Final Report on Critical Codes and Standards Gaps for Microreactors

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EXECUTIVE SUMMARY

Codes and standards (C&S) provide the technical foundation for the safe, reliable, and repeatable deployment of nuclear technologies, but the current body of nuclear C&S, which were primarily developed for large light-water reactors (LWRs). Microreactors are compact, factory-fabricated, transportable, highly integrated, and possibly remotely operated. Most designs are non-light water designs, and their small source term presents a different, most likely smaller, risk profile. These dissimilarities from the current fleet of large LWRs present challenges in the application of existing C&S to microreactors. This report presents the final results of a multi-phase effort sponsored by the U.S. Department of Energy (DOE) Microreactor Program to identify, evaluate, and prioritize C&S gaps specific to microreactor deployment, and to recommend actionable steps DOE can take to support their resolution.

The assessment drew on structured engagement with two primary groups: (1) microreactor designers, vendors, and consultants and (2) subject-matter experts affiliated with Standards Development Organizations (SDOs) responsible for the codes governing reactor design, construction, and operation. This input was supplemented by targeted research into recurring technical themes. Seven specific subject areas (plus a general subject of applicability) were selected for inclusion based on their broad applicability across multiple microreactor concepts and the near-term feasibility of recommended actions: C&S applicability, inspection and condition monitoring, materials qualification, reactor transportation, fuel characterization, unique deployment configurations, and quality assurance.

Across nearly every subject area, the assessment reached a consistent central conclusion: the primary barrier to microreactor deployment is not a wholesale absence of applicable C&S, but uncertainty, inefficiency, and lack of clarity in applying an LWR-focused regulatory and technical framework to microreactor-specific attributes. Existing American Society of Mechanical Engineers, American Concrete Institute, American Institute of Steel Construction, American Society of Civil Engineers, American Nuclear Society, and Nuclear Regulatory Commission frameworks generally provide a technical foundation; the more pressing needs are targeted crosswalks, worked examples, implementation guidance, and—in a smaller number of cases—new data, standards development, or code revision.

The results of the assessments present 11 specific recommendations applying a prioritization methodology weighing applicability, deployment criticality, likelihood of resolution without DOE involvement, and the expected value of DOE investment. Recommendations were further assessed for potential execution in FY27, whether as individually funded activities or integrated under other DOE Project, including those external to the DOE Microreactor Program. Five recommendations were identified for action in FY27: development of a navigation guide for microreactor specific implementation of codes and standards; providing support to industry in communication of requirements for and approaches to inspection and condition monitoring; re-investment in the previously developed Accelerated Material Deployment suggested framework; development of a microreactor transportation applicability and responsibility crosswalk; and development of a pathway for use of alternative QA Program requirements. Notably, the higher-priority actions are not requests for new codes or standards, but efforts to demonstrate existing regulatory pathways, especially with the current proposed Part 57 rulemaking, and improve industry-wide awareness and coordination.

Collectively, the findings indicate that resolving microreactor C&S gaps is achievable primarily through targeted, near-term coordination and implementation support rather than large-scale new standards development.

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ACRONYMS

ACI	American Concrete Institute
AISC	American Institute of Steel Construction
AMD	Accelerated Material Deployment
AMMT	Advanced Materials and Manufacturing Technologies
ANS	American Nuclear Society
ARCSC	Advanced Reactor Codes and Standards Collaborative
ART-GCR	Advanced Reactor Technologies–Gas-Cooled Reactor
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BPVC	Boiler and Pressure Vessel Code
C&S	Codes and Standards
CGD	Commercial-grade dedication
DