apply to NRC licensed activities. Section 4.3.5.2 has a discussion of Specific Administrative Controls (SACs) NRC reviewers should be familiar with because SACs serve similar purposes to limiting safety system settings and limiting conditions for operation included in NRC TS for safety-related structures, systems, and components. Appendices A and B contain Format and Structure guidance and an Example TSR, respectively.

DOE-STD-1186-2016, "Specific Administrative Controls" provides requirements and guidance on acceptable methods for developing and implementing SACs at nuclear facilities operated by the DOE.

DOE-STD-1104-2016, "Review and Approval of Nuclear Facility Safety Basis and Safety Design Basis Documents" describes a framework and the criteria for DOE review and approval of safety basis documents consistent with 10 CFR 830. Section 5 of the document is dedicated to the review and approval of TSRs. Generally, the guidance provided identifies that determining the adequacy of TSRs rests upon being able to conclude that TSR provisions are appropriate and consistent with the DSA; and TSR provisions are clear, implementable, and consistent with DOE G 423.1-1B.

4. Staff Considerations on NRC and DOE Requirements and Guidance

This section contains high-level NRC staff observations on NRC and DOE requirements and guidance and how DOE requirements and standards may be used for various NRC licensing frameworks with respect to technical specifications.

Leveraging Across NRC Licensing Frameworks

The NRC licensing frameworks contain substantially similar requirements and acceptable approaches with respect to technical specifications, such that there are not significant differences between NRC frameworks in how DOE information is leveraged. Table 1 highlights these similarities across the frameworks.

Considerations on NRC and DOE Requirements and Guidance

Table J.1: Illustration between DOE TSRs and NRC TSs

Approving Organization	DOE	NRC Part 50/52	NRC Part 53	NRC Proposed Part 57
Base Regulation	10 CFR 830.205(a)(1) and table 3 of Appendix A to 830 Subpart B	10 CFR 50.36	10 CFR 53.710	Proposed 57.60(a)(8)(vi) would require TS to be in accordance with 10 CFR 50.36
Essential Elements	- Safety limits; - Operating limits (Limiting Control Settings and Limiting Conditions for Operation); - Surveillance requirements; - Administrative controls; - Use and application - Design features.	- Safety limits - Limiting Safety System Settings - Limiting Conditions for Operation. - Surveillance Requirements. - Design Features. - Administrative Controls.	-Limits on inventories of radioactive materials - Operating limits -limiting conditions for operation (LCOs) -SRs -Design Elements -Administrative Controls	- Safety limits - Limiting Safety System Settings - Limiting Conditions for Operation. - Surveillance Requirements. - Design Features. - Administrative Controls.

In general, TS and TSRs are similar in presentation; however, there are differences between the scope and depth of analyses and evaluations required to be submitted by applicants for DOE facilities compared to applicants for NRC licenses. Furthermore, fundamental differences exist in the criteria for determining which SSCs are required to be covered by TS for NRC compared to TSR for DOE. For example, while TS LCOs under Parts 50 and 52 are based upon meeting one or more of the four LCO criteria, TSR LCOs are typically based upon quantitative dose consequences to postulated receptors or qualitative assessments of hazards to facility workers. On a more minor note, DOE requires that TSRs contain a “Use and application” section, whereas the NRC regulations do not *require* such a section to be part of the TS. Instead, the NRC guidance *recommends* that TS contain a “Use and application” section. NRC staff should be cognizant of these differences and pay particular attention to the bases for a facility’s TSR.

Both NRC TSs and DOE reactor facility TSRs are to be derived from the analyses and evaluations submitted by the applicant to the approving authority. Both TSs and TSRs also have similar change control mechanisms. That is, changes to TS and TSR both generally require the approval of the associated application approving authority.

Use of DOE standards for NRC licensing

As discussed above, the NRC staff notes that DOE requirements and standards, such as those in 10 CFR 830.205, are generally similar to the NRC’s requirements for technical specifications; however, the NRC’s regulations may not fully be met by these DOE standards, for example, in the area of how a particular technical specification is met.

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE TSR information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

Consistent with the discussion in the main body of this white paper, this table outlines three of the key factors in determining the extent to which DOE authorization information related to technical specifications may be leveraged.

Table J.2: Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	Both TSs and TSRs are required to be based on the analyses and evaluations provided to obtain permission for facility operation. There are key differences in the criteria for determining which SSCs are required to be covered by TS for NRC compared to TSR for DOE. These differences should be appropriately considered by the NRC staff in determining the degree of leveraging of DOE information.

Application quality and justification of relevance of DOE authorization to NRC review	Since TS are based on an applicant's analyses, differences between the analyses in a DSA and those in a SAR will impact the extent to which TSRs may be leveraged to meet TS requirements. The closer the analyses in a DSA are to the scope and depth of those required for a SAR submitted to the NRC for an NRC license, the higher the degree of leveraging may be possible.
Safety-significance of information leveraged	TS are important to provide reasonable assurance of adequate protection of public health and safety by protecting against uncontrolled release of radioactive material. Given the focus of TS on important safety functions, this factor will generally limit the degree of NRC leveraging with respect to TS that can be achieved.

NRC Staff Approach to Leveraging DOE TSR Information

NRC staff should consider the similarity of analyses and evaluations required in the DSA compared to those expected in a SAR. The NRC staff should pay particular attention to any SACs in a DOE-authorized facility's TSRs. Generally, NRC TS would specify a limiting safety system setting or limiting condition for operation for a safety-related SSC that performs a function that rises to the level of an SAC in DOE TSRs. Therefore, while the SACs for a particular design may largely be appropriate for leveraging, the NRC staff should check that an applicant's proposed TS limiting safety system settings and limiting conditions for operation cover all safety-related SSCs necessary for commercial power operation. On this basis, NRC leveraging of DOE TSR information can generally be organized into three categories:

Maximum leveraging:

- In a maximum leveraging scenario, the NRC staff review will focus on verifying and confirming the TS requirements, analyses, and bases satisfy NRC requirements.
- *Example:* An example of maximum leveraging is when analyses and evaluations in the DSA supporting the TSRs adhere with a high degree of fidelity to the scope and depth of those required for establishing the bases for TS submitted to the NRC for an NRC license, which enables direct translation of those elements into TS.

Moderate leveraging:

- In a moderate leveraging scenario, NRC staff can use DOE information as a foundation, but meaningful additional work is required to reconcile differences and ensure that the resulting TS fully satisfy NRC requirements. Detailed NRC evaluation is needed to establish sufficient TS bases. The DOE analyses may also require additional NRC evaluation or gap resolution to support TS adequacy.
- *Example:* An example of moderate leveraging includes an application where the analyses and evaluations in the DSA adhere inconsistently to the scope and depth of those required for a SAR submitted to the NRC for an NRC license. Preapplication engagement could provide the applicant information to help it determine appropriate necessary modifications and additions and maximize the degree of leveraging.

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Limited leveraging:

- In a limited leveraging scenario, the NRC staff would use DOE information as relevant technical background information with limited ability to directly leverage the prior DOE authorization.
- *Example:* An example of limited leveraging is one where the analyses and evaluations in the DSA do not align with the scope and depth of those required for a SAR submitted to the NRC for an NRC license and the NRC LCO criteria for determining when TS are needed are not followed.

APPENDIX K—Site Characterization and Design for External Hazards

1. NRC Regulatory Requirements

Under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” and Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” 50.34, 52.17, and 52.79, address overall content of applications for construction permits (CPs), early site permits (ESPs), and combined licenses (COLs), respectively, and refer to 10 CFR Part 100, “Reactor Site Criteria,” for specific reactor siting criteria. For 10 CFR Part 53, “Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants,” subpart D, “Siting Requirements,” provides reactor siting criteria. For applications under the proposed 10 CFR Part 57, “Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles,” proposed 10 CFR 57.60(a)(5) would provide the siting criteria. These and other regulations pertaining to siting and design for external hazards for NRC applications are listed below.

10 CFR Part 50 governs the issuance of CPs and operating licenses (OLs) for production and utilization facilities. Siting information is primarily provided and reviewed at the CP stage.

- 10 CFR Part 50.34 provides the requirements for the content of CP applications, including the evaluation of the site on which the facility is located following the site evaluation factors identified in 10 CFR Part 100.
- Appendix A to 10 CFR Part 50, “General Design Criteria for Nuclear Power Plants,” provides general design criteria (GDC). Criterion 2 (GDC 2), “Design bases for protection against natural phenomena,” requires, in part, that structures, systems, and components (SSCs) important to safety be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. The design bases for these SSCs must reflect appropriate combinations of the effects of normal and accident conditions.
- Appendix S to 10 CFR Part 50, “Earthquake Engineering Criteria for Nuclear Power Plants,” provides seismic and geologic design basis requirements pursuant to 10 CFR 100.23, “Geologic and seismic siting criteria.”

10 CFR Part 52 governs the issuance of ESPs, standard design certifications (DCs), COLs, standard design approvals (SDAs), and manufacturing licenses (MLs) for nuclear power plants. Siting information is provided in support of ESP and COL applications.

- 10 CFR 52.17(a)(1)(vi) for an ESP and 10 CFR 52.79(a)(1)(iii) for a COL state that the technical information in the site safety analysis report or final safety analysis report (FSAR), respectively, shall include “geologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated.”

10 CFR Part 53 governs the issuance of ESPs, CPs, OLs, COLs, DCs, SDAs, and MLs in a fully risk-informed, performance-based framework for commercial power plants. Siting information is provided in support of ESP, CP and COL applications.

- Part 53 Subpart D includes requirements to identify and assess site characteristics related to meteorology, geology, hydrology and other areas.

Proposed 10 CFR Part 57.60(a)(5), if approved, would provide the siting requirements for microreactors and other reactors with comparable risk profiles.

10 CFR Part 100 establishes approval requirements for proposed sites for stationary power and testing reactors subject to 10 CFR Part 50 or Part 52. 10 CFR 100.10 and 100.20, both titled, “Factors to be considered when evaluating sites,” apply to testing reactors and power reactors respectively and state the NRC will take certain factors into consideration in determining the acceptability of a site, including, in part: population density; characteristics of the site environs; and the physical characteristics of the site, including seismology, meteorology, geology, and hydrology. For example, 10 CFR 100.21 describes the requirements for non-seismic siting criteria for commercial power reactors, such as establishing an exclusion area boundary (EAB) and low population zone (LPZ), evaluating site atmospheric dispersion characteristics and dispersion parameters, evaluating potential hazards associated with nearby transportation routes, industrial and military facilities, and civilian and military airports.

Changes proposed to 10 CFR Part 100 (91 FR 44560) would allow an applicant to use an unmitigated consequence analysis to determine the appropriate level of site characterization for a power reactor application with lower consequence power reactors following the siting criteria in proposed 10 CFR Part 100 Subpart A and higher consequence power reactors following the siting criteria in 10 CFR Part 100 Subpart B.

2. NRC Guidance and Related References

Guidance on Overall Application Content

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” provides guidance to NRC staff in performing safety reviews of CP or OL applications, including requests for amendments under 10 CFR Part 50, and ESP, DC, COL, SDA, or ML applications under 10 CFR Part 52 (including requests for amendments). Though direct applicability is limited to light water reactors, it provides detailed, hazard specific guidance that serves as a baseline to safety reviews of applications. Hazard-specific chapters of the Standard Review Plan (SRP) are identified below.
- DANU-ISG-2022-02, “Advanced Reactor Content of Application Project Chapter 2 – Site Information” (ML23277A140), provides guidance on the contents of a risk-informed, performance-based, non-LWR application for a CP or OL under 10 CFR Part 50 or for a COL, ML, SDA, or DC under 10 CFR Part 52. The guidance found in this ISG supports the development of the portion of non-LWR application associated with an applicant’s “Site Information,” and provides guidance to NRC staff on how to review such an application.
- NUREG-2271, “Guidelines for Preparing and Reviewing Applications Under 10 CFR Part 57 — Draft Report for Comment,” (ML26111A077) provides draft guidance on the content and review of applications that would be submitted under proposed 10 CFR Part 57 for microreactors and other reactors with comparable risk profiles. Specific guidance on site characterization and external hazards evaluations is provided in Chapter 2 and Appendix F. Guidance to the NRC staff on how to review such an application is also provided.
- Draft Guide (DG)-4036, “Graded Approach to Site Characterization for New Reactor Applicants,” (ML25272A272) provides guidance for on graded approaches to site characterization for all external hazards licensed under 10 CFR Parts 50, 52 and 53. This includes draft guidance on performing an unmitigated consequence analysis to determine the nuclear siting category (NSC) which informs graded site characterization for 10 CFR Parts 50, 52, and 53 applications. Per the proposed changes to Part 100, the determination of NSC informs subpart selection for applications submitted under 10 CFR Parts 50 and 52.

Guidance Related to Human-Induced External Events

- RG 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," describes methods for determining the risk of damage caused by an explosion (including from liquids, cryogenically liquefied hydrocarbons, vapor clouds, etc.) at a nearby facility or on a transportation route. (ML21260A242).
- RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," describes methods for evaluating the habitability of a nuclear power plant control room during a postulated hazardous chemical release. (ML21253A071).
- NUREG-0800 sections pertaining to human-induced external events from are highlighted below:
 - Section 2.2.1-2.2.2, "Identification of Potential Hazards in Site Vicinity," focuses on potential hazards from presence of hazardous materials or activities conducted at nearby facilities that are currently present or reasonably be expected to be present during the projected lifetime of the proposed facility.
 - Section 2.2.3, "Evaluation of Potential Accidents," evaluates the applicant's frequency analyses of potential accidents involving hazardous materials or activities on site and at the nearby industrial and military facilities and transportation routes to determine the completeness of the applicant's analyses and the bases upon which the facility design did or did not accommodate those potential accidents.
 - Section 3.5.1.6, "Aircraft Hazards," evaluates the applicant's assessment of aircraft crash hazards to the proposed facility so as to determine whether the aircraft hazards would be a design-basis concern.

Guidance Related to Meteorology

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," presents criteria for an acceptable onsite meteorological monitoring program and the resulting meteorological database that may be used as input to the atmospheric dispersion estimates. (ML070350028).
- RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Lighter-Water-Cooled Reactors." (ML003740354)
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," presents criteria for characterizing atmospheric dispersion conditions for evaluating the consequences of radiological releases to the EAB and LPZ. (ML12216A014).
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," describes an acceptable method to meet the regulations concerning the determination of atmospheric relative concentration (χ/Q) values in support of design-basis control room habitability assessments at nuclear power plants. (ML031530505).
- RG 1.249, "Use of ARCON Methodology for Calculation of Accident-Related Offsite Atmospheric Dispersion Factors," provides guidance for determining atmospheric relative concentration (χ/Q) values, also known as atmospheric dispersion factors, in support of modeling onsite releases to offsite boundaries from a design-basis accident, including procedures for using the ARCON code to estimate χ/Q s at the EAB and the outer boundary of the low-population zone (LPZ) out to distances of 1,200 meters (m) (3,937 feet (ft)) from the nearest edge of a building within the nuclear island. (ML22024A241).

- RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," (Ref. 66) describes an acceptable method for selecting the design-basis tornado and design-basis tornado-generated missiles that a nuclear power plant should be designed to withstand to prevent undue risk to the health and safety of the public within the contiguous United States. (ML070360253).
- RG 1.221, "Design-Basis Hurricane and Hurricane Missile for Nuclear Power Plants."
- DC/COL-ISG-07, "Interim Staff Guidance on Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures." (ML110940300).
- NUREG-0800 sections pertaining to meteorology are highlighted below:
 - Section 2.3.1, "Regional Climatology"
 - Section 2.3.2, "Local Meteorology"
 - Section 2.3.3, "Onsite Meteorological Measurements Programs"
 - Section 2.3.4, "Short-term Dispersion Estimates for Accident Releases"
 - Section 2.3.5, "Long-Term Atmospheric Dispersion Estimates for Routine Releases" (ML052340514).

Guidance Related to Hydrology

- RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," describes methods and procedures to establish ultimate heat sink (UHS) features of plant systems required by NRC regulations. (ML14107A411).
- RG 1.59, "Design Basis Floods for Nuclear Power Plants," describes acceptable methods for use in the determination of design-basis floods for nuclear power plants. (ML19289E563).
- RG 1.102, "Flood Protection for Nuclear Power Plants," describes the types of flood protection that the NRC staff finds acceptable for the SSCs important to safety identified in RG 1.29. This guide also describes acceptable methods for protecting nuclear power plants from flooding effects resulting from probable maximum precipitation (PMP). (ML003740308).
- RG 1.256, "Guidance for Assessment of Flooding Hazards Due to Water Control Structure Failures and Incidents," provides guidance on acceptable methods for evaluating design-basis flooding hazards due to failure or other incidents at manmade water control structures including, but not limited to, dams and levees. (ML25129A075).
- NUREG/CR-7046, "Design-Basis Flood Estimation for Site Characterization at Nuclear Power Plants in the United States of America," provides a description of approaches and methods for estimation of the design-basis flood at nuclear power plant sites. (ML11321A195).
- NUREG-0800, section 2.4, "Hydrology," provides guidance for reviewing water-related site characteristics affecting nuclear power plant safety focusing on various flood causing mechanisms, low water events, flooding protection requirements, groundwater, and accidental release of radioactive liquid effluents. The SRP contains general review guidance related to site characteristics and site parameters, together with site-related design parameters and design characteristics, as applicable.
 - Section 2.4.13, "Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters" describes the technical basis and provides guidance for the evaluation of site characteristics to assess the effects of a postulated accident scenario resulting from failure of a tank by assessing the effluent transport in groundwater and surface water and the effects on existing and future water use. (ML052340514).

- BTP 11-6, Postulated Radioactive Releases due to Liquid-Containing Tank Failures,” provides the NRC staff with guidance in defining the mechanism of the failure assumptions used for the analysis, and approaches applied in assessing the radiological impact. (ML15027A401).

Guidance Related to Geology, Seismology and Geotechnical Engineering

- RG 4.26, “Volcanic Hazards Assessment for Proposed Nuclear Power Reactor Sites,” provides guidance for facilitating the NRC staff’s review of volcanic hazards assessments performed by applicants to support the siting of new nuclear power reactors. The RG also provides applicants with methods and approaches the NRC staff consider acceptable for the assessment of volcanic hazards in applications. (ML23167A078).
- RG 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” provides guidance on seismic site characterization and contents of applications for new nuclear power plants. (ML070310619).
- RG 1.132, “Geologic and Geotechnical Site Characterization Investigations for Nuclear Power Plants,” provides guidance on field investigations for determining the geologic, geotechnical, geophysical, and hydrogeologic characteristics of a prospective site for engineering analysis and design of nuclear power plants. (ML21298A054).
- RG 1.138, “Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants,” provides guidance on sampling, storage, and laboratory investigations of the properties of soils for engineering analysis and design of nuclear power plants. (ML14289A600).
- RG 1.198, “Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites,” provides guidance to applicants on acceptable methods for evaluating the potential for earthquake-induced instability of soils resulting from liquefaction and strength degradation, including the conditions under which the potential for such response should be addressed in safety analysis reports and the procedures and criteria currently applied to assess the liquefaction potential of soils ranging from gravel to clays. (ML033280143).
- NUREG-0800 sections pertaining to geology, seismology and geotechnical engineering are highlighted below:
 - Section 2.5.1, “Basic Geologic and Seismic Information,” provides guidance on the review of geologic characteristics of the site and its environs in sufficient scope and detail to evaluate the suitability of the site for the proposed facility.
 - Section 2.5.2, “Vibratory Ground Motion,” provides guidance on the review of seismological, geological, geophysical, and geotechnical investigations carried out to determine the site-specific ground motion response spectra, and eventually the safe shutdown earthquake ground motion for the site.
 - Section 2.5.3, “Surface Deformation,” provides guidance on the review of surface deformation or subsidence of geologic features due to faulting, dissolution of carbonate rock or salt, diapirism, and anthropogenic effects.
 - Section 2.5.4, “Stability of Subsurface Materials and Foundations,” provides guidance on the review of the characteristics and stability of the soil and rock underlying the site that could affect the safe design and siting of the plant.
 - Section 2.5.5, “Stability of Slopes,” provides guidance on the review of the stability of all earth and rock slopes both natural and manmade (cuts, fill, embankments, dams, etc.) whose failure, under any of the conditions to which they could be exposed during the life of the plant, could adversely affect the safety of the plant.

- Section 3.8.5, “Foundations,” provides guidance on the review of the design of the foundation and its structural characteristics of all seismic category I structures, including any seismic interactions with the foundation(s) of adjacent structures. (ML052340514).

3. Summary of DOE Requirements, Orders, and Standards

10 CFR 830.204 (b)(2) requires the “systematic identification of both natural and man-made hazards associated with the facility.” 10 CFR 830.204(b)(3) requires the evaluation of “normal, abnormal, and accident conditions, including consideration of natural and man-made external events, identification of energy sources or processes that might contribute to the generation or uncontrolled release of radioactive and other hazardous materials, and consideration of the need for analysis of accidents which may be beyond the design basis of the facility.”

For hazard category 1, 2 and 3 nuclear facilities, NE-STD-1020-2025 provides requirements and guidance for (a) the analysis and design of facility SSCs necessary to implement the natural phenomena hazard (NPH) requirements of DOE NE Order (O) 420.1, Facility Safety, and (b) the use and reference of industry building codes and voluntary consensus standards in the NPH analysis and design of SSCs in DOE facilities. In addition to general criteria and guidance for NPH, hazard specific requirements and guidance are provided for seismic design, extreme straight-line wind, tornado and hurricane design, flood, seiche and tsunami design, lightning design, precipitation design, and volcanic eruption design to inform the determination of the natural phenomena design category (NDC). This standard also addresses the period reevaluation of NPH assessments for existing nuclear facilities. As a starting point, this standard specifies that a new facility should select a site the minimizes NPH to the extent possible.

4. Staff Considerations on NRC and DOE Requirements and Guidance

This section contains high-level NRC staff observations on NRC and DOE requirements and guidance and how DOE requirements and standards may be used for various NRC licensing frameworks with respect to site characterization and the evaluation of external hazards.

Per above NE-STD-1020-2025 categorizes nuclear facilities by NDC based on an unmitigated consequence analysis following ANSI/ANS-2.26-2004 (R2021); however, alternative approaches may be available (e.g., DOE-STD-3009-2014). In the NRC licensing framework, DG-4036 uses ANSI/ANS-2.26-2004 (R2021) to perform an unmitigated consequence analysis to determine the nuclear siting category (NSC) to inform graded site characterization and external hazard analysis for nuclear power reactor sites.

Leveraging across NRC Licensing Frameworks

The NRC licensing frameworks contain substantially similar requirements and acceptable approaches with respect to site characterization and the evaluation of external hazards, such that there are not significant differences between NRC frameworks in how DOE information is leveraged.

Considerations on NRC and DOE Requirements and Guidance

For site characterization and the evaluation of external hazards, there are two distinct pathways for leveraging DOE information with different practical considerations for each pathway. The first pathway is a facility that was authorized, constructed and operated under the DOE framework and is converting the same facility to an NRC-issued commercial license. The second pathway is for an

application for a new site deploying a reactor technology that was previously authorized, constructed and operated under the DOE framework.

For the first pathway, the NRC requirements for site characterization and hazard evaluation should inform the scope of site investigations and the level of hazard analysis performed. Although NE-STD-1020-2025 is generally aligned with NRC requirements for natural phenomena design category (NDC) 5 facilities, pre-application engagement with the NRC is encouraged for all facilities, particularly those assigned to lower NDCs. Recommended pre-application engagements would be beneficial to align on the appropriate level of site characterization and hazard analysis commensurate with applications for longer-term operations (e.g., 40-year license), that would meet the requirements for DOE authorization and the subsequent NRC commercial license for other than NDC-5 facilities. For example, some analyses or data may be sufficient to meet both NRC and DOE requirements, while other analyses or data may need enhancements to meet both requirements.

For the second pathway deploying a previously authorized technology at a new commercial site, applicable NRC guidance for siting and external hazards evaluation should be followed similar to any new reactor application. The site characterization and hazard analysis may include the implementation of a graded approach that could be further informed from the construction and operational experience at the authorized DOE site. However, because each deployment site will have a unique hazard profile, the characterization of the new site should be performed to meet the applicable NRC regulatory requirements and consistent with the available guidance.

In some cases a previously authorized technology may be deployed at a different location on the same site. In this case there may be opportunities to leverage information for the new location if the surface and subsurface geological and geotechnical conditions of the new footprint area are substantially similar and proximity to nearby hazard sources are comparable (e.g., similar distance to sources of flooding).

Because the second pathway would only be able to leverage DOE authorization information for site characterization and hazard analysis that is demonstrated to be representative of the new deployment site, similar to any other new reactor application, the content of this paper focuses on leveraging DOE information to support the conversion of an existing DOE-authorized facility to an NRC commercial license.

Use of DOE standards for NRC licensing

Given the general similarities between the NRC and DOE approaches, use of DOE standards in the area of site characterization and external hazards may be a starting point for developing an NRC licensing application. However, the applicant would need to address the key difference between the NRC and DOE frameworks and ensure that explicit documentation of the site characterization and external hazards commensurate with applications for longer-term operations (e.g., 40 year license), is provided for NRC review.

- **NDC/NSC Determination:** NE-STD-1020-2025 references American National Standards Institute (ANSI)/American Nuclear Society (ANS)-2.26-2004 (R2021) for establishing SSC limit states and categorization. Section 6.2 of ANSI/ANS 2.26 (R2021) describes the methodology for conducting an unmitigated consequence analysis to support the seismic categorization of SSCs. Similarly, sections 3, 4, 5, and 7 address seismic; wind, tornado, and hurricane; flood; and precipitation hazards, respectively, and direct users to the same ANS 2.26 methodology. An unmitigated consequence analysis should be performed to determine

the appropriate NDC for DOE authorizations or nuclear siting category (NSC) for NRC applications. As an alternative, applicants may conservatively assign SSCs to NDC-5 or NSC-5 without conducting an unmitigated consequence analysis.

- **Seismic Hazards:** NRC practices for SDC-5 facilities are generally aligned with the DOE practices outlined in NE-STD-1020-2025. This includes alignment on the use of American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI) 43-19, ANSI/ANS-2.27-2020, ANSI/ANS-2.29-2020, RG 1.208, NRC Research Information Letter 2021-15 and NUREG/CR-6728. For other than SDC-5 facilities, pre-application engagement with the NRC is encouraged for prospective applicants to align on a suitable approach.
- **Meteorology:** The use of RG 1.145 for design basis accident releases and RG 1.23 for atmospheric dispersion evaluations is appropriate for NRC licensing. Similarly, the application of RG 1.76 for tornado hazards and RG 1.221 for hurricane hazards aligns with the expectations of both DOE and NRC regulatory frameworks. Prospective applicants are encouraged to engage with the NRC during the pre-application phase to align on the use of alternative codes or standards to support DOE authorizations.
- **Flooding hazards:** Section 5.1.2 of NE-STD-1020-2025 identifies the return periods applicable to design-basis flood hazards and references NRC JLD-ISG-2012-05 and JLD-ISG-2016-01. NRC guidance indicates that the probable maximum flood (PMF) be derived from the probable maximum precipitation (PMP) using conservative hydrologic and hydraulic assumptions. For power reactor applications, use of the flood hazard methodologies described in NUREG-0800, RG 1.59, and NUREG/CR-7046 generally correspond to DOE requirements for flood design category (FDC)-5. To demonstrate compliance, applicants should supplement any DOE probabilistic flood hazard assessment with supporting evidence to show how it meets or is bounded by design-basis flood heights, warning time, etc. produced using NRC PMP/PMF methodology.

Table 5-3 of NE-STD-1020-2025 identifies combinations of design-basis flood events for evaluation. NRC guidance similarly specifies flood hazard combinations in Appendix H of NUREG/CR-7046 to ensure a comprehensive deterministic assessment of flooding hazards. Under a probabilistic flood hazard assessment (PFHA), however, all credible flooding mechanisms and event combinations must be considered across a range of annual exceedance probabilities to estimate flood levels corresponding to the target return period for each FDC. As a result, the combinations identified in Table 5-3 do not necessarily bound all hazard scenarios considered in a PFHA. Applicants that begin with the NRC flood hazard framework and then incorporate the additional design and performance requirements specified by DOE are likely to achieve compliance with both regulatory approaches.

- **Volcanic hazards:** Section 8.1 of NE-STD-1020-2015 specifies that volcanic hazards should be evaluated for Quaternary-age volcanic sources located within 250 miles of the site. NRC RG 4.26 establishes a minimum regional study area extending 200 miles from the site but also requires consideration of Quaternary-age volcanic deposits located within 25 miles of the site, even when the volcanic source lies beyond the 200-mile radius. In such cases, the study area may need to be expanded to include the source volcano associated with those deposits. To streamline compliance with both DOE and NRC guidance, applicants could define the region of interest as extending to either 250 miles from the site or to the source of any Quaternary-age volcanic deposits identified within 25 miles of the site, whichever distance is greater.

- **Human-induced External Hazards:** NRC NUREG-0800, section 3.5.1.6 "Aircraft Hazards" and DOE Standard 3014, "Accident Analysis for Aircraft Crash into Hazardous Facilities," give similar methods of analysis for aircraft crash hazards, and either may be used for NRC licensing. NRC applications should follow the guidance in RG 1.91 "Evaluation of Explosions Postulated to Occur at Nearby Facilities and Transportation Routes Near Nuclear Plants," Revision 3, to analyze explosion-related hazards for NRC applications.
- **Geologic and Geotechnical Site Characterization:** NRC guidance for conducting geologic and geotechnical site characterization may also be used to support development of the DOE safety basis. Pre-application engagement is encouraged for prospective applicants following DOE guidance on geologic and geotechnical site investigations described in DOE-STD-1022-94, "Natural Phenomena Hazards Site Characterization Criteria," to align on the appropriate scope and scale of geologic and geotechnical site characterization of the selected site.

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE site characterization and external hazards evaluation information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement commensurate with applications for longer-term operations (e.g., 40 year license). Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

Consistent with the discussion in the main body of this white paper, this table outlines three of the key factors in determining the extent to which DOE authorization information related to site characterization and external hazards evaluation may be leveraged. Because each deployment site may have a different external hazard profile, applicants relying on a DOE authorized technology at a new site would be subject to all applicable regulatory requirements for the selected NRC licensing framework.

Table K.1: Factors for NRC Leveraging of DOE Information for existing sites converting to an NRC license

Factor	Description
Similarity of DOE and NRC requirements and guidance	General similarities between the NRC and DOE approaches to site characterization and external events create greater opportunities for leveraging.
Application quality and justification of relevance of DOE authorization to NRC review	The NRC provides guidance on the use of existing site characterization information in DG-4036. This justification addresses data quality, methods of data collection and analysis, representativeness of the existing data to the proposed site, and alterations of the site since data was obtained. If these conditions are addressed, the existing data, from DOE authorization or other sources, may be used in the development of an NRC application. The general approach to site characterization implemented for the DOE authorization may be suitable for the characterization of

	a subsequent deployment site for the same technology. NRC guidance on determining the appropriate level of graded site characterization based on an unmitigated consequence analysis should be followed in the development of the site investigation program.
Safety-significance of information leveraged	Because each deployment site may have a different external hazard profile that needs to be evaluated, this factor may limit the amount of NRC leveraging that can be achieved.

NRC Staff Approach to Leveraging DOE Site Characterization and External Hazards Information

NRC staff should consider the similarity of site characterization and external hazards requirements and guidance and should seek to understand the aspects of the application that were reviewed by DOE as a part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the application. A DOE authorized site planning to convert to an NRC license should plan site characterization to ensure compliance with applicable NRC regulatory requirements since certain site characterization activities can only be performed before construction is completed. Insights gained during pre-application engagement, including observation of the DOE authorization process, can support a more efficient review. Based on these factors, NRC leveraging of DOE site characterization and external hazards information for an existing site converting to an NRC license can generally be organized into three categories:

Maximum leveraging:

- If sufficient, representative information is available from the DOE authorization process, it may be leveraged in support of an NRC application.
- Any site characteristic that will be covered, altered, or otherwise rendered inaccessible or unable to obtain additional information once the construction of the DOE authorized facility commences should be carefully considered to support a future NRC license application.
 - If sufficient information from the DOE authorization process is not available to be fully leveraged in support of an NRC license, additional data collection from the surrounding area may be needed to interpolate that data with appropriate consideration of uncertainty to characterize the site.
- *Example:* Types of information may include geologic and geotechnical characterization of the reactor footprint and foundation-bearing geologic materials, such as borings, downhole geophysics, installation of monitoring wells, and other in situ data collection tools or methods of analysis

Moderate leveraging:

- Site characteristics for which representative data may still be collected following construction may be leveraged to support an NRC license application.
- Alternatively, if sufficient, representative data is not available, it may still be reasonably obtained following NRC guidance to support an NRC license.
- *Example:* Onsite meteorological data for the site may be obtained before, during, or after construction of a DOE authorized facility.

Limited leveraging:

THIS NRC STAFF WHITE PAPER HAS BEEN PREPARED AND IS BEING RELEASED TO SUPPORT INTERACTIONS WITH THE PUBLIC. THIS PAPER HAS NOT BEEN SUBJECT TO NRC MANAGEMENT AND LEGAL REVIEWS AND APPROVALS, AND ITS CONTENTS SHOULD NOT BE INTERPRETED AS OFFICIAL AGENCY POSITIONS. A PROSPECTIVE APPLICANT SHOULD NOT USE THE CONTENTS OF THIS PAPER OR RELY ON ITS CONTENTS IN PREPARING AN APPLICATION.

- The NRC provides guidance on demonstrating the representativeness of existing site characterization for a new site.
- Site characterization information from an existing site may not be suitable for use at a new location unless the data is demonstrated to be representative of the site and hazard conditions at the new site.
- *Example:* Co-location or close proximity does not automatically imply that site characterization information is representative of a new site due to factors such as subsurface variability, proximity to hazard sources or other factors. Site characterization information developed for a co-located or nearby site would need to follow NRC guidance on the demonstrating the representativeness of existing site characterization information.

Notably, as mentioned above, another factor related to DOE leveraging is the ability to obtain additional site characterization information after the facility has been constructed. Once a facility is constructed the opportunity to obtain certain site characterization information is no longer feasible. Accordingly, for applicants converting a DOE authorized facility to an NRC licensed-facility, certain site characterization data must be obtained during the DOE authorization process to ensure it is available to be leveraged for the subsequent NRC license application. This would include information related to the geologic and geotechnical properties of the reactor footprint and foundation-bearing materials, borings, downhole geophysics, installation of monitoring wells, and other in situ data collection tools or methods of analysis. There are other external hazards where the collection of additional site characterization information would still be possible for a constructed facility, such as onsite meteorological data.

APPENDIX L—Fuel Performance

1. NRC Regulatory Requirements

10 CFR Part 50 provides requirements related to fuel qualification and performance, including:

- 10 CFR 50.43(e), which states that reactor designs that differ significantly from light water reactor (LWR) designs licensed before 1997 will be approved only if there has been appropriate demonstration of their safety features. In particular, 10 CFR 50.43(e)(1)(i) and (ii) require demonstration of safety feature performance and interdependent effects through analysis, appropriate test programs, experience, or a combination thereof; and 10 CFR 50.43(e)(1)(iii) requires that sufficient data exist on the safety features to assess the analytical tools for safety analysis over a sufficient range of plant conditions. Fuel is one of the features that would require this demonstration.
- 10 CFR 50.34(a)(1)(ii)(D), which requires construction permit (CP) applicants to evaluate a postulated fission product release from the core into the containment. Fuel performance under accident conditions governs the postulated release.
- Per 10 CFR 50.46(a)(1)(i), each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated LOCAs conforms to the criteria set forth in Section 50.46(b). ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. The 10 CFR 50.46(b) criteria require that post quench ductility, coolability, and long-term cooling are maintained. LOCA analysis typically involves a substantial amount of fuel performance work to demonstrate that the 10 CFR 50.46(b) criteria are met.
 - In the Executive Order 14300 “Modernizing Reactor Licensing, Safety Oversight, and Siting Practices [W1-6]” Rulemaking, the NRC staff have proposed to risk inform LOCAs in a voluntary alternative to 10 CFR 50.46, 10 CFR 50.46a, where LOCAs above a certain transition break size would be recategorized as beyond design basis accidents. In this rulemaking, performance-based fuel technology neutral criteria are proposed to be established for LOCAs. These criteria, if approved, will state that fuel designs must have NRC-approved limits that (1) Address cladding degradation phenomena; (2) Maintain fuel coolability; (3) Avoid explosive concentration of combustible gas; and (4) Demonstrate that, after any calculated successful initial operation of the ECCS, the ECCS must provide sufficient coolant to remove decay heat and prevent further cladding failure for the extended period of time required by the long-lived radioactivity remaining in the fuel.
- Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100, “Reactor Site Criteria,” and 10 CFR 50.67, “Accident source term,” for postulated accidents.
- Appendix A to Part 50 provides general design criteria (GDC) which establish the minimum requirements for principal design criteria (PDC) for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The GDC are also considered to be generally applicable to other types of

nuclear power units and are intended to provide guidance in establishing the PDC for such other units. Several GDC relate to fuel performance, as discussed below.

- GDC 2, “Design bases for protection against natural phenomena,” requires that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunamis, and seiches without loss of capability to perform their safety functions. Fuel is typically an SSC that requires protection from natural phenomena, particularly seismic events.
- GDC 10, “Reactor design,” states that the reactor core and associated coolant, control, and protection systems shall be designed with the appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. GDC 10 establishes specified acceptable fuel design limits to ensure that the fuel is “not damaged.” That means that fuel rods do not fail, fuel system dimensions remain within operational tolerances, and functional capabilities are not reduced below those assumed in the safety analysis.
- GDC 27, “Combined Reactivity Control Systems Capability” states that the reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.
- GDC 28, “Reactivity Limits” states that reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither: (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.
- GDC 35, “Emergency Core Cooling,” as it relates to providing an ECCS to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

The requirements related to fuel performance in Part 50 are applicable to Part 52.

10 CFR Part 53 provides a number of requirements related to fuel qualification and performance, including:

- 10 CFR 53.210 establishes offsite dose criteria against which design basis accident analyses, including credited fuel retention and release behavior, are evaluated.
- 10 CFR 53.230 requires limitation of radioactive-material release during normal operation and licensing-basis events; supporting functions include control of reactivity, heat generation, heat removal, and chemical interactions.
- 10 CFR 53.440(a)(1), which requires “analysis, appropriate test programs, prototype testing, operating experience, or a combination thereof” to demonstrate design features meet the functional design criteria, similar to the requirements of 10 CFR 50.43(e).
- 10 CFR 53.440(c) requires that the materials used for each safety-related and non-safety-related but safety-significant system, structure, and component (SSC) be qualified for their service conditions over the design life of the SSC.

- 10 CFR 53.440(d) requires evaluation of degradation from aging, fatigue, chemical interactions, temperature, irradiation, and other environmental factors to inform design and integrity assessment programs.
- 10 CFR 53.440(h) requires plants to have the capability to maintain long-term cooling of the fuel during normal operation and following any licensing basis event.
- 10 CFR 53.450, which includes general analysis requirements, and 10 CFR 53.450(d) specifically requires analytical codes (including fuel performance, thermodynamics, reactor physics, and mechanistic source term codes) to be qualified for the range of conditions for which they are to be used. This requirement is required to be met for:
 - Standard design approvals (10 CFR 53.1209(b)(2))
 - Design certifications (10 CFR 53.1239(a)(19))
 - Manufacturing licenses (10 CFR 53.1279(a)(2))
 - Construction permits (10 CFR 53.1309(a)(2)), though design and analyses may be preliminary in accordance with 10 CFR 53.1309(a)(2)(ii) and the CP application may include research and development items (which may include code validation activities) in accordance with 10 CFR 53.1309(a)(2)(iii)
 - Operating licenses (10 CFR 53.1369(b))
 - Combined licenses (10 CFR 53.1416(a)(2))
- 10 CFR 53.450 also requires estimation of doses for licensing basis events, which would require an estimation of fuel failures.

Proposed 10 CFR Part 57 would provide several requirements related to fuel qualification and performance, including:

- Proposed 10 CFR 57.25 would provide a requirement to demonstrate applicability of Part 57 through analysis showing a dose criterion of 1 rem will be met.
- Proposed 10 CFR 57.30 would establish design criteria attributes, including reactivity control, net negative reactivity feedback, heat removal, and fission product retention,
- Proposed 10 CFR 57.60(a)(1)(iii) would require CP and OL applicants to provide the evaluation required by proposed 10 CFR 57.25 in the plant final safety analysis report (FSAR)
- Proposed 10 CFR 57.60(a)(1)(vii) would require analysis, testing, etc., similar to the requirements of 10 CFR 50.43(e) as discussed above.
- Proposed 10 CFR 57.155(a) would require proposed 10 CFR 57.60(a)(1) to be met by ML applicants.

2. NRC Guidance and Related References

Guidance documents related to fuel qualification and performance include, but are not limited to:

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor (LWR) Edition" (SRP), Section 4.2, "Fuel System Design," provides review guidance to the NRC staff concerning the establishment of specified acceptable fuel design limits to assure: (1) the fuel system is not damaged because of normal operation and AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) core coolability is maintained. (ML070740002).

- NUREG-2246, “Fuel Qualification for Advanced Reactors,” provides guidance on fuel qualification for non-LWRs. This NUREG builds on guidance for LWRs contained in NUREG-0800, Section 4.2. (ML22063A131).
- NUREG/CR-7305, “Metal Fuel Qualification,” provides a generic assessment of uranium-zirconium alloy metallic fuel following the NUREG-2246 process. (ML23214A065).
- NUREG/CR-7299, “Fuel Qualification for Molten Salt Reactors,” provides guidance on molten salt fuel qualification. (ML22339A161).
- ATF-ISG-2020-01, “Supplemental Guidance Regarding the Chromium-Coated Zirconium Alloy Fuel Cladding Accident Tolerant Fuel Concept” provides supplementary guidance to NUREG-0800, Section 4.2 specific to LWR applications involving fuel products with chromium-coated zirconium alloy cladding. (ML19343A075).
- Regulatory Guide (RG) 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non- Light-Water Reactors” endorses NEI 18-04, which defines risk-informed, performance based, and technology-inclusive processes for the selection of licensing basis events; safety classification of structures, systems, and components (SSCs); and determination of defense-in-depth adequacy for non-LWRs. NEI 18-04 provides a frequency-consequence target curve that is used to assess events, SSCs, and programmatic controls. Because the fuel is the primary source of radionuclides in the reactor, its performance under both normal and off-normal conditions plays a key role in evaluating consequences for the NEI 18-04 process. (ML20091L698).
- RG 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” discusses development of principal design criteria (PDC) for non-LWRs. RG 1.232 recognized that for certain non-LWR designs, specified acceptable radionuclide release design limits (SARRDLs) may be more appropriate than the use of SAFDLs, such as in certain reactors that contain tristructural isotropic TRISO fuel. (ML17325A611)
- RG 1.236, “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents,” provides guidance of the analyses of control rod ejection accidents for PWRs and control rod drop accidents for boiling-water reactors. RG 1.236 describes acceptable methods and procedures for analyzing these accidents, analytical limits for demonstrating compliance with GDC 28, and defines fuel cladding failure thresholds and limits to ensure coolability. (ML20055F490).
- As part of the proposed rule changes related to LOCAs in the Executive Order 14300 W1-6 Rulemaking, the NRC staff developed several draft RGs (DGs) related to establishing performance-based ECCS limits for cylindrical zirconium-alloy clad uranium dioxide fuel. DG-1261, Revision 1 provides guidance on measuring breakaway oxidation, DG-1262, Revision 1, provides guidance on determining post-quench ductility, and DG-1263 provides guidance on establishing the analytical LOCA limits. Additionally, the proposed rule package contains a DG on addressing the consequences of fuel dispersal during a LOCA, including estimating the amount of fragmented fuel that will disperse into the coolant during a postulated LOCA following ballooning and burst of the cladding. (ML24284A338).

- DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap,” provides guidance on NRC review of fuel qualification for construction permit applications for non-LWRs. This guidance states that fuel is not required to be qualified at the CP stage, and NRC staff should focus on understanding the role of the fuel in the safety analysis and determine that the plan to qualify the fuel is adequate. (ML23277A139).

3. Summary of DOE Requirements, Orders, and Standards

DOE requirements related to fuel qualification and performance include:

- 10 CFR 830 defines Safety limits (SLs) as the limits on process variables associated with those safety class physical barriers, generally passive, that are necessary for the intended facility function and that are required to guard against the uncontrolled release of radioactive materials.
- DOE Standard (DOE-STD)-1271-2026, “Authorization Pathway for Nuclear Facilities,” contains high level guidance on fuel and safety limits (SLs). It encourages the use of qualified fuel.
 - For fuel that deviates from qualified designs and manufacturing processes, the standard states that fuel could be subject to a formal qualification process such as NUREG-2246 and monitored with fission breaks in real time with a condition that if primary coolant activity went above an acceptance threshold, the reactor would be shut down, until an assessment of the situation and cause determination was performed.
 - Qualified fuel subject to operating conditions outside of the qualified operating range is also subject to fission break monitoring
 - SLs for reactor facilities should be established to protect the integrity of the principle physical barrier that guards against an uncontrolled release of radioactive material. For reactor facilities, the standard states that this physical barrier is often the fuel cladding. SSCs protecting reactor facility SLs will be designated consistent with the specified safe harbor methodology.

4. Staff Considerations on NRC and DOE Requirements and Guidance

Leveraging Across NRC Licensing Frameworks

The NRC licensing frameworks contain substantially similar requirements and acceptable approaches with respect to fuel qualification and performance. However, the licensing stage influences *when* information demonstrating fuel qualification is needed.

For CP applications under Part 50 or Part 53 based on a preliminary design, fuel qualification need not be complete but should be complete for the OL application with the final design information. Similarly, COL applications require, and applications under proposed 10 CFR Part 57 would require, final design information demonstrating fuel qualification at the time of application. These considerations may affect the timing and purpose of the NRC staff’s use of DOE information.

Considerations on NRC and DOE Requirements

The DOE requirements are generally at a higher level than the NRC requirements. The NRC staff may be able to leverage a DOE review of a fuel design if NUREG-2246 is used in the DOE review

and if it clearly documented how each goal of the fuel qualification assessment framework described in NUREG-2246 has been met. However, even if the DOE review leveraged NUREG-2246, there are situations that may affect the ability of the NRC to use DOE review information. For example, it is important for the NRC staff to understand the role that the fuel plays in the overall safety case of the DOE-authorized reactor. If the DOE-authorized reactor design has many SSCs that would prevent or mitigate the release of radionuclides into the environment (i.e., significant defense-in-depth), then it is possible that the DOE may have required less justification for satisfying each goal of NUREG-2246 compared to if there were fewer such SSCs. As such, if there are fewer compensating safety features providing defense-in-depth in the reactor design that an applicant is requesting NRC licensing of compared to the DOE-authorized reactor, then the NRC staff may need to evaluate the fuel qualification in more detail. Similarly, if the DOE-authorized fuel operational envelope does not match the applicant's planned operational envelope in their NRC application, then the NRC staff will need to perform a more detailed review.

Unlike NRC requirements, DOE-STD-1271-2026 allows for the use of unqualified fuel or for fuel outside of its qualification range. If unqualified fuel is authorized by the DOE or if qualified fuel is authorized outside of its qualified range, the NRC may not be able to leverage significant portions of the DOE authorization. However, the NRC staff may be able to consider data gained from the operation of the fuel under the DOE authorization. If significant data is available that covers the fuel performance envelope, including transient conditions, then this data may support the qualification basis. If such data is used, the NRC should ensure that the pedigree of the data is acceptable.

Overall, the primary attributes that should be considered when the NRC staff is determining how much information related to DOE authorizations can be leveraged include:

- Similarity of fuel designs: Differences in fuel design, including manufacturing specifications and manufacturing techniques, can impact fuel performance. Larger-scale manufacturing for a commercial license application (compared to the small-scale manufacturing employed for a limited-scope DOE authorization) may present a challenge for meeting fuel design tolerances or as-manufactured defects, for example. Differences in the fuel designs operated in DOE-authorized reactors and those planned to be operated in NRC-licensed reactors should be evaluated.
- Similarity of fuel performance envelopes: Similarities between the fuel performance envelope for fuel in DOE-authorized reactors and planned NRC-licensed reactors allow for greater opportunities for leveraging. The envelope considers the environmental conditions and the radiation exposure under which the fuel is expected to perform during both normal operation and transients. Relevant factors may include temperature, pressure, power, burnup, exposure, and residence time. Differences in the performance envelopes should be evaluated. The NRC staff expects that differences are likely to occur in the maximum burnup and operating duration.
- Relevance of fuel performance in the overall reactor safety case / level of defense-in-depth in the design: The role that the fuel plays in the safety case in the DOE-authorized reactors and the planned NRC-licensed reactors should be considered. If certain safety systems or features of the reactor that limit the release of radionuclides in the case of an accident (e.g., a robust containment or other systems, structures, or components that are important to safety) were present in the DOE-authorized reactor, but not the planned NRC-licensed reactor, the NRC staff should evaluate potential increased confidence needed in fuel performance in order to establish a similar level of adequate protection of public health and safety.
- Surveillance/oversight: The NRC staff should examine if increased surveillance/oversight was leveraged in the DOE authorization to expand the performance envelope past its prior

qualified envelope or when there are deviations from qualified specifications. While fission break monitoring may be acceptable for short-duration, critical/physics testing experiments, use of such a system at the commercial scale could have safety implications. Other controls may also be in place at the DOE-authorized facility that will not be at the NRC-licensed facility that may affect the safety evaluation.

- Evaluation models (EMs): The ability to accurately analyze and predict the performance of fuel is an important part of demonstrating that a fuel design is qualified. The NRC staff should evaluate the EM's capability to predict the performance of the specific fuel type over the intended application range. As a part of this evaluation, the NRC staff should consider the quantity and applicability of the data underlying the validation basis of the constituent models and EM as a whole. See the "Analytical Methods Development, Verification, and Validation" appendix of this white paper for more details on this topic.
- Crediting data gained during DOE operation: Data gained on the fuel during the DOE-authorized operation may be credited as part of the fuel qualification basis presented to the NRC. This data may be useful, but the NRC should evaluate the similarity of the design (e.g., specifications) and the performance envelope of this data to that anticipated for NRC design application. The NRC staff should note that although this data is beneficial, it is likely to primarily be at steady-state operation and unless further testing was performed on the fuel, it may not be applicable to qualification during transients (i.e., the data envelope may not fully encompass the entire fuel performance envelope, which considers transient conditions).

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE fuel qualification and performance information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

This table outlines three of the key factors in determining the extent to which DOE authorization information related to fuel qualification and performance may be leveraged.

Table L.1: Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	The DOE and NRC regulatory frameworks share several foundational concepts in fuel qualification. In some cases, both frameworks may utilize the same guidance documents (e.g., NUREG-2246). There is a potential for leveraging DOE authorization information, but differences must be considered to ensure information or methodologies developed under DOE standards are appropriately adapted to NRC regulatory expectations. Implementation of NUREG-2246 may also vary based on several other factors described in this table, so the NRC staff should consider the level of justification used to demonstrate that the NRC guidance (e.g., each goal of NUREG-2246) is in line with NRC fuel qualification reviews.
Application quality and justification of relevance of DOE authorization to NRC review	To maximize potential leveraging of DOE information, applicants should provide a thorough and complete justification for their fuel qualification against NRC requirements.

Safety-significance of information leveraged	Fuel performance is a highly significant component of plant safety which will result in a greater focus of independent NRC review.
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NRC Staff Approach to Leveraging DOE Fuel Qualification and Performance Information

NRC staff should consider the similarity of designs, performance envelope, and defense-in-depth in the overall reactor design. On this basis, NRC leveraging of DOE fuel qualification and performance information can generally be organized into three categories:

Maximum leveraging

- Under maximum leveraging, DOE information and supporting engineering documentation supporting fuel qualification can be incorporated into an NRC application with minimal modification and can serve as the foundation for NRC review. To achieve this, the applicant must provide a clear technical justification against NRC requirements.
- NRC will tailor the review to the extent practical. Due to the high safety significance of this technical area and the differences in NRC and DOE scope of approval, a significant reduction in effort may be challenging.
- *Examples:* Maximum leveraging could be indicated if all of the following examples are true:
 - The applicant's fuel design matches the design DOE authorized
 - The applicant's fuel performance envelope matches the envelope that the DOE authorized, including burnup, fluence, lifetime, etc.
 - The role of the fuel in the safety case and need for additional measures to prevent or mitigate accidents in the applicant's reactor design matches the same characteristics of the design DOE authorized

Moderate leveraging

- Under moderate leveraging, the NRC staff would verify that the technical rationale and data supporting fuel qualification is clearly documented and provided to the NRC *and* would work with the applicant to fill in the areas of fuel qualification that may be not fully sufficiently. For example, the NRC staff may focus its review on how the applicant is using conservative modeling (e.g., additional added uncertainties) and/or limits in regimes where limited data exists. Thus, additional work would be required. DOE information provides a strong technical starting point but cannot serve as the licensing foundation without refinement.
- *Examples:* Moderate leveraging would be supported by, for example, any of the following:
 - The applicant's fuel performance envelope is slightly different than the envelope evaluated by the DOE.
 - The defense-in-depth in the applicant's overall reactor design is slightly different than the defense-in-depth attributes of the design DOE authorized.
 - Any data retrieved from DOE authorized operation that may be used to inform the qualification basis (including data from normal operation and transient testing).

Limited leveraging:

- Under limited leveraging, DOE information may serve only as background material for understanding the evolution of the fuel design and/or qualification envelope, rather than forming the basis of fuel qualification under NRC frameworks. Accordingly, the DOE information would inform the NRC's review but would not be a significant part of the technical

basis for the NRC's safety determination. No significant reduction in NRC effort is anticipated.

- *Examples:* Minimal leveraging would be indicated if any of the following examples are true:
 - The applicant's fuel design is substantially different than the design DOE authorized.
 - The applicant's fuel performance envelope contains significant differences compared to the envelope that the DOE authorized.
 - The defense-in-depth in the applicant's overall reactor design is much less than the defense-in-depth attributes of the design DOE authorized.
 - Any data retrieved from DOE authorized operation that is used to form the qualification basis includes data from only normal operation without any transient testing.

APPENDIX M—Analytical Methods Development, Verification, and Validation

Safety analysis is a key element of ensuring nuclear safety because accident progression and consequences in general can be determined only via analyses and/or simulations. These analyses inform important safety aspects of the design and are used to develop plant operating limits. Applicants conduct analyses using calculational frameworks for evaluating the behavior of physical systems (referred to here as analytical methods), which may include one or more computer programs or models used to represent physical phenomena. To have confidence in the results of an applicant's analyses, the NRC must ensure that these analytical methods were developed under appropriate quality standards, adequately documented, verified to perform numerically correct calculations, and validated against applicable experimental data and operating experience to demonstrate that they adequately predict physical responses using models that are applicable to the phenomena of interest for the analysis.

This appendix relates DOE and NRC requirements and guidance involving safety analysis, including requirements and guidance related to analytical method development, verification, and validation used to provide confidence in the results of the analysis.

1. NRC Regulatory Requirements

NRC requirements concerning safety analyses and associated analytical methods are included in various NRC regulations under Title 10 of the *Code of Federal Regulations* (10 CFR) for applicants for and holders of NRC permits, licenses, certifications, and approvals related to design, construction, and operation of nuclear reactors and utilization facilities.

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," provides requirements related to safety analysis, including:

- 10 CFR 50.2, which provides a definition for the term "basic component" that includes safety-related design and analysis. Based on this definition, issues (including errors) with some analyses may need to be reported to the NRC under 10 CFR Part 21, "Reporting of Defects and Noncompliance," and/or 10 CFR 50.55. In addition, the definition of "design bases" states that specific values or ranges of values chosen for controlling parameters as reference bounds for the design may be derived from safety analysis.
- 10 CFR 50.34 establishes that construction permit (CP) and operating license (OL) applications must analyze the design and performance of structures, systems, and components (SSCs) and their adequacy for the prevention of accidents and mitigation of the consequences of accidents.
 - As noted in 10 CFR 50.34(a)(4), a CP application is required to provide preliminary analyses and evaluations of the design and performance of SSCs. The NRC staff has interpreted this to mean that, except as required by other regulations, the analytical methods used to perform the analysis for a preliminary safety analysis report (PSAR) may also be preliminary and not subject to full verification and validation as would be expected for an OL application.
- For boiling or pressurized light-water nuclear power reactors fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding, 10 CFR 50.46 requires applicants and licensees to use or develop either (1) an emergency core cooling system (ECCS) evaluation model with justification showing that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident (LOCA), with comparisons to experimental data, or (2) an ECCS model consistent with 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

- 10 CFR Part 50 Appendix K defines required and acceptable features of ECCS evaluation models and required documentation referenced in 10 CFR 50.46. Section I of Appendix K provides prescriptive requirements for ECCS evaluation models developed in accordance with 10 CFR 50.46(a)(1)(ii) as an alternative to the best-estimate plus uncertainty model requirements of 10 CFR 50.46(a)(1)(i). Section II of Appendix K provides documentation requirements for evaluation models, as required by 10 CFR 50.46(a)(1)(i), including certain verification and validation activities.
 - Under the EO 14300 rulemaking effort, an alternative to the requirements of 10 CFR 50.46 is proposed in 10 CFR 50.46a (91 FR 44560). The proposed rule maintains the same basic requirements related to code validation as the existing rule.
- For applications for facilities to generate commercial power with nuclear reactor designs that differ significantly from light-water reactor designs or use simplified, inherent, passive, or other innovative means to accomplish their safety functions, 10 CFR 50.43(e) states Commission approval will only be granted if there is testing, analysis, or operating experience demonstrating the safety features of the design, including interdependent effects, and sufficient data exists on the safety features to assess analytical tools used for safety analysis; or if acceptable testing of a prototype plant has been performed over an appropriate range of conditions, including normal operations, transients, and specified accidents.
 - Recent rulemaking proposed in response to EO 14300 would clarify that it is explicitly allowed for applicants to incorporate information from facilities that have demonstrated the ability to operate safely under DOE or DOW authorization to satisfy the requirements of 10 CFR 50.43(e) (91 FR 16584).
- Applicants may perform a criticality analysis that complies with the requirements of 10 CFR 50.68 in lieu of compliance with 10 CFR 70.24.
 - Proposed rulemaking in response to EO 14300 proposes to revise 10 CFR 50.68 to allow flexibility in enrichment of uranium 235 of up to less than 20 percent by weight (91 FR 44560). This change does not otherwise affect the requirements for criticality analysis.
- 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” includes requirements concerning quality assurance. These requirements apply to all activities affecting the safety-related functions of SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. This includes design, fabrication, testing, operating, and other activities that are informed by safety analysis. Appendix B notes that “‘quality assurance’ comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service” – as discussed above, ensuring confidence in SSC performance is a key objective supported by safety analysis. To that end, Criterion III of Appendix B requires design control measures to be applied to “reactor physics, stress, thermal, hydraulic, and accident analyses.” Additional criteria of Appendix B are relevant to analytical methods, including requirements for documented instructions, corrective actions, document control, and records retention.

10 CFR Part 52 includes requirements for safety analyses to be provided for standard design certifications in 10 CFR 52.47, for combined licenses in 10 CFR 52.79, for standard design approvals in 10 CFR 52.137, and for manufacturing licenses in 10 CFR 52.157. More detailed requirements for the analytical methods typically refer back to the requirements in 10 CFR Part 50, including the quality assurance requirements of 10 CFR Part 50, Appendix B.

10 CFR Part 53 provides a number of requirements related to safety analysis, including:

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- 10 CFR 53.440(a)(1), which requires “[a]nalysis, appropriate test programs, prototype testing, operating experience, or a combination thereof” to demonstrate design features meet the functional design criteria, similar to the requirements of 10 CFR 50.43(e).
- 10 CFR 53.440(m), which includes criticality analysis requirements that may be met in lieu of compliance with 10 CFR 70.24, similar to 10 CFR 50.68.
- 10 CFR 53.450, which includes general analysis requirements, and 10 CFR 53.450(d) specifically requires analytical codes (including thermodynamics, reactor physics, fuel performance, and mechanistic source term codes) to be qualified for the range of conditions for which they are to be used. This requirement must be met for:
 - Standard design approvals (10 CFR 53.1209(b)(2))
 - Design certifications (10 CFR 53.1239(a)(19))
 - Manufacturing licenses (10 CFR 53.1279(a)(2))
 - Construction permits (10 CFR 53.1309(a)(2)), though design and analyses may be preliminary in accordance with 10 CFR 53.1309(a)(2)(ii) and the CP application may include research and development items (which may include code validation activities) in accordance with 10 CFR 53.1309(a)(2)(iii)
 - Operating licenses (10 CFR 53.1369(b))
 - Combined licenses (10 CFR 53.1416(a)(2))
- Various 10 CFR Part 53 requirements refer to the quality assurance requirements of 10 CFR Part 50, Appendix B.

Finally, proposed 10 CFR Part 57 would provide several requirements related to safety analyses, including:

- Proposed 10 CFR 57.25 would require applicants to demonstrate applicability, in part, of proposed 10 CFR Part 57 through analysis showing a dose criterion of 1 rem will be met.
- Proposed 10 CFR 57.60(a)(1)(iii) would require CP and OL applicants to provide the evaluation required by proposed 10 CFR 57.25(a) in the plant final safety analysis report (FSAR).
- Proposed 10 CFR 57.60(a)(1)(vii) would require analysis, testing, etc., similar to the requirements of 10 CFR 50.43(e) as discussed above.
- Proposed 10 CFR 57.60(a)(12) would require a description of the design-specific risk analysis methods applied to demonstrate adequate defense in depth and safety margin and the results of the analysis.
- Proposed 10 CFR 57.155(a) would require proposed 10 CFR 57.60(a)(1) to be met by ML applicants.

2. NRC Guidance and Related References

A key NRC guidance document related to analytical method development, verification, and validation is Regulatory Guide (RG) 1.203, “Transient and Accident Analysis Methods” (ML053500170). This document describes the evaluation model development and assessment process (EMDAP), a systematic process to demonstrate adequacy of analytical methods. The EMDAP encompasses the whole life cycle of development, including determination of requirements for the evaluation model, development of the assessment base and evaluation model itself, and

assessment of model adequacy. RG 1.203 also emphasizes the importance of quality assurance throughout the EMDAP and proper evaluation model documentation. RG 1.203 is a companion to section 15.0.2, "Review of Transient and Accident Analysis Methods," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) (ML070810350).

While RG 1.203 and SRP 15.0.2 focus on transient and accident analysis methods for LWRs, the overarching concepts of the EMDAP are broadly applicable to any analytical methods and are generally consistent with NRC staff expectations for analytical method development, verification, validation, and uncertainty quantification. However, more detailed guidance on specific topic areas is available:

- Additional guidance on code development, verification, and validation that is specific to best-estimate loss-of-coolant accident (LOCA) analyses is provided in RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance." (ML003739584) NRC exploration of best-estimate LOCA analysis methodologies in the 1980s that culminated in revisions to 10 CFR 50.46 and development of RG 1.157 also led to the development of the code scaling, applicability, and uncertainty (CSAU) evaluation methodology documented in NUREG/CR-5249, "Quantifying Reactor Safety Margins," (ML030380473) which was a predecessor of RG 1.203.
- Additional guidance on validation for nuclear criticality safety analysis is provided in NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Computational Methodology" (ML050250061). This document includes detailed guidance on topics relevant to compliance with 10 CFR 50.68, including the definition of a validation area of applicability and development of an upper subcriticality limit based on validation data. NUREG/CR-7311, "Determination of Bias and Bias Uncertainty for Criticality Safety Computational Methods," (ML25120A337) builds upon NUREG/CR-6698 and other existing guidance to provide recommendations in areas where guidance was limited.
- Additional guidance on development and validation of fuel performance models is provided in NUREG-2246, "Fuel Qualification for Advanced Reactors," (ML22063A131) which includes frameworks for assessing the fuel performance evaluation model as well as the experimental data used to qualify fuel and validate the fuel performance model. Additional discussion on fuel qualification is provided in Appendix L of this document.
- Source term and consequence analyses methodologies have not historically made use of the guidance in RG 1.203, though similar concepts have been applied in their development, validation, and verification. RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 1, (ML23082A305) provides guidance for accident source term evaluations, including considerations related to models and codes used for the analysis. RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, (ML003740205) provides guidance on the modeling of atmospheric dispersion of radionuclides in the analysis of accident dose consequences. Additional discussion on source term and consequence analysis is provided in Appendices E and F of this document.
- RG 1.168, "Verification, Validation, Reviews and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants,"⁹ (ML13073A210) provides guidance on implementation of software used in safety systems, which can include analytical methods. It

⁹ Note that the terms "verification" and "validation" as used when describing software development and quality assurance may not correspond to the same activities used in development of scientific computer simulations. Care should be taken not to confuse the subjects, which can appear in overlapping contexts.

endorses the Institute of Electrical and Electronics Engineers (IEEE) standards for software verification and validation and for software reviews and audits.

Applicants seeking NRC approval of analytical methodologies often submit topical reports providing details on these methods for NRC review and approval. As such, NRC review of these topics is often considered outside of reviews of requests for permits, licenses, certifications, and approvals. However, NRC content of application guidance underscores the need to validate, verify, and document analytical methods in applicable sections as discussed below:

- RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)” (ML011340122) – applicable to 10 CFR Part 50 applications
 - Chapter 4 provides guidance on demonstrating fuel performance, nuclear design, and thermal-hydraulic models are validated and verified.
 - Chapter 6 provides guidance on demonstrating analytical methods for mass and energy releases into containment resulting from LOCAs and secondary system pipe breaks are validated.
 - Chapter 15 provides guidance on demonstrating analytical methods for transient and accident analyses are validated and verified.
- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (ML070810350) – applicable to NRC staff reviews of 10 CFR Part 50 and 52 applications for LWRs
 - Chapter 4 provides NRC staff expectations for demonstrating the adequacy of fuel performance, nuclear design, and thermal-hydraulic analytical methods.
 - Chapter 6 provides NRC staff expectations for demonstrating the adequacy of analytical methods used to calculate mass and energy releases into containment.
 - Chapter 15 provides NRC staff expectations for demonstrating the adequacy of analytical methods for transients and accidents.
- NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors” (ML04240048) – applicable to 10 CFR Part 50 applications for non-power reactors (Class 104 reactors and Class 103 commercial non-power production and utilization facilities). Part 1 provides content of application guidance and Part 2 provides NRC staff review guidance.
 - Chapter 4 provides guidance for the reactor design and explicitly describes the need to discuss and justify analytical methods for neutronic and thermal-hydraulic analyses and estimate their accuracy.
 - Chapter 13 provides guidance on accident analysis and explicitly describes the need to describe computer codes, the basis for their validity as compared to experiments and other operating reactors and estimates of the accuracy of the codes.
- DNRL-ISG-2022-01, “Safety Review of Light-Water Power-Reactor Construction Permit Applications” (ML22189A099) – applicable to NRC staff reviews of 10 CFR Part 50 CP applications for advanced LWRs
 - Appendix A provides clarifications on NRC staff expectations relative to NUREG-0800 regarding the maturity of safety analysis methodologies at the CP stage.
- DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap” (ML23277A139) – applicable to NRC staff reviews of 10 CFR Part 50 and 52 applications for non-LWRs using the risk-informed, performance-based, and technology-inclusive Licensing Modernization Project (LMP) framework described RG 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors” (ML20091L698)

- Section 20 provides guidance on compliance with 10 CFR 50.43(e).
 - Appendix A provides recommendations for pre-application engagement on safety analysis methodologies and associated validation through the topical report process.
 - Appendix C provides NRC staff review guidance in the areas of fuel qualification and safety analysis methodologies for CP applications, which clarifies NRC staff expectations for the maturity of these areas at the CP stage.
- Draft report for comment NUREG-2271, “Guidelines for Preparing and Reviewing Applications Under 10 CFR Part 57” (ML26111A077) is applicable to applications that would be submitted under proposed 10 CFR Part 57 and provides the following guidance relevant to analytical methods.
 - Chapter 4 provides guidance stating analytical methods used for nuclear and thermal-hydraulic design should be described and their validity should be justified based on experiments or predictions of operating reactors. Estimates of the accuracy of the analytical methods should be provided.
 - Chapter 12 provides guidance related to accident analyses, including methods used to analyze design basis accidents and severe accidents. Note that draft NUREG-2271 uses the term “accident analysis methodology” to include not just the computational models and methods associated with accident analysis (which is the focus of this Appendix), but the entire spectrum of activities from hazard analysis and initiating event identification approaches through to accident consequence analysis.

3. Summary of DOE Requirements, Orders, and Standards

10 CFR Part 830, “Nuclear Safety Management,” provides DOE requirements for nuclear facility safety.

- § 830.202, “Safety basis,” defines the requirements for safety basis documentation, which includes an identification and analysis of the hazards, categorization of the facility, and preparation of documented safety analysis (DSA).
- § 830.204, “Documented safety analysis,” requires the DSA to evaluate normal, abnormal, and accident conditions, including consideration of the need for analysis of beyond design basis accidents. § 830.206 provides requirements related to the preliminary documented safety analysis, which must document the nuclear safety design criteria to be satisfied.
- Table 1 of Appendix A to Subpart B of Part 830 provides acceptable methodologies for preparing a DSA. For DOE reactors, the table references RG 1.70 or successor documents. For nonreactor nuclear facilities, the table references DOE-STD-3009.

The NRC staff reviewed DOE-STD-1271-2026, “Authorization Pathway for Nuclear Facilities,” to identify DOE orders and standards relevant to safety analysis that are referenced as part of the DOE authorization process. It is important to note that DOE Orders (DOE O) and DOE Policies (DOE P) are supplements to DOE regulations and the contractors must meet the requirements. DOE standards are similar to NRC regulatory guides and standard review plans; however, they are not optional in the sense NRC RGs are.

DOE orders relevant to safety analysis include:

- NE O 414.1, “Quality Assurance,” Attachment 2 Part D requires quality assurance (QA) for software that performs a nuclear safety function or design verification. Quality assurance is required to follow appropriate industry standards.
- NE O 420.1, “Facility Safety,” establishes programmatic and safety requirements for nuclear facilities authorized by the DOE Office of Nuclear Energy. Attachment 2, Chapter I, Section

3.a requires safety analyses to be used to identify SSCs needed to fulfil safety functions, the functional requirements of these SSCs, and administrative controls needed to fulfil safety functions.

- NE O 458.1A, “Radiation Protection of the Public and the Environment,” Section 4.d provides a requirement that:
 - “Models or other information for dose calculations must be appropriate for their purpose. Dose evaluation models that are codified or approved for use by regulators of DOE or by DOE may be used where applicable. Alternatives to such codified or approved dose evaluation models to be used for demonstrating compliance with this Order must be approved by the SBAA [safety basis approval authority].”

DOE standards relevant to safety analysis include:

- DOE-STD-1271-2026, “Authorization Pathway for Nuclear Facilities,” provides the process by which DOE authorizes nuclear reactors. Section 7.2.2 states that the PDSA should include a justification of the analysis tools and methods to be used for confirming adequacy of the design and safety analysis. Section 7.3 states that the final design, as documented by the DSA, must include final system performance analyses demonstrating safety functions are satisfied.
- NE-STD-1020-2025, “Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities,” establishes natural phenomena hazard (NPH) design category (NDC) based on unmitigated consequence analysis performed using DOE-STD-3009-2014 (see below). Section 10.2 of this standard also provides additional guidance on quality assurance for “software used in the analysis and design of SSCs,” and specifies computer programs should be validated and verified. Commercial grade dedication is required for software not developed under an approved QA program.
- DOE-STD-3009-2014, “Preparation of Nonreactor Nuclear Facility Documented Safety Analysis,” provides the DOE standard for contents of a DSA. Section 3.2 of this document provides a discussion on accident analysis. Since DOE-STD-3009-2014 is intended for nonreactor facilities, it focuses on source term and consequence analysis.
- DOE-STD-3007-2017, “Preparing Criticality Safety Evaluations at DOE Nonreactor Nuclear Facilities,” provides the DOE standard for criticality safety evaluations.
 - Section 3.4 provides requirements concerning the methodology used to establish criticality limits and the validation thereof. This standard includes references to and encourages use of consensus standards in the ANSI/ANS 8 series (nuclear criticality safety) but does not require the use of such standards.
- DOE-STD-1104-2016, “Review and Approval of Nuclear Facility Safety Basis and Safety Design Basis Documents,” section 4.3, “Hazard and Accident Analyses,” provides the following guidance related to DOE’s review of accident analysis methodologies:
 - “The accident analysis methodology is clearly identified and appropriate, including identification of initial conditions and assumptions. The technical basis for source term values is provided, valid, and appropriate for the physical situation being analyzed. The completeness and level of detail in the technical basis should increase as the parameters depart from the default or bounding values described in Part 830’s safe harbor methods. Supporting calculations and technical documents are identified, where appropriate, and reviewed for critical aspects of safety controls, where appropriate.”

4. Staff Considerations on NRC and DOE Requirements and Guidance

This section contains high-level NRC staff observations on NRC and DOE requirements and guidance and how DOE requirements and standards may be used for various NRC licensing frameworks with respect to analytical methods development, verification, and validation.

Leveraging Across NRC Licensing Frameworks

The NRC licensing frameworks contain substantially similar requirements and acceptable approaches with respect to analytical methods development, verification, and validation, such that there are not significant differences between NRC frameworks in how DOE information may be leveraged.

However, the NRC staff notes that the licensing stage influences the point at which applicants need to demonstrate final analytical methods verification and validation. For CP applications under Part 50 or Part 53 based on a preliminary design, analytical methods verification and validation need not be complete at the CP stage but should be complete for the OL application with the final design information. An exception is that LWRs are required to demonstrate verification and validation of LOCA EMs even at the CP stage, but this does not apply for other transients. COL applications require, and joint applications for CPs and OLs under proposed 10 CFR Part 57 would require, final design information demonstrating adequate analytical methods verification and validation at the time of application. These considerations may affect the timing and purpose of the NRC staff's use of DOE information.

Considerations on NRC and DOE Requirements and Guidance

NRC and DOE requirements, guidance, and standards reflect a shared understanding of the need to provide confidence in analytical methods through appropriate development processes, documentation, quality assurance, verification, and validation. However, while the NRC provides detailed guidance on how this may be accomplished for several different technical disciplines, DOE requirements are at a higher level.

Consistent with use cases focused primarily on short-term technology demonstrations, higher-level requirements in DOE's process maintain the flexibility to authorize facilities with limited verification, validation, or documentation of some analytical methods. Such flexibility is needed for some designs and in some technical disciplines where validation data is difficult to obtain from legacy test facilities and reactors. Gaps in validation data would result in lower confidence and increased uncertainty in the analysis results. This reduced confidence and increased uncertainty is potentially compensated for in the DOE process with conservatism in analysis inputs and more stringent controls on reactor operation, over a relatively limited lifetime that helps reduce risk. Additional measures may also be imposed by DOE to mitigate risk, including additional safety features not intended to be used in later commercial applications.

Consistent with the primarily longer-term commercial operation use cases, NRC requirements and guidance are intended to provide confidence in the analytical methods sufficient to demonstrate safe operation within a large envelope over an extended period of time. While the NRC has some tools available to constrain reactors to operate within a specified envelope (e.g., the prototype plant requirements of 10 CFR 50.43(e)(2)), these are likely not practical for larger gaps in the validation database or for long-term commercial operation.

While it is possible that applicants may approach DOE with mature analytical methods that have been developed, verified, validated, and documented in a manner that is consistent with NRC guidance, this is not required by DOE guidance. The NRC staff finds it more likely that DOE will authorize facilities based on less mature analytical methods or analytical methods with gaps in their assessment databases, with appropriate measures to ensure safety during the period of the DOE authorization. Operation of the authorized facility would then generate data demonstrating performance of the reactor and its safety systems, which could be used to inform analytical methods later used to support NRC licensing.

Information useful to analytical methods development and validation that could be gained from a DOE authorized facility includes (but is not limited to):

- Critical experiment results
- Reactor startup physics test results
- Data on steady-state reactor operation, including pressures, temperatures, power levels, fluxes, critical control rod positions, coolant poison concentrations, etc.
- Data on plant transients similar to the above for steady-state, but including safety system performance (e.g., control rod drop times, peak transient power, flow rates through passive decay heat removal systems)
- Fuel performance data, including post-irradiation examinations
- Data on material performance under irradiation and other environmental factors

Use of DOE standards for NRC licensing

To maximize the information useful for an NRC review, applicants should provide information on the DOE's review of their analytical methods, including any gaps identified and how these gaps were resolved in the DOE's authorization (e.g., with additional conservatism in analysis, limits on operation, added safety features). Applicants should demonstrate the validity of their analytical methods using, in part, data obtained from operation under the DOE authorization. For maximum overlap in approaches in the area of analytical methods, applicants should be encouraged to develop and document analytical methods following the general guidance of RG 1.203, or other more specific guidance for other topic areas as discussed above. Applicants should also be encouraged to engage in pre-application discussions with the NRC on analytical methods, up to and including submittal of topical reports on analytical methods prior to submittal of requests for NRC licensing.

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE analytical methods development, verification, and validation information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

Consistent with the discussion in the main body of this white paper, this table outlines three of the key factors in determining the extent to which DOE authorization information related to analytical methods development, verification, and validation may be leveraged.

Table 1: Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	NRC requirements and guidance are significantly more detailed for analytical methods than DOE requirements and guidance. Analytical methods used by DOE authorized facilities may need enhancements in development, verification, validation, documentation, or quality assurance for use in later licensing stages in the NRC's process (e.g., OL, COL).
Application quality and justification of relevance of DOE authorization to NRC review	The maturity of the analytical methods is the most significant factor in determining how relevant DOE authorization will be to the NRC review. Less mature analytical methods with more gaps will indicate a need for an applicant or DOE to apply larger conservatisms in the analysis, tighter control of operation, or additional safety features. This would make validation data generated during DOE authorization more important for leveraging in NRC licensing, but care should be taken to ensure the conditions of the DOE authorized facility reflect the intended NRC application. More mature analytical methods would be expected to have fewer gaps in validation, documentation, etc., and could support operation closer to that of a commercial reactor.
Safety-significance of information leveraged	Analytical methods are one of the most important factors used in determining safety significance. However, not all analytical methods are equally safety-significant in and of themselves, with some analytical methods playing a larger role in a facility's safety case than others. Applicants should be able to explain the role of the various analytical methods in demonstrating the performance of plant safety systems and how they interact with one another for NRC to gain an understanding of their safety significance.

NRC Staff Approach to Leveraging DOE Analytical Methods Development, Verification, and Validation Information

NRC staff should consider the similarity of the approach used to develop, verify, validate, document, and provide quality assurance for analytical methods. On this basis, NRC leveraging of DOE analytical methods information can generally be organized into three categories:

Maximum leveraging:

- Under a maximum leveraging scenario, NRC staff verifies that the analytical methods used in the DOE application were previously approved by the NRC and were used consistent with that prior NRC approval. While this may be possible for certain codes or methods, the NRC staff notes it is unlikely that all analytical methods used in a DOE application were previously approved by the NRC.
- NRC verifies that appropriate QA processes were followed for development or dedication of analytical methods.
- *Example:* An example of maximum leveraging is one where an applicant uses an analytical method approved for use via an NRC topical report in a manner consistent with its NRC approval.

Moderate leveraging:

- NRC staff reviews the development, verification, and validation of analytical methods consistent with NRC guidance. If DOE review identified gaps in the validation basis, NRC staff focuses on those gaps in the review but ensures DOE's conclusions on other aspects of the analytical method are adequate for NRC applications. NRC staff identifies whether data obtained during operation under DOE authorization support conclusions regarding validation in the DOE documentation and are useful to fill any identified gaps in the validation basis.
- *Example:* An example of moderate leveraging is one where a DOE applicant relied on NRC guidance (e.g., RG 1.203) to develop, assess, and document analytical methods. A similar example would be one where an applicant uses an analytical method that is commonly used in similar applications but may not have NRC staff approval. The applicant provides an assessment of the analytical method against data from the DOE authorized facility, consistent with NRC guidance.

Limited leveraging:

- NRC staff performs a full review of analytical methods following guidance detailed above. If available, staff makes use of data obtained during operation under DOE authorization and identifies whether it supports validation of the analytical method.
- *Examples:* Examples of limited leveraging include:
 - An applicant uses an analytical method that does not have a known pedigree or qualification basis.
 - An applicant does not provide justification for analytical method adequacy.
 - An applicant uses an approach for developing, assessing, and documenting analytical methods that is inconsistent with NRC guidance.

APPENDIX N—Materials Performance

Materials performance in this appendix refers to ensuring structural materials for passive mechanical SSCs (e.g. vessels, piping) will perform safety functions and fulfill design requirements over the intended design life. This generally involves a consideration of relevant environmental effects (e.g. corrosion, creep, embrittlement) on materials and may be supported by appropriate performance monitoring programs to monitor and manage degradation before safety functions are impacted.

1. NRC Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” provides regulations for licensing production and utilization facilities, including these excerpts associated with materials performance:

- 10 CFR 50.34(a)(3)(iii) requires information relative to materials of construction.
- 10 CFR 50.55a requires, for light-water reactors (LWRs), that certain groupings of components meet *American Society of Mechanical Engineers* (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division (Div.) 1, which defines the allowable materials.
- 10 CFR 50.55a has requirements, for LWRs, for inservice inspection, preservice inspection, and inservice testing.
- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” contains the general design criteria (GDC) that establish the minimum requirements for the principal design criteria (PDC) for water-cooled nuclear power plants. Appendix A also establishes that the GDC are generally applicable to other types of nuclear power units and are intended to provide guidance in determining the PDC for such other units.
 - GDC 1 states, in part, that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions being performed.
 - GDC 4 states, in part, that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.
 - GDCs 14, 15, 30, 31, and 32 relate, in part, to material performance for the reactor coolant pressure boundary (RCPB).
- 10 CFR Part 50 Appendix G has fracture toughness requirements for ferritic RCPB materials.

10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” governs the issuance of early site permits (ESPs), standard design certifications (DCs), combined licenses (COLs), standard design approvals (SDAs), and manufacturing licenses (MLs) for nuclear power facilities.

- 10 CFR 52.79(a)(11), 10 CFR 52.47, and 10 CFR 52.137 require information relative to materials of construction for COLs, DCs, and SDAs, respectively. These requirements refer to 10 CFR 50 Appendix A for the PDC.
- 10 CFR 50.55a requires, for LWRs, that certain groupings of components meet ASME BPVC Section III, Div. 1, which defines the allowable materials.

10 CFR Part 53, “Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants,” provides a technology-inclusive, performance-based framework for ESPs, CPs, COLs, DCs, SDAs, and MLs for commercial nuclear plants.

- 10 CFR 53.440(c) requires that materials used for each safety-related (SR) and non-safety-related but safety-significant (NSRSS) SSC be qualified for their service conditions.

- 10 CFR 53.440(d) requires degradation mechanisms that may affect SR and NSRSS SSCs be evaluated and inform the design and integrity assessment programs.
- 10 CFR 53.460 has requirements for safety categorization and special treatments.
- 10 CFR 53.880 has requirements for inservice inspection, preservice inspection, and inservice testing.

Proposed 10 CFR Part 57, "Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles," would include provisions for CPs, OLs, MLs, SDAs, and provisions for affording regulatory finality to nuclear plant designs and essentially complete standardized operational programs.

- Proposed 10 CFR 57.30(c) would require a design to provide for the protection of engineered fission product barriers.
- Proposed 10 CFR 57.60(a)(1)(ii) would require a final safety analysis report (FSAR) that includes a description of the safety features and barriers.
- Proposed 10 CFR 57.60(a)(6) would require the FSAR to include a description of an evaluation of safety-related SCCs, sufficient to permit understanding of the design and relationship to safety analyses.
- Proposed 10 CFR 57.60(a)(8)(iii) would require the FSAR to include information related to operational programs with implementation milestones, including plans for maintenance, surveillance, and periodic testing.
- Proposed 10 CFR 57.60(a)(8)(xii) would require the FSAR to include programs and their implementation, necessary to ensure that the systems and components meet the requirements of identified codes and standards.

2. NRC Guidance and Related References

Light-Water Reactor Technologies

Examples of key guidance documents related to material performance include:

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (ML070660036). Numerous sections in the following chapters include information on reviews of materials selection, environmental compatibility, and performance monitoring programs:
 - Chapter 3: Design of Structures, Components, Equipment, and Systems
 - Chapter 4: Reactor
 - Chapter 5: Reactor Coolant System and Connected Systems
 - Chapter 6: Engineered Safety Features
 - Chapter 9: Auxiliary Systems
 - Chapter 10: Steam and Power Conversion Systems
- RG 1.26, Revision (Rev.) 6, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants" (ML21232A142).
- RG 1.44, Rev. 1, "Control of the Processing and Use of Sensitized Stainless Steel" (ML101680225).
- RG 1.54, Rev. 3, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants" (ML17031A288).
- RG 1.84, Rev. 40, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III" (ML25079A307).
- RG 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials" (ML003740284).
- RG 1.147, Rev. 21, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1" (ML23291A003).

Non-Light-Water Reactor Technologies:

Examples of key guidance documents related to material performance include:

- RG 1.87, Rev. 3, "Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors'" (ML25176A084).
- RG 1.246, Rev. 0, "Acceptability of ASME Code, Section XI, Division 2, 'Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,' for Non-Light Water Reactors" (ML22061A244).
- RG 1.232, Rev. 0, "Developing Principal Design Criteria for Non-Light Water Reactors" (ML17325A611) provides guidance for applying the GDC for LWRs to non-LWRs. The design criteria are provided for three categories of non-LWRs: a technology-inclusive class of advanced reactors (ARDC), sodium-cooled fast reactors (SFR-DC), and modular high-temperature gas-cooled reactors (MHTGR-DC).
 - For the design criteria related to materials selection, environmental compatibility, and performance monitoring (principally criteria 1, 4, 14, 15, 30, 31, and 32), RG 1.232 determines that the LWR GDC are directly applicable or applicable with adaptations.
- "Draft Interim Staff Guidance: Material Compatibility for Non-Light Water-Reactors," DANU-ISG-2023-01, Revision 1, (91 FR 59813; September 21, 2026).

3. Summary of DOE Requirements, Orders, and Standards

Key aspects of the DOE framework related to materials performance include:

- 10 CFR 830.202, "Safety basis," requires the contractor for a Hazard Category 1, 2 and 3 nuclear facility to categorize the facility consistent with DOE-STD-1027-92.
- 10 CFR 830.206, "Preliminary documented safety analysis," requires a preliminary documented safety analysis (PDSA) prior to construction and DOE approval of the PDSA before the contractor can procure materials or components or begin construction.
 - DOE may authorize the contractor to perform limited procurement and construction activities without approval of a PDSA if DOE determines that the activities are not detrimental to public health and safety and are in the best interests of DOE.
- DOE Office of Nuclear Energy (NE) Order (O) 420.1D, "Facility Safety," Attachment 3, "Design Criteria for Safety Structures, Systems, And Components," establishes requirements for the design and construction of safety SSCs, both safety class and safety significant, by identifying an applicable set of industry codes and standards, as well as DOE design criteria, standards, and directives.
- NE O 420.1D Attachment 3(3)(a)(7) requires a quality assurance program must be established that satisfies 10 CFR Part 830, Subpart A, "Quality Assurance Requirements," early in the project, such that safety SSCs and their associated support systems are designed, procured, fabricated, erected, and tested to standards and quality requirements commensurate with their importance to safety.
- NE O 420.1D Attachment 3(b) requires the selection and use of an appropriate set of applicable codes and standards and establishes design criteria to provide assurance that the SSCs are designed to reliably perform their intended functions.
 - The DOE technical standards and industry codes and standards, which are widely used for nuclear facility design and construction, must be evaluated for applicability. Once a code or standard is identified as applicable, the applicable requirements (i.e., "mandatory statements" in DOE terminology) must be applied in the design unless an equivalent approach is documented in the Code of Record and approved by the Safety Basis Approval Authority.

- NE O 433.1C, “Maintenance Management Program for DOE Nuclear Facilities,” requires maintenance of all SSCs that are part of the safety basis in compliance with a Nuclear Maintenance Management Program through a graded approach. The order includes a requirement for facility condition inspection but does not include specific discussion related to inservice inspection or non-destructive examination to assess component integrity relative to design criteria.
- DOE-STD-1271-2026, “Authorization Pathway for Nuclear Facilities,” Table 1, “Selection and Content Guide for Nuclear Safety Design Agreement,” describes how the Code of Record at the Nuclear Safety Design Agreement (NSDA) phase should detail the applicability of industry codes and standards to the design at a system level and address the need for variances, equivalencies, and exemptions to the extent known.

4. Staff Considerations on NRC and DOE Requirements and Guidance

Leveraging Across NRC Licensing Frameworks

The NRC staff notes that all NRC licensing frameworks require information related to materials of construction and will include design criteria addressing environmental compatibility (note Parts 50 and 52 require, and proposed 10 CFR Part 57 would require PDC, while Part 53 uses functional design criteria). Therefore, the NRC staff notes that the NRC licensing frameworks contain substantially similar requirements and acceptable approaches with respect to materials performance, such that there would not be significant differences between NRC frameworks in how DOE information is leveraged.

Considerations on NRC and DOE Requirements and Guidance

Both DOE and NRC require applicants to identify materials of construction and codes and standards as part of the safety analysis report. DOE and NRC guidance with respect to mechanical system codes and standards contain some differences between endorsed codes and standards for safety-related SSCs for NRC and specified codes and standards under DOE Order 420.1 for safety class and safety significant components. In addition, DOE Order 420.1 contains requirements similar to NRC requirements in GDC/PDC 1 for standards and quality requirements commensurate with their importance to safety.

In general, commensurate with applications for longer-term operations (e.g., 40-year license), NRC frameworks include comprehensive requirements and guidance related to environmental compatibility and performance monitoring (e.g., RG 1.246 and DANU-ISG-2023-01). As noted in the main body, DOE authorizations are typically for short-duration testing or demonstration purposes for which long-term materials degradation is not a significant consideration. Accordingly, DOE requirements and guidance are more limited in this area consistent with the primary mission and purpose of DOE-authorized facilities. This is a key difference in the DOE and NRC frameworks that directly ties to materials performance and will limit the degree of leveraging in this area. A DOE-authorized applicant could support leveraging by engaging with the NRC early in the DOE review and considering during design and construction long-term materials performance for safety-significant SSCs, including environmental compatibility and performance monitoring as appropriate.

Use of DOE standards for NRC licensing

As noted above, DOE requirements and guidance are more limited in the area of long-term materials performance, including environmental compatibility and performance monitoring. Therefore, there

are limited DOE standards available to be leveraged for NRC licensing in this topical area. Mechanical system codes and standards are addressed separately in the codes and standards appendix.

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE materials performance information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

Consistent with the discussion in the main body of this white paper, this table outlines three of the key factors in determining the extent to which DOE authorization information related to materials performance may be leveraged.

Table N.1: Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	For long-term materials performance, including environmental compatibility and performance monitoring, NRC requirements and guidance are more comprehensive than in the DOE framework. Therefore, the NRC staff expects limited ability to leverage from prior DOE authorization unless the DOE-authorized facility seeks to follow NRC requirements and guidance through the DOE authorization process. Pre-application technical engagement with NRC could enhance the efficiency and success of a leveraging review.
Application quality and justification of relevance of DOE authorization to NRC review	Important factors to clearly describe and justify the relevance of the DOE authorization for NRC licensing include identification of materials, environmental conditions, use of codes and standards, an environmental compatibility assessment, and performance monitoring. Draft Rev. 1 to DANU-ISG-2023-01 (ML26155A114) is currently available for public comment and, if approved, will provide additional information on acceptable approaches in this area (Docket ID NRC-2022-0215).
Safety-significance of information leveraged	There is greater potential for leveraging in the materials performance area for SSCs of lower safety significance since these SSCs are not subject to more rigorous regulatory treatments. Therefore, a greater degree of leveraging may be possible for NSRST or NSRSS SSCs than SR SSCs.

NRC Staff Approach to Leveraging DOE Materials Performance Information

NRC staff should consider the adequacy of materials selection, environmental compatibility assessment, and performance monitoring approaches for long-lived passive SSCs. On this basis, NRC leveraging of DOE materials performance information can generally be organized into three categories:

Maximum leveraging:

- Under a maximum leveraging scenario where the NRC staff has familiarity with the approaches being taken and the DOE authorization is following NRC requirements and

guidance, the NRC licensing review may be able to significantly leverage the prior DOE authorization and focus detailed review on selected areas to verify the adequacy of the applicant's approaches and the basis for performance monitoring approaches to ensure long-term component reliability.

- *Example:* An example of maximum leveraging for materials performance involves the DOE authorization following NRC requirements and guidance with respect to long-term materials performance, environmental compatibility and performance monitoring, through the DOE authorization process. The applicant would engage early with NRC staff on specific materials, environmental conditions, codes and standards, and performance monitoring approaches to adequately satisfy NRC requirements and guidance.

Moderate leveraging:

- Under a moderate leveraging scenario, the NRC staff may be able to leverage some aspects of materials performance and short-term environmental compatibility. Given limited DOE requirements and guidance and the generally short-term duration of DOE authorizations, the NRC staff review would likely focus on long-term materials performance and the plan to manage degradation that may impact safety functions due to extended time at elevated temperatures, exposure to corrosive environments, and irradiation.
- *Example:* A moderate leveraging example would be one where the DOE authorization addresses materials, environmental conditions, codes and standards, and performance monitoring requirements inconsistently with some areas addressed clearly to satisfy the corresponding NRC requirements and guidance, while others are not clearly addressed.

Limited leveraging:

- A limited leveraging example would be for the DOE-authorized facility to only address DOE requirements for specifying design codes and a nuclear maintenance management program without consideration of the more comprehensive NRC requirements and guidance. In this scenario, a thorough independent NRC review on these topics would be performed with the DOE authorization documentation primarily as relevant background material; the applicant would need to provide a substantially new technical basis addressing long-term materials performance, environmental compatibility, and performance monitoring.

APPENDIX O—Codes and Standards

1. NRC Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” provides regulations for licensing production and utilization facilities, including the following associated with codes and standards (C&S):

- 10 CFR 50.55a “Codes and Standards” incorporates by reference The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Section XI, Division 1 “Rules for Inservice Inspection of Nuclear Power Plant Components;” and the ASME Code for Operation and Maintenance of Nuclear Power Plants. This regulation mandates that systems and components of boiling water reactors (BWRs) and pressurized water reactors (PWRs) that are licensed under 10 CFR Part 50 meet these codes and provide conditions on their use.
- 10 CFR Part 50, Appendix A, “General Design Criteria [GDC] for Nuclear Power Plants,” GDC 1, requires, in part, that structures systems and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Where generally recognized C&S are used, GDC 1 requires that they be identified and evaluated to determine their applicability, adequacy, and sufficiency and be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” provides regulations for issuing early site permits, standard design certifications, combined licenses (COLs), standard design approvals, and manufacturing licenses (MLs) for nuclear facilities. The requirements in 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 1 are also applicable to light water reactors (LWRs) licensed under 10 CFR Part 52.

10 CFR Part 53, “Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants,” provides an optional technology-inclusive, performance-based framework for licenses, permits, certifications, and approvals for commercial nuclear plants. Requirements pertaining to C&S include:

- 10 CFR 53.440(b) requires that design features classified as safety related (SR) be designed using generally accepted consensus C&S.
- 10 CFR 53.880 requires each holder of an OL or COL to develop, implement, and maintain a program for inservice inspection (ISI) and inservice testing (IST) prior to receiving an OL or COL. The ISI and IST programs must, wherever applicable, be in accordance with generally accepted consensus C&S that have been endorsed or otherwise found acceptable by the NRC.

Proposed 10 CFR Part 57, “Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles,” would include the following provisions related to C&S:

- Proposed 10 CFR 57.60(a)(8)(xi) would require the final safety analysis report (FSAR) to include programs and their implementation, necessary to ensure that the systems and components meet the requirements of identified C&Ss.

- Proposed 10 CFR 57.60(a)(9) would require the applicant to provide information related to the C&S used to design the facility.
- Proposed 10 CFR 57.160(d)(1) would require an ML application to include a description of the relevant C&S used in the procurement, fabrication, and assembly of components comprising the manufactured reactor.

2. NRC Guidance and Related References

Examples of NRC guidance documents related to C&S include:

- RG 1.232, Revision 0, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors” (ML17325A611), describes the NRC’s proposed guidance on how the GDC in 10 CFR Part 50, Appendix A, may be adapted for non-light water reactors (non-LWR) designs. RG 1.232 provides guidance that is similar to that established under 10 CFR Part 50, Appendix A, Criterion 1, which sets forth requirements for use of C&S commensurate with their contribution to safety.
- RG 1.100, Revision 4, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants” (ML19312C677), describes methods that the NRC staff considers acceptable for use in the seismic qualification of electrical and active mechanical equipment and the functional qualification of active mechanical equipment for nuclear power plants. It endorses, with exceptions and clarifications, Institute of Electrical and Electronics Engineers (IEEE) Standard (Std) 344-2013, “IEEE Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations,” IEEE Std C37.98-2013, “IEEE Standard for Seismic Qualification Testing of Protective Relays and Auxiliaries for Nuclear Facilities,” and ASME QME-1-2017 “Qualification of Active Mechanical Equipment Used in Nuclear Facilities.”

Mechanical Systems

- RG 1.26, Revision 6, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” (ML21232A142) describes a quality classification system related to specified national standards that may be used to determine quality standards for satisfying 10 CFR 50, Appendix A, GDC 1 for components containing water, steam, or radioactive material in LWRs.
- RG 1.87, Revision 3, “Acceptability of ASME Code, Section III, Division 5, ‘High Temperature Reactors’,” (ML25176A084) describes an approach that is acceptable to the NRC staff of the NRC to assure the mechanical and structural integrity of components that operate in elevated temperature environments. It endorses, with exceptions, the 2023 Edition of ASME BPVC, Section III, “Rules for Construction of Nuclear Facility Components,” Division 5, “High Temperature Reactors.”
- RG 1.220, Revision 0, “Acceptability of ASME OM-2 Code, Component Testing Requirements at Nuclear Facilities,” (ML25329A088) describes an approach that is acceptable to the NRC staff for the development and implementation of an Inservice Testing (IST) Program for nuclear facilities. It endorses, with a regulatory position, the 2024 Edition of ASME OM-2 Code, “Component Testing Requirements at Nuclear Facilities”.
- RG 1.246, Revision 0, “Acceptability of ASME Code, Section XI, Division 2, ‘Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,’ For Non-Light Water Reactors,” (ML22061A244) describes an approach that is acceptable to the NRC staff for the development and implementation of a preservice and ISI program for non-LWRs. It endorses, with exceptions, the 2019 Edition of ASME BPVC, Section XI, “Rules for

Inservice Inspection of Nuclear Power Plant Components,” Division 2, “Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants.”

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” section 3.9.3, “ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures,” (ML14043A231) and section 3.9.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and their Associated Supports,” (ML14042A513) provide expectations for review of pressure-retaining components and piping systems designed in accordance with ASME BPVC Section III, Division 1.
- Draft DANU ISG 2023-01 Revision 1, “Material Compatibility for non-Light Water Reactors” (ML26155A114) identifies areas of staff review for a submittal seeking to use materials allowed under ASME BPVC, Section III, Division 5, or under other commercial C&S used for the design and construction of mechanical components.
 - Appendix A to this draft ISG, “Content of Applications for Materials Performance for Advanced Non-Light-Water Reactors,” describes information the NRC staff generally has found necessary to perform its review of materials of construction and passive SSC performance for non-LWRs, including the use of C&S.

Electrical Systems

- RG 1.32, Revision 4, “Criteria for Power Systems for Nuclear Power Plants,” (ML24306A036) provides acceptable methods for complying with regulations regarding the design, operation, and testing of SR electrical systems in nuclear facilities. It endorses, with exceptions, IEEE Std. 308-2020, “IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations.”
- RG 1.53, Revision 2, “Application of the Single-Failure Criterion to Safety Systems,” (ML033220006) provides acceptable methods for satisfying regulations regarding the application of the single-failure criterion to the electrical power, instrumentation, and control portions of nuclear power plant safety systems. It endorses IEEE Std 379-2000, “Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems.”
- RG 1.75, Revision 3, “Criteria for Independence of Electrical Safety Systems,” (ML043630448) provides the acceptable methods for complying with regulations regarding the physical independence and electrical isolation of SR (Class 1E) circuits and equipment. It endorses, with exceptions, IEEE-Std 384-1992 “Standard Criteria for Independence of Class 1E Equipment and Circuits.”
- RG 1.89, Revision 2, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” (ML22272A602) describes an approach that is acceptable to meet regulatory requirements for environmental qualification of certain electric equipment important to safety for nuclear power plants. It endorses, with exceptions, International Electrotechnical Commission (IEC)/IEEE 60780-323 “Nuclear Facilities—Electrical Equipment Important to Safety—Qualification.”

Structures

- RG 1.57, Revision 2, “Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components,” (ML12325A043) describes an approach that the NRC staff considers acceptable for use in designing metal primary reactor containment system components, including an endorsement of portions of ASME Code. The RG also provides methods for demonstrating containment structural integrity.
- RG 1.136, Revision 4, “Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments,” (ML20301A167) describes an approach that is acceptable to the NRC staff to meet regulatory requirements for materials, design,

construction, fabrication, examination, and testing of reinforced or prestressed concrete containments in nuclear power plants. It endorses with exceptions, the 2019 edition of ASME Section III, Division 2/American Concrete Institute (ACI) 359 code, "Code for Concrete Containments."

- RG 1.142, Revision 3, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," (ML20141L613) describes a method acceptable to the NRC staff for the analysis, design, construction, testing, and evaluation of SR nuclear concrete structures, excluding concrete reactor vessels and concrete containments. It endorses, with exceptions, ACI 349-13, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary"
- RG 1.243, Revision 0, "Safety-Related Steel Structures and Steel-Plate Composite Walls for Other than Reactor Vessels and Containments," (ML21089A032) describes a method acceptable to the NRC staff for the design, fabrication, and erection of SR steel structures and steel-plate composite walls for other than reactor vessels and containments. It endorses, with exceptions, the 2018 edition of American National Standards Institute (ANSI)/American Institute of Steel Construction (AISC) N690, "Specification for Safety-Related Steel Structures for Nuclear Facilities".

3. Summary of DOE Requirements, Orders, and Standards

Key aspects of the DOE framework related to C&S include:

- DOE Office of Nuclear Energy (NE) Order (O) 420.1D, "Facility Safety," provides an acceptable method for determining which standards will be applicable. DOE approves the contractor's list of applicable DOE technical standards or industry C&S via different means, including approval of safety design documents, the fire protection program, and inclusion of the technical C&S in the contract.
- DOE NE O 420.1D Attachment 3, "Design Criteria for Safety Structures, Systems, And Components" establishes requirements for the design and construction of safety SSCs, both safety class and safety significant, by identifying an applicable set of industry C&S, as well as DOE design criteria, standards, and directives.
- DOE-STD-1271-2026, "Authorization Pathway for Nuclear Facilities," outlines required steps in the authorization process for an applicant to comply with applicable requirements and receive DOE authorization. A Nuclear Safety Design Agreement (NSDA) and its associated Code of Record (COR) discusses designer's planned application of C&S. Where a code or standard is deemed generally applicable, but exemptions provide a better approach, the COR identifies those exemptions and refers to the technical evaluation demonstrating the alternative approach that provides equivalent or better safety compared to the associated code or standard.
- DOE-STD-1237-2021, "Documented Safety Analysis for DOE Reactor Facilities," describes an acceptable methodology for preparing a Documented Safety Analysis (DSA) for a reactor facility. A DSA is required for each DOE reactor in accordance with 10 CFR 830 Subpart B, "Safety Basis Requirements." This 1237 standard identifies other industry and DOE technical standards that are required to be used "as applicable."
- NE-STD-1020-2025, "Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities," provides requirements and guidance for the analysis and design of facilities SSC necessary to implement the Natural Phenomena Hazard (NPH) requirements of DOE NE O

420.1 and the use and reference of building codes and voluntary consensus standards in the NPH design and analysis of those facilities.

4. Staff Considerations on NRC and DOE Requirements and Guidance

This section contains high-level NRC staff observations on NRC and DOE requirements and guidance and how DOE requirements and standards may be used for various NRC licensing frameworks with respect to C&S.

Leveraging Across NRC Licensing Frameworks

As noted above in the “NRC Regulatory Requirements” section, applications for LWRs under 10 CFR Parts 50 and 52, are required to meet provisions in 10 CFR 50.55a regarding applicable C&S. While conformance with 10 CFR 50.55a is not required for applications for non-LWRs and those under 10 CFR Part 53 or proposed 10 CFR Part 57, applications must provide information on C&S and how those C&S provide appropriate SSC treatments commensurate with the importance of the safety function to be performed. All NRC licensing frameworks require information related to C&S used in the design and operation of the facility and application of C&S serve substantially the same purpose. Therefore, the NRC staff notes that the NRC licensing frameworks contain substantially similar requirements and acceptable approaches with respect to C&S, such that there would not be significant differences between NRC frameworks in how DOE information is leveraged.

Considerations on NRC and DOE Requirements and Guidance

Both NRC and DOE frameworks require the use of C&S in the design of the facility. DOE specifies applicable requirements in NE O 420.1D and requires applicants to document the use of C&S in the COR. For LWRs, NRC regulations require ASME BPVC Section III, Division 1, for design and Section XI, Division 1, for in-service inspection under 10 CFR 50.55a. For non-LWRs, the NRC has endorsed several C&S in RGs as acceptable to use. Some NRC C&S endorsements include limitations and conditions for applicants to address. While both NRC and DOE frameworks require the use of C&S, the applicable C&S may differ and the information needed to justify use may vary.

Both NRC and DOE frameworks identify the applicable C&S based on the classification of the SSCs. However, the classification of SSCs is different between the two frameworks. See Appendix C, “Licensing Event Selection and Classification of Structures, Systems, and Components” for further discussion on classification of SSCs. DOE NE O 420.1D identifies applicability of C&S for safety significant or safety class SSCs. NRC RG 1.26 describes a quality classification system related to specified C&S that may be used for components containing water, steam, or radioactive material in LWRs. For non-LWRs, RG 1.87 provides guidance on the selection of C&S for the design and construction of mechanical SSCs. In some cases, the acceptable C&S in NE O 420.1D are not aligned with the acceptable C&S in NRC regulations and guidance.

Mechanical Equipment

There are significant differences in the application of ASME BPVC requirements for the design of pressure boundary components. Under the DOE framework, ASME BPVC Section III is required for reactor vessel and internal support structures only. For other pressure vessels classified as safety class or safety significant, DOE refers to ASME Section VIII, “Rules for Construction of Pressure Vessels.” For piping classified as safety class or safety significant, DOE specifies ASME B31.3, “Process Piping.”

Under the NRC framework, the applicable ASME C&S are based on the classification of the SSCs. For LWRs, NRC regulations in 10 CFR 50.55a mandate the use of ASME Section III, Division 1 for the design and construction of SR SSCs. For non-LWRs, RG 1.87 endorses the use of ASME Section III, Division 5 for the design and construction of SR SSCs, including pressure vessels and piping. Similarly, 10 CFR 53.440(b) requires SR SSCs to use endorsed C&S or those otherwise found acceptable by NRC. For non-safety related with special treatment pressure vessels and piping, RG 1.87 provides for the use of ASME Section VIII, and B31.1, "Power Piping", or B31.3, with appropriate justification and consideration of special treatments. These differences in the application of C&S will need to be addressed when leveraging DOE authorization information.

Structures

DOE requires the use of ACI-349 for the design and construction of SR concrete structures. For safety significant concrete structures, DOE permits ACI-318, "Building Code Requirements for Structural Concrete" or ACI-349. For steel structures, DOE requires ANSI/AISC N690 for the design and construction of SR steel structures. For the design and construction of safety-significant steel structures, DOE permits the use of ANSI/AISC 360, "Specification for Structural Steel Building," ANSI/AISC 325, "Steel Construction Manual," or ANSI/AISC N690.

Under the NRC framework, RG 1.142 endorses ACI-349 for the design and construction of SR concrete structures other than concrete containments, while RG 1.243 endorses ANSI/AISC N690 for the design and construction of SR steel structures other than metal containments. For concrete containments, RG 1.136 endorses ASME Section III, Division 2 for the design and construction of concrete containments, while 10 CFR 50.55a (applicable to LWRs under 10 CFR Parts 50 and 52) mandates the use of ASME Section III, Division 1, Subsection NE for the design and construction of metal containments and RG 1.57 provides supplemental guidance. For further discussion on selection and use of C&S for structural design for external hazards, see Appendix K, "Site Characterization and Design for External Hazards."

Electrical Systems

For electrical systems, the selection of C&S depends on whether an electrical SSC supports a safety-credited function. When a safety function is credited, the applicable C&S that enhance reliability, independence, environmental qualification, and seismic qualification are applied to meet the performance objectives.

NRC applies the following consensus standards for the design, safety and protection of nuclear power generating stations. IEEE Std 308 establishes the overarching requirements for Class 1E electrical distribution systems. It addresses design features, criteria for sharing power systems among multiunit stations, surveillance and testing requirements, and documentation standards. IEEE Std 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," establishes the minimum functional and design criteria for the power, instrumentation, and control components of nuclear power plant safety systems. It relies on IEEE Std 308 for the power supply infrastructure while specifying the operation of safety logic and controls during design-basis events. IEEE Std 741, "IEEE Standard for Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations," provides specialized requirements for protecting the electrical components and pathways addressed by IEEE Std 308 and IEEE Std 603. The NRC consensus standards above are similar to the DOE IEEE standards adapted to non-commercial and defense nuclear facilities.

The NRC endorses IEEE Std 379 in RG 1.53, which establishes requirements and provides guidance for applying the single-failure criterion to the electrical power, instrumentation, and control

components of nuclear power generating station safety systems. DOE also applies IEEE Std 379 to the design of SSCs in new nuclear facilities and major modifications. The standard focuses on electrical power, instrumentation, and control elements of nuclear safety systems and is aligned with IEEE Std 603. IEEE Std 379-2014 is directly mapped to DOE Order 420.1C and DOE Guide 420.1, which explicitly require its use in SSC design. Accordingly, it serves as the primary industry consensus standard for demonstrating compliance with the single-failure criterion. Requirements for electrical independence requirements are also addressed in the complementary standard IEEE Std 384 which is endorsed by NRC RG 1.75.

NRC and DOE frameworks reference IEEE Std 323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," although they apply and endorse different versions. The DOE incorporates IEEE Std. 323-2003 (R2008) by reference into its nuclear safety and facility safety design requirements. This standard establishes environmental qualification criteria for Safety Class SSCs, ensuring that they perform under normal, abnormal, and design-basis event conditions while mitigating vulnerabilities to common-cause failure mechanisms.

Use of DOE Standards for NRC Licensing

As noted above, both NRC and DOE frameworks require the use of C&S in the design and construction of the facility. However, there are differences in how C&S are applied. The applicant should address the differences between the NRC and DOE frameworks and ensure that explicit documentation of the C&S used and the justification for suitability based on safety significance and safety function is provided in the PSAR or FSAR.

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE C&S information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

Consistent with the discussion in the main body of this white paper, this table outlines three of the key factors in determining the extent to which DOE authorization information related to C&S may be leveraged.

Table O.1: Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	The NRC and DOE frameworks rely on consensus C&S for the design and construction of safety significant SSCs. There is a potential for significant leveraging if applicants select C&S that conform to NRC requirements and guidance for components with an equivalent safety classification. Pre-application engagement with NRC could enhance the efficiency and success of a leveraging review.

Application quality and justification of relevance of DOE authorization to NRC review	To maximize potential NRC leveraging of DOE information, applications should be of high quality by identifying the applicable C&S at the system level or, when multiple C&S apply, at the component level with appropriate justification based on safety classification and safety functions. Because C&S requirements may vary according to the classification of SSCs, clearly identifying each SSC's classification in relation to NRC's classification will support information leveraging.
Safety-significance of information leveraged	There is greater potential for leveraging for C&S on SSCs of lower safety significance. Therefore, a greater degree of leveraging may be possible for NSRST or NSRSS SSCs than SR SSCs. The NRC staff should verify that the appropriate C&S are applied.

NRC Staff Approach to Leveraging DOE C&S Information

NRC staff should consider the selection of C&S used for the design and construction of safety-significant SSCs. On this basis, NRC leveraging of DOE C&S information can generally be organized into three categories:

Maximum leveraging:

- Under a maximum leveraging scenario with significant leveraging of prior DOE authorization C&S information, the NRC review would focus on verifying the selected C&S conform with NRC requirements or guidance for components with an equivalent safety classification.
- *Example:* An example of maximum leveraging would be when the applicant specifies C&S that align with corresponding NRC requirements or guidance.

Moderate leveraging:

- Under a moderate leveraging scenario, the NRC review will ensure the selected C&S and the justifications are acceptable considering the importance of the safety functions to be performed while leveraging DOE information to the extent possible.
- *Example:* An example of moderate leveraging is one where the applicant has considered differences between DOE and NRC requirements and selected some C&S that do not conform to NRC requirements or guidance and has not provided clear justifications for why C&S treatments are appropriate to ensure safety function(s).

Limited leveraging:

- Under limited leveraging, a detailed NRC review of the proposed C&S or alternative approaches would be necessary.
- *Example:* Examples of limited leveraging would be when the applicant specifies C&S that do not conform to NRC requirements or guidance without providing justification or the applicant proposes non-consensus C&S or alternative approaches to comply with the design requirements.

APPENDIX P—Instrumentation and Controls

1. NRC Regulatory Requirements

Instrumentation and controls (I&C) systems provide protection, control, monitoring, indication, actuation, data communication, and supporting functions that may be credited in the safety analysis. The applicable NRC requirements depend on the licensing framework, the safety significance and classification of the function and equipment, and the role of the I&C system in preventing, detecting, responding to, or mitigating licensing-basis events.

10 CFR Part 50

- 10 CFR 50.55a(h), "Protection and safety systems," establishes requirements for protection and safety systems. For new reactor applications within its scope, the current regulation incorporates IEEE Std 603-1991, including the January 30, 1995 correction sheet, as the principal consensus standard for safety-system criteria.
- Appendix A to 10 CFR Part 50 contains General Design Criteria (GDC) applicable to I&C.
 - GDC 13, "Instrumentation and control," and GDC 20 through 25 address protection-system functions, reliability and testability, independence, failure modes, separation of protection and control, and protection-system requirements for reactivity-control malfunctions.
 - Other GDC, including GDC 1, 2, 4, 19, 28, 29, 33, 34, 35, 38, 41, and 44, may apply depending on the credited I&C function and supporting systems.
- 10 CFR 50.49 establishes environmental qualification requirements for electrical equipment important to safety within its scope.
- Appendix B to 10 CFR Part 50 establishes quality assurance criteria applicable to safety-related design, procurement, fabrication, testing, and associated records.
- Additional I&C-specific requirements may apply under 10 CFR 50.34(f) and 10 CFR 50.62 depending on the reactor technology and licensing basis.

10 CFR Part 52

- For design certifications, standard design approvals, combined licenses, and other approvals under 10 CFR Part 52, the application must provide the design and technical information needed for the NRC to determine compliance with applicable Part 50 requirements and the selected licensing basis.
 - I&C requirements incorporated through Part 50, including 10 CFR 50.55a(h)(3), the applicable General Design Criteria of 10 CFR Part 50 Appendix A (or the plant's principal design criteria (PDC)), environmental qualification, and quality assurance requirements, remain applicable under Part 52.

10 CFR Part 53

- 10 CFR Part 53 provides a risk-informed, performance-based, and technology-inclusive framework. The I&C review is therefore anchored to the safety functions, functional design criteria, licensing-basis events, and safety classification established for the design rather than to a single prescriptive I&C architecture. Relevant provisions include 10 CFR 53.400, 53.410, 53.420, 53.440, 53.450, and 53.460.
 - 10 CFR 53.440 requires demonstration through analysis, appropriate testing, prototype testing, operating experience, or a combination thereof, that design features meet their defined functional design criteria throughout the plant's lifetime, considering applicable operating conditions and system interactions. For I&C, this

includes demonstrating that credited monitoring, control, and protection systems can perform their assigned safety functions, while addressing system interfaces, human-system interactions, and the allocation of safety functions among automation, operator actions, and active, passive, or inherent safety features.

- Part 53 also requires a functional requirements analysis and function allocation sufficient to describe how safety functions are assigned among human actions, automation, active and passive safety features, and inherent safety characteristics. This is particularly relevant to I&C for highly automated, remotely monitored, or autonomous reactor concepts.
- The Part 53 PRA or other safety risk evaluation identifies failures and challenges to plant control and safety systems, while the safety categorization and special-treatment provisions establish the treatment appropriate to the function and safety significance of the SSC.

Proposed 10 CFR Part 57

- Proposed 10 CFR Part 57 would establish a licensing framework for microreactors and other reactors with comparable risk profiles. I&C-related provisions in proposed 10 CFR Part 57 relevant to this appendix include the following:
 - Proposed 10 CFR 57.30 would establish design-criteria attributes, and proposed 10 CFR 57.60(a)(1) would require design information, including features associated with remote or autonomous operation or remote monitoring, together with analysis, test programs, prototype testing, operating experience, or a combination thereof to demonstrate the design criteria.
 - Proposed 10 CFR 57.60 would also address principal design criteria, quality assurance, programs for remote operation and monitoring, use of codes and standards, technical specifications, defense in depth, and environmental qualification of safety-related electric equipment. These provisions are particularly relevant where DOE demonstration reactors use highly automated or remote I&C architectures.

2. NRC Guidance and Related References

The principal NRC review guidance for I&C under the traditional Part 50/52 framework for light water reactors (LWRs) is NUREG-0800, Chapter 7, "Instrumentation and Controls." Chapter 7 provides the review framework for safety-related and important-to-safety I&C systems, including reactor trip, engineered safety features, safe shutdown, information and control systems important to safety, interlocks, control systems, diverse I&C systems to prevent to mitigate common cause failure (CCF), and data communication systems. Within Chapter 7 are Branch Technical Positions (BTPs) for related topics; while all are important, the more relevant are listed below.

Representative I&C industry and review guidance references, as applicable to the reactor design and applicant's chosen licensing approach:

- NUREG-0800, Chapter 7, including Section 7.0, Section 7.1 and its appendices, and the system-specific Sections 7.2 through 7.9. (ML16020A054)
- BTP 7-8 Guidance for Application of Regulatory Guide 1.22 (ML14303A143)
- BTP 7-10 Guidance on Application of Regulatory Guide 1.97 (ML16019A169)
- BTP 7-12 Guidance on Establishing and Maintaining Instrument Setpoints (ML16019A200)
- BTP 7-14 Guidance on Software Reviews for Digital Computer-Based I&C Systems (ML070670183).
- BTP 7-17, Guidance on Self-Test and Surveillance Test Provisions (ML16019A316)

- BTP 7-18 Guidance on the Use of Programmable Logic Controllers in Digital I&C Systems (ML070550073).
- BTP 7-19 Guidance for Evaluation of Defense in Depth and Diversity to Address Common-Cause-Failure (CCF) Due to Latent Design Defects in Digital I&C Systems (ML24018A123)
- BTP 7-21 Guidance on Digital Computer Real-Time Performance (ML16014A266).

Regulatory Guides (RGs)

- RG 1.22 Periodic Testing of Protection System Actuation Functions (ML083300530).
- RG 1.28 Quality Assurance Program Criteria (Design and Construction) (ML23177A002).
- RG 1.30 QA Requirements for Installation, Inspection, and Testing of I&C and Electric Equipment (ML24040A161).
- RG 1.47 Bypassed and Inoperable Status Indication for NPP Safety System (ML092330064).
- RG 1.53 Application of the Single-Failure Criterion to Safety Systems (ML033220006).
- RG 1.62 Manual Initiation of Protection Actions (ML092530559).
- RG 1.75, Criteria for Independence of Electrical Safety Systems (as applicable to licensing basis.) (ML043630448).
- RG 1.97 Accident monitoring instrumentation (ML18136A762).
- RG 1.105 for safety-related instrumentation setpoints; and RG 1.118 for periodic testing of electric power and protection systems. (ML20330A329).
- RG 1.118 Periodic Testing of Electric Power and Protection Systems (ML003739468).
- RG 1.151 Instrument Sensing Lines (ML19156A129).
- RG 1.153 Criteria for Safety Systems (ML003740022).

Digital I&C (DI&C) Related Regulatory Guides, Policies and Interim Staff Guidance (ISG)

- RG 1.152, Criteria for Programmable Digital Devices in Safety-Related Systems of NPPs (ML23054A463).
- RG 1.168 V&V, Reviews & Audits for Digital Computer Software Used in Safety Systems of NPPs (ML13073A210).
- RG 1.169 - Configuration Management Plans for Software Used in Safety Systems of NPPs (ML12355A642).
- RG 1.170 - Software Test Documentation for Software Used in Safety Systems of NPPs (ML13003A216).
- RG 1.171 - Software Unit Testing for Digital Computer Software Used in Safety Systems of NPPs (ML13004A375).
- RG 1.172 Software Requirements Specifications for Software Used in Safety Systems of NPPs (ML13007A173).
- RG 1.173 Developing Software Lifecycle Process for Software Used in Safety Systems of NPPs (ML13009A190)
- SRM-SECY-22-0076 - Expansion of Current Policy on Potential CCFs in Digital I&C Systems (ML23145A176).
- DI&C-ISG-06 Licensing process. (ML18269A259)

Commercial Grade Dedication Related Regulatory Guides

- RG 1.152, Criteria for Programmable Digital Devices in Safety-Related Systems of NPPs (ML23054A463).
- RG 1.164 - Dedication of Commercial-Grade Items for Use in NPPs (ML24038A310).
- RG 1.250 Dedication of Commercial-Grade Digital I&C Control Items for use in Nuclear Power Plants (ML22153A408).

Equipment Qualification Related Regulatory Guides

- RG 1.89 Environmental Qualification of Certain Electric Equipment Important for Safety for NPPs (ML22060A287).
- RG 1.100 Seismic Qualification of Electric and Mechanical Equipment for NPPs (ML19312C677).
- RG 1.152, Criteria for Programmable Digital Devices in Safety-Related Systems of NPPs (ML23054A463).
- RG 1.180 Guidelines for Evaluating EMI and RFI¹⁰ in Safety-Related I&C Systems (ML19175A044).
- RG 1.204 Guidelines for Lightning Protection of NPPs (ML23241A957)
- RG 1.209 Guidelines for EQ of Safety-Related Computer-Based I&C Systems in NPPs (ML070190294).

Cyber Security Related Regulatory Guides – (Reviewed Primarily by NRR/NSIR, not I&C Staff)

- RG 5.71 Cyber Security Programs for Nuclear Facilities (ML22258A204).
- RG 5.83 Cyber Security Event Notifications (ML23087A018).

Non-Light Water Reactor Regulatory Guides and Other Guidance Documents

- Design Review Guide for I&C for Non-LWR Reviews (ML21011A140)- Provides guidance for the NRC staff to use in reviewing the I&C portions of applications for advanced non-LWRs within the bounds of existing regulations.

Cybersecurity and human-factors engineering (HFE) requirements interface closely with I&C; however, those subjects are outside the scope of this appendix. The I&C review should nevertheless confirm that cybersecurity controls, human-system interfaces, remote/automated functions, and data communications do not invalidate credited I&C safety functions or assumptions.

A separate NRC rulemaking, “Incorporation by Reference of Institute of Electrical and Electronics Engineers Standard 603-2018” (90 FR 59402, Docket NRC-2024-0045), has proposed incorporating IEEE Std 603-2018 into 10 CFR 50.55a(h). Until that rulemaking is finalized and effective, the currently incorporated IEEE Std 603-1991 requirements remain controlling for applications within the scope of 10 CFR 50.55a(h).

3. Summary of DOE Requirements, Orders, and Standards

DOE Office of Nuclear Energy reactor authorization uses a safety-basis framework that includes 10 CFR Part 830, DOE/NE directives, the Nuclear Safety Design Agreement (NSDA), the project Code of Record, DOE-STD-1271 authorization products, and the facility-specific Preliminary Documented Safety Analysis (PDSA), Documented Safety Analysis (DSA), and Technical Safety Requirements (TSRs). I&C requirements are therefore established through both generally applicable DOE requirements and the project-specific safety basis and Code of Record.

For DOE NE facilities, NE O 420.1, Change 1, "Facility Safety," provides nuclear safety design criteria. For I&C, the Order requires safety-class instrumentation and control systems to incorporate sufficient independence, redundancy, diversity, and separation to ensure that credited safety functions can be performed. Safety-significant components are evaluated for redundancy on a case-by-case basis. The Order also identifies DOE-STD-1195-2011 as an acceptable method for

¹⁰ electromagnetic interference (EMI) and radio-frequency interference (RFI)

achieving high reliability of safety-significant safety instrumented systems, however that standard is for non-reactor facilities.

NE O 420.1 also identifies consensus standards used for safety-significant and safety-class electrical and I&C systems. The referenced standards include, as applicable, IEEE standards addressing equipment qualification, single-failure considerations, separation and independence, seismic qualification, safety systems, grounding, and programmable digital devices. In particular, the I&C table identifies IEEE 603 and IEEE 7-4.3.2 for safety-class I&C, while related electrical-system tables identify standards such as IEEE 323, IEEE 344, IEEE 379, IEEE 384, and IEEE 603.

The DOE safety analysis identifies safety-class and safety-significant SSCs and their functional requirements. Supporting design documents, software and firmware lifecycle records, verification and validation (V&V), qualification reports, setpoint calculations, failure analyses, surveillance and test requirements, and configuration-control records may therefore form part of the authorization basis or supporting record. DOE quality assurance requirements under 10 CFR Part 830 and applicable NE quality assurance directives provide additional pedigree for these records, subject to the actual project implementation.

4. Staff Considerations on NRC and DOE Requirements and Guidance

This section contains high-level NRC staff observations on NRC and DOE requirements and guidance and how DOE requirements and standards may be used for various NRC licensing frameworks with respect to I&C.

Leveraging Across NRC Licensing Frameworks

The NRC licensing frameworks contain similar requirements and acceptable approaches with respect to I&C, such that there are not significant differences between NRC frameworks in how DOE information is leveraged.

Considerations on NRC and DOE Requirements and Guidance

There is technical overlap between NRC and DOE I&C expectations because both frameworks rely on defined safety functions, dependable protection and control systems, appropriate independence and separation, equipment qualification, testing, configuration control, and recognized industry codes and standards. This overlap may support leveraging when the design, function, configuration, operating envelope, and technical basis remain applicable to the proposed NRC-regulated facility. Refer to Table 1 for a high-level comparison of DOE and NRC regulations and guidance included as Attachment 1.

The strongest candidates for direct technical leveraging are generally evidentiary-based items whose adequacy does not depend solely on a DOE-specific regulatory classification or acceptance criterion. Examples may include:

- I&C system architecture and functional descriptions, including allocation of functions to sensors, processors, actuation equipment, displays, monitoring, communications, and power supplies;
- Equipment environmental and seismic qualification, EMI/RFI and lightning protection evaluations, equipment qualification test results, and applicable operating-envelope limits;
- Software and firmware lifecycle documentation, requirements, V&V and test results, configuration management, anomaly disposition, and change history;
- Hardware and software platform qualification, factory and site acceptance testing, surveillance testing, calibration, setpoint methodology and uncertainty analyses, and operating experience;

- Failure analyses, including single-failure, independence/separation, defense-in-depth and diversity (D³), common-cause-failure (CCF) evaluations, fail-safe behavior, communications-loss response, and recovery testing, where the methods and acceptance criteria are sufficiently comparable; and
- DOE review records, safety evaluation findings, exemption/equivalency decisions, readiness-review observations, and test/operating data that document how the authorized I&C design performed.

NRC staff should evaluate the actual technical information and approval basis rather than assume that a change in reviewing agency, by itself, requires new analysis or repeated testing. Where DOE testing or qualification is technically adequate and remains applicable, the NRC review should focus on confirming applicability, identifying any differences, and establishing the NRC regulatory finding.

Additional NRC evaluation(s) will often be needed in areas where the DOE and NRC frameworks are not directly equivalent. Important I&C-specific differences can include the following:

- Mapping DOE safety-class and safety-significant functions and SSCs to the safety classifications, special treatments, GDC/PDC, and licensing-basis functions applicable under the selected NRC framework;
- Differences in the credited safety function, initiating events, acceptance criteria, mission time, plant operating modes, accident environment, or commercial operating lifetime;
- NRC-specific expectations for protection-system independence, single-failure treatment, testability, bypass and inoperable-status indication, CCF, D³, defense-in-depth, and accident monitoring;
- Different versions of consensus standards, project-specific Code-of-Record commitments, or DOE-approved equivalencies and exemptions;
- Quality assurance pedigree and design-control records needed to support the specific NRC finding, particularly for digital software, firmware, commercial-grade items, supplier controls, and configuration management; and
- Interfaces with cybersecurity, human factors, remote operation, autonomous functions, communications, electrical power, and other supporting systems that may be treated differently in the DOE authorization and NRC licensing bases.

Use of DOE standards for NRC licensing

Limited-scope DOE demonstrations may still provide valuable I&C evidence, especially test data, equipment qualification, operating experience, and failure-response data. However, the degree of leveraging should remain bounded by what was actually designed, reviewed, tested, and operated. A low-power or short-duration DOE demonstration should not be assumed to establish the full I&C basis for long-term commercial operation if the NRC application introduces different functions, plant configurations, service conditions, or lifecycle assumptions.

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE I&C information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable

NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

Consistent with the discussion in the main body of this white paper, this table outlines three of the key factors in determining the extent to which DOE authorization information related to I&C may be leveraged.

Table P.1: Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	DOE and NRC I&C requirements share many core technical principles, including defined safety functions, independence and separation, reliability, qualification, testing, configuration control, software assurance, and use of consensus standards. Leveraging potential is highest when the same I&C architecture, credited functions, failure assumptions, standards, qualification envelopes, and acceptance criteria apply. Particular attention should be given to classification differences, single-failure and common-cause-failure (CCF) treatment, defense-in-depth and diversity (D ³), accident monitoring, and any DOE-approved alternatives or exemptions.
Application quality and justification of relevance of DOE authorization to NRC review	The application should clearly map credited safety functions to specific equipment, software, communications, supporting power, and applicable NRC requirements. The applicant should identify the DOE documents proposed for leveraging and provide the associated design basis, software development and /V&V records, qualification reports, test data, failure analyses, configuration history, standards and editions, exemptions/equivalencies, open-item dispositions, and operating experience needed to establish traceability and continued applicability. A clear DOE-to-NRC gap analysis can likely improve review efficiency.
Safety-significance of information leveraged	NRC staff should apply greater independent review focus to I&C information directly supporting reactor trip, engineered safety features, credited automatic protective actions, safety-related actuation, accident monitoring, or other high-significance safety functions. Lower-significance control, monitoring, or support functions may support a greater degree of leveraging when their failure cannot challenge credited safety functions.

NRC Staff Approach to Leveraging DOE I&C Information

NRC staff should consider the similarity of the I&C system(s) architecture and determine what design characteristics are necessary to meet the acceptance criteria based on the reactor design. Design attributes such as independence, redundancy, predictable and repeatable performance, and defense-in-depth must have been or will need to be evaluated in adequate detail to verify that the I&C systems have been designed in such a manner to prevent or mitigate AOOs and DBAs so the SAFDLs, SARRDLs, and/or “design limits for fission product barriers” will not be exceeded. The type of limit applied would be based on the reactor’s design. Additionally, the manner by which technical specifications (TS) controlled setpoints are created and maintained to ensure safety limits are not

exceeded would have needed to have been evaluated. On this basis, NRC leveraging DOE information can generally be organized into three categories:

Maximum leveraging

- Where the DOE-authorized and NRC-proposed I&C design is materially the same and the DOE technical record is complete, NRC staff may use the existing DOE design, qualification, V&V, test, analysis, and operating evidence as direct technical support for the NRC review. The NRC staff should independently confirm the applicant's mapping to NRC requirements, verify that material design or environmental differences do not invalidate the evidence provided or conclusions drawn, review relevant DOE conditions or exemptions, and document the basis for the NRC safety determination. Repeating previously conducted, technically adequate testing solely because the reviewing agency has changed may not be necessary.
- *Example:* The NRC staff understands that the DOE-authorized facility and proposed NRC facility use essentially the same credited I&C architecture, hardware and software platform, functions, interfaces, supporting power, environmental qualification envelope, and relevant code/standard design basis. DOE review records and test/V&V evidence are complete and traceable, significant changes have not been made, and no material gap exists in the acceptance criteria needed for the NRC safety determination. With clear mapping provided by the applicant, NRC staff can directly leverage the underlying technical evidence and focus its independent review on applicability and the NRC regulatory mapping rather than repeating technically adequate testing or analysis.

Moderate leveraging

- Where the DOE information is technically useful but one or more important differences exist, NRC staff should leverage the common evidence and perform a targeted review of the identified differences between the two systems. Examples include changed safety classification or credited function, revised software or setpoints, different plant interfaces, additional commercial-duty-cycle or lifetime conditions, an NRC-specific CCF/D3 issue, a different standard, including revisions edition, or a qualification/environmental gap. The applicant should identify the differences and provide focused supplemental analysis, testing, or justification sufficient to establish the NRC finding.
- *Example:* The design for NRC licensing retains a common DOE-tested platform or substantial portions of the DOE I&C architecture, but differences exist in safety classification, credited functions, setpoints, software configuration, plant interfaces, service conditions, standards editions, or NRC-specific CCF/D3 and qualification expectations. With clear mapping provided by the applicant, NRC staff can leverage common qualifications, V&V, test, and operating evidence while requiring targeted difference analyses, supplemental testing, or additional evaluation to close the identified gaps.

Limited leveraging

- Where the DOE-authorized I&C design or review basis is not comparable, NRC staff may use the DOE information as background, operating experience, or supplemental technical evidence but should not rely on it as the principal basis for the NRC safety determination. The applicant should provide the NRC-specific design and licensing basis needed to satisfy the applicable requirements. Even in this bin, NRC staff should preserve appropriate credit

for valid component qualifications, platform evidence, generic test data, or operating experience that remains technically applicable.

- *Example:* The proposed NRC facility uses different I&C functions or architecture, revised software/ firmware, different safety functions or failure assumptions, an unbounded service environment, or DOE records that do not provide sufficient technical or QA traceability. DOE information may still inform hazards, platform operating experience, failure modes, and review focus, but the applicant would need to establish the primary NRC-specific I&C design and qualification basis. Sound component- or platform-level evidence should not be discarded solely because overall facility-level leveraging is limited

APPENDIX Q—Cybersecurity

1. NRC Regulatory Requirements

10 CFR Part 73 establishes requirements for the physical protection of nuclear facilities and materials. Sections 10 CFR 73.54 and 10 CFR 73.1110 specifically provide the regulatory framework for cybersecurity. These requirements are designed to protect digital computer systems, communication systems, and networks whose compromise could adversely affect nuclear safety, physical security, or emergency preparedness. The applicable NRC requirements depend on the relevant licensing framework and, under 10 CFR Part 53 and proposed 10 CFR Part 57, on the cybersecurity regulatory option selected by the applicant.

10 CFR Part 50

- 10 CFR 50.34(c)(2), "Physical security plan," requires each applicant for an operating license for a utilization facility that will be subject to 10 CFR 73.55 to include a cyber security plan in accordance with the criteria in 10 CFR 73.54.
- 10 CFR 73.54, "Protection of digital computer and communication systems and networks," requires high assurance that in-scope digital systems and networks are adequately protected against cyber-attacks, up to and including the design-basis threat in 10 CFR 73.1. The scope includes systems associated with safety-related and important-to-safety functions, security functions, emergency preparedness functions including offsite communications, and support systems and equipment whose compromise could adversely impact those functions. The rule also addresses asset identification, security controls, defense in depth, cyber risk management, change evaluation, incident response and recovery, program review, and records retention.
- 10 CFR 73.77, "Cyber security event notifications," establishes notification, recordkeeping, and follow-up reporting requirements for licensees subject to 10 CFR 73.54 or 10 CFR 73.110. The current rule includes one-hour, four-hour, and eight-hour notification categories, together with requirements for recording specified events and written follow-up reports.

10 CFR Part 52

For combined-license applications submitted under 10 CFR Part 52, 10 CFR 52.79(a)(36)(iii) requires a cyber security plan that complies with 10 CFR 73.54. Paragraphs 52.79(a)(36)(iv) and (v) address the implementation of the safeguards, training, and cyber security plans, as well as the protection of those plans and related Safeguards Information. Accordingly, the Part 52 cyber security review applies the Part 73 framework in conjunction with the applicable Part 52 application-content requirements.

PROPOSED — "Modernizing Security Requirements" Rulemaking: The NRC's proposed rule (91 FR 38928, June 26, 2026; Docket NRC-2025-1303) would allow applicants for new nuclear power plants under 10 CFR Parts 50 and 52 to select either the cybersecurity requirements of 10 CFR 73.54 or the technology-inclusive framework of 10 CFR 73.110. The proposal includes corresponding revisions to 10 CFR 50.34, 10 CFR 52.79(a)(36)(iii), 10 CFR 73.54, 10 CFR 73.77, and 10 CFR 73.110.

10 CFR Part 53

10 CFR Part 53 became effective April 29, 2026 and provides a risk-informed, performance-based, technology-inclusive framework for commercial nuclear plants. Its cyber security

provisions provide applicants an existing choice between the traditional 10 CFR 73.54 framework and the technology-inclusive 10 CFR 73.110 framework.

- CFR 53.860(d), "Cybersecurity," requires each holder of an operating license or combined license under Part 53 to develop, implement, and maintain a cybersecurity program under either 10 CFR 73.54 or 10 CFR 73.110.
- 10 CFR 53.1369(q) requires an operating-license application to include a cybersecurity plan in accordance with 10 CFR 73.54 or 10 CFR 73.110. Section 53.1369(r) requires a description of implementation of the security, safeguards, and cybersecurity plans and requires protection of the plans and related Safeguards Information in accordance with applicable Part 73 requirements.
- 10 CFR 73.110, "Technology-inclusive requirements for protection of digital computer and communication systems and networks," establishes a consequence-based, graded cybersecurity framework for Part 53 licensees electing to use it. The program must be commensurate with the potential consequences of cyber-attacks up to and including the design-basis threat, identify the digital assets that require protection, implement defense-in-depth protective strategies and security controls, evaluate and manage cybersecurity risk, maintain the program as the plant and threat environment change, and retain supporting records.

Proposed 10 CFR Part 57

Proposed 10 CFR Part 57 would establish a licensing framework for microreactors and other reactors with comparable risk profiles. The proposed cybersecurity provisions relevant to this appendix would include the following:

- Proposed 10 CFR 57.60(a)(8)(v)(B) would require each applicant and licensee to develop, implement, and maintain a cybersecurity program in accordance with either 10 CFR 73.54 or 10 CFR 73.110. The proposed rule would offer a consequence-based approach as an option for lower-consequence reactor designs within the proposed 10 CFR Part 57 framework.
- Proposed 10 CFR 57.18(e) would permit an applicant to reference a relevant DOE or DOW authorization for a utilization facility that has been tested and demonstrated to function safely. The applicant would still need to demonstrate that the referenced information is applicable to the NRC's licensing and cybersecurity findings.
- Proposed 10 CFR 57.60(a)(8)(xi) would require an applicant to provide a description and implementation plan for remote operation or monitoring, as applicable. Because these capabilities may create significant digital communication and access pathways, the cybersecurity review would need to address their relationship to the selected program under 10 CFR 73.54 or 10 CFR 73.110.

2. NRC Guidance and Related References

The applicable NRC cybersecurity guidance depends on the selected licensing framework and governing cybersecurity requirements. The following documents are directly relevant to the review of cybersecurity programs for commercial reactors:

- Regulatory Guide (RG) 5.71, Revision 1, "Cyber Security Programs for Nuclear Power Reactors," February 2023 (ADAMS Accession No. ML22258A204), provides an NRC-endorsed approach for implementing the requirements of 10 CFR 73.54, including the development of a cybersecurity plan and the identification and protection of Critical Digital Assets.

- NUREG-0800, “Standard Review Plan”, Section 13.6.6, Revision 0, “Cyber Security Plan,” November 2010, (ML102630477) provides staff review guidance for cybersecurity plans submitted under the traditional power-reactor licensing framework.
- RG 5.74, Revision 1, “Managing the Safety/Security Interface,” April 2015, (ML14323A549) provides guidance for maintaining coordination between safety and security activities when security measures, including cybersecurity measures, may affect plant safety.
- RG 5.83, Revision 1, “Cyber Security Event Notifications,” July 2023, (ML23087A017) provides guidance for implementing the current cybersecurity event-notification requirements under 10 CFR 73.77.
- RG 5.96, Revision 0, “Establishing Cybersecurity Programs for Commercial Nuclear Plants Licensed Under 10 CFR Part 53,” March 2026, (ML25307A090) provides implementation guidance for the technology-inclusive, consequence-based cybersecurity option under 10 CFR 73.110.
- Draft NUREG-2271, “Guidelines for Preparing and Reviewing Applications Under 10 CFR Part 57,” (ML26111A077) published for public comment in 2026, provides proposed application and review guidance for the proposed 10 CFR Part 57 framework.
- PROPOSED — The proposed “Modernizing Security Requirements” rule is accompanied by DG-5085, “Cybersecurity Programs for Nuclear Power Reactors,” which proposes Revision 2 to RG 5.71 (June 2026; ML24051A205), and DG-5103, “Establishing Risk-Informed and Technology-Inclusive Cybersecurity Programs for Commercial Nuclear Plants,” which proposes Revision 1 to RG 5.96 (June 2026; ML25307A090). The rulemaking would also amend 10 CFR 73.77. If finalized, these amendments would support the withdrawal of RG 5.83 and the use of established safety- or security-event notification processes in its place.

Cybersecurity interfaces directly with digital instrumentation and control (I&C), physical security, emergency preparedness, information security, human factors, remote operations, and supporting communications. These interfaces should be coordinated across all topical reviews to ensure that cybersecurity controls and architecture do not compromise credited safety functions, physical-security functions, emergency communications, or assumptions incorporated into the licensing basis.

3. Summary of DOE Requirements, Orders, and Standards

Cybersecurity requirements for a DOE-authorized reactor derive from DOE-wide requirements, the project-specific authorization basis, applicable contracts or agreements, and DOE-STD-1271-2026. The facility’s baseline must therefore be determined from its applicable requirements, not assumed from a generic DOE directive.

DOE Order 205.1D, “Department of Energy Cybersecurity Program,” approved on April 30, 2024, establishes the Department of Energy’s enterprise cybersecurity program. The Order adopts a shared, distributed risk-management approach; aligns with the NIST Risk Management Framework outlined in NIST SP 800-37 and the NIST Cybersecurity Framework; and requires risk-based implementation proportionate to mission impact, national security considerations, assessed risk, and the potential magnitude of harm. When incorporated as specified, its Contractor Requirements Document applies to contractor-operated information systems and operational technology (OT). This order applies to both information technology (IT) and OT and establishes requirements and program elements for risk management, system and

asset protection, security controls, configuration management, vulnerability management, monitoring and detection, incident response, recovery, supply-chain risk management, assessments, and related records. Specific contractor obligations are determined by the applicable contract and any incorporated Contractor Requirements Document.

DOE-STD-1271-2026, "Authorization Pathway for Nuclear Facilities," (September 2026), establishes requirements for IT and cybersecurity at NE-authorized facilities. The standard recognizes that certain systems and data may not constitute Federal information, requires an Interconnection Security Agreement for specified DOE/INL connections, and requires approval documents to identify applicable security standards and cybersecurity requirements, including any that were reduced or eliminated. The standard also defines cybersecurity requirements for autonomous reactor operations, including compliance with DOE O 205.1D and the Office of Nuclear Energy Program Cyber Security Plan. Facility IT and OT systems must have dedicated security plans, secure remote-operation data flows, and physical and logical separation between autonomous controls and reactor protection and safety systems.

These DOE provisions may support cybersecurity evidence for NRC licensing however; they do not directly correspond to the requirements of 10 CFR 73.54 or 10 CFR 73.110. Because DOE-STD-1271-2026 permits project-specific security standards and cyber requirements, any NRC reliance requires verification of the DOE baseline, covered systems and functions, approved reductions or alternatives, equivalencies, exemptions, and review basis.

4. Staff Considerations on NRC and DOE Requirements and Guidance

This section contains high-level NRC staff observations on NRC and DOE requirements and guidance and how DOE requirements and standards may be used for various NRC licensing frameworks with respect to cybersecurity.

Leveraging Across NRC Licensing Frameworks

The NRC licensing frameworks contain substantially similar requirements and acceptable approaches with respect to cybersecurity, such that there are not significant differences between NRC frameworks in how DOE information is leveraged. DOE and NRC cybersecurity programs share key technical principles, including risk-informed protection, defense in depth, access control, system protection, monitoring, configuration management, vulnerability management, incident response, recovery, and continuous assessment. Because DOE Order 205.1D also draws on NIST frameworks and standards, DOE evidence may support NRC compliance when it applies to the NRC-regulated configuration and is mapped to the applicable NRC cybersecurity rule.

Considerations on NRC and DOE Requirements and Guidance

The most suitable candidates for direct technical use are generally objective records documenting design, implementation, testing, assessment, and operations, whose technical validity does not depend solely on a DOE-specific administrative or regulatory classification.

Examples may include:

- system and network architecture, security-zone and segmentation information, data-flow diagrams, interconnection inventories, remote-access architecture, communication paths, and documented trust boundaries;

- digital asset inventories and function mappings, including software and firmware versions, interfaces, dependencies, supporting infrastructure, and links to safety, security, emergency preparedness, or remote-operation functions;
- implemented cyber security controls, hardening configurations, authentication and access controls, account and privilege management, removable-media controls, network protections, and monitoring/detection capabilities;
- configuration and change-management records, vulnerability assessments, patch and remediation records, penetration or security testing where applicable, control assessments, corrective actions, and closure evidence;
- incident-response, backup, recovery, and restoration plans and exercise/test results, including evidence demonstrating recovery of important IT/OT functions;
- secure-development, software/firmware provenance, supplier and cyber supply-chain evidence, and other lifecycle records where applicable to the system being credited;
- DOE Interconnection Security Agreements, project Cybersecurity Plans, IT/OT security plans, remote-operation communication security evidence, and autonomous-control cyber security test information, when applicable; and
- DOE review findings, approval conditions, assessment/audit results, exemption or tailoring decisions, open-item dispositions, cyber incident history, and relevant operating experience.

NRC staff should assess the underlying technical evidence and its current applicability rather than presume that a change in the reviewing agency, by itself, requires the underlying architecture analysis, control assessment, or testing to be repeated. Nevertheless, cybersecurity evidence is generally more time-sensitive than the design documentation. Before credit is given for prior evidence, the current configuration, software and firmware versions, interconnections, vulnerability status, threat information, and implemented controls should be verified.

Additional NRC evaluation will often be necessary because the DOE and NRC frameworks serve different regulatory purposes and may define the scope of cybersecurity activities differently. Key cybersecurity-specific differences may include:

- mapping DOE information technology/operational technology (IT/OT) system boundaries and protected assets to the digital systems, networks, assets, and functions within the scope of the applicable NRC framework under 10 CFR 73.54 or 10 CFR 73.110;
- differences between the DOE enterprise, mission, and contract-based risk framework and the NRC nuclear safety and security framework, including the design-basis-threat provisions in Part 73 and the consequence-based scope of 10 CFR 73.110;
- changes to plant design, network topology, external connectivity, remote-operation interfaces, site services, communications, software or firmware, or functions credited for safety, physical security, or emergency preparedness;
- differences in cybersecurity control sets, defense-in-depth implementation, asset categorization, risk and consequence assumptions, incident notification, corrective actions, program reviews, recordkeeping, and change-control requirements;
- DOE project-specific reductions or eliminations of requirements, approved alternatives, equivalencies, or exemptions, as well as cybersecurity provisions that applied exclusively to Federal information or DOE/INL interconnections; and
- the availability, quality, traceability, and lawful NRC access to protected or nonpublic DOE cybersecurity records needed to understand the prior review and document an independent NRC finding.

Use of DOE Standards for NRC Licensing

For limited-scope DOE demonstrations, cybersecurity information may remain valuable for assessing the tested architecture, implemented controls, operational vulnerabilities, remote communications, and observed cyber performance. However, its applicability should be limited to the configuration and threat exposure present during the DOE authorization. A short-term or isolated demonstration should not, by itself, establish the cybersecurity foundation for a commercial plant with additional external interfaces, fleet-management capabilities, remote operations, or a substantially different operational lifecycle.

5. Leveraging DOE Authorization Information in NRC Reviews

The extent to which DOE cybersecurity information may be leveraged in an NRC review is determined by the extent to which the information provides a technically adequate basis for meeting the applicable NRC regulatory requirement. Ensuring a sufficiently comparable basis for addressing NRC regulatory requirements can enhance NRC licensing review efficiency.

Consistent with the discussion in the main body of this white paper, this table outlines three of the key factors in determining the extent to which DOE authorization information related to cybersecurity may be leveraged.

Table Q.1: Factors for NRC Leveraging of DOE Information

Factor	Description
Similarity of DOE and NRC requirements and guidance	DOE and NRC cyber security programs share many technical concepts, including risk management, defense in depth, system and network protection, access control, configuration management, vulnerability management, monitoring, incident response, recovery, and continuing assessment. The potential for leveraging is greatest where the applicant maps the actual DOE program and asset scope to the selected NRC framework under 10 CFR 73.54 or 10 CFR 73.110 and demonstrates that the protected functions, architecture, threat/consequence basis, and control objectives are sufficiently comparable.
Application quality and justification of relevance of DOE authorization to NRC review	The application should provide an indexed and traceable cyber security evidence package identifying the DOE-authorized configuration, system/network architecture, digital asset inventory and functions, applicable DOE Cybersecurity Plan and IT/OT plans, implemented controls, interconnections, vulnerability and configuration status, assessment/test results, incident and recovery records, DOE review findings, and any tailoring, reductions, equivalencies, or exemptions. The applicant should clearly identify changes from the DOE configuration and explain how the prior evidence supports each NRC finding.
Safety-significance of information leveraged	NRC staff should apply greater independent review focus to cyber security information associated with digital functions that protect or support safety, physical security, emergency preparedness, remote or autonomous operation, or shared communications and supporting infrastructure. Information associated with lower-consequence or non-credited functions may provide greater opportunity for direct leveraging, while

	evidence used to establish the scope or consequence significance of protected assets warrants careful NRC confirmation.
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NRC Staff Approach to Leveraging DOE Cybersecurity Information

NRC staff should consider the similarity of relevant cybersecurity considerations, including protected functions; digital-asset scope; system and network architecture; software and firmware configurations; communications and external interfaces; threat or consequence bases; implemented controls; assessment and testing methods; incident and recovery experience; configuration and vulnerability status; and the scope and documentation of the DOE review. Based on this evaluation, the NRC's use of DOE cybersecurity information can generally be organized into three categories:

Maximum leveraging

- The DOE-authorized facility and proposed NRC facility use materially the same digital architecture, software/firmware baseline, network segmentation, interconnections, protected functions, and cyber security controls. The DOE program was documented and implemented under a clearly defined cyber security baseline that is mapped to the selected NRC framework, and DOE assessment, test, and operating records are complete and traceable. NRC staff can leverage the existing architecture, control-implementation, assessment, and test evidence directly while independently confirming the NRC scope, current configuration and vulnerability posture, applicable DOE conditions, and the basis for the NRC finding.
- *Example:* When the DOE-authorized and NRC-proposed digital configurations and cybersecurity implementations are materially equivalent, and the DOE record is complete and current, NRC staff may use the existing DOE architecture, security-control implementation, assessment, testing, and operational evidence as direct technical support for the NRC review. NRC staff should independently verify the applicant's mapping to 10 CFR 73.54 or 10 CFR 73.110, confirm the current configuration and vulnerability posture, evaluate relevant DOE conditions or tailoring decisions, and document the basis for the NRC determination. Technically adequate testing or assessments should not be repeated solely because the reviewing agency has changed; however, time-sensitive threat and vulnerability information should be updated as appropriate.

Moderate leveraging

- The NRC facility retains substantial portions of the DOE-tested architecture and cyber security controls, but meaningful differences exist in network topology, asset scope, external connectivity, remote-operation features, software or firmware versions, selected controls, threat/consequence assumptions, or NRC programmatic requirements. NRC staff can leverage the common DOE evidence and focus review on the identified deltas, with targeted supplemental analysis, testing, control assessment, or justification.
- *Example:* Where DOE cyber security information is technically useful but one or more important differences exist, NRC staff should leverage the common evidence and perform a targeted review of the identified deltas. Examples include added remote access or external communications, changed network segmentation, new digital assets

or software versions, different safety or security functions, a different 10 CFR Part 73 cyber framework, changed threat/consequence assumptions, incomplete mapping of DOE controls to NRC requirements, or an NRC-specific reporting or program-maintenance gap. The applicant should identify the differences and provide focused supplemental analysis, assessment, testing, or justification sufficient to establish the NRC finding.

Limited leveraging

- The DOE-authorized and NRC-proposed facilities have materially different digital architectures, protected functions, connectivity, cyber security baselines, or control implementations, or the prior DOE record is not sufficiently accessible, current, or traceable to support the NRC finding. DOE information may still be used as background, operating experience, or supplemental technical evidence, but the applicant should provide the NRC-specific cyber security basis and current evidence required to demonstrate compliance.
- *Example:* When the DOE cybersecurity program, digital configuration, or review basis is not sufficiently comparable to or current with the NRC's requirements, NRC staff may use DOE information as background, operating experience, or supplemental evidence. However, it should not serve as the primary basis for the NRC's cybersecurity determination. The applicant should provide an NRC-specific cybersecurity plan, scope analysis, architecture, controls, and supporting evidence sufficient to demonstrate compliance with the applicable requirements. NRC staff should continue to credit valid test data, security assessments, operating experience, and design evidence that remain technically applicable.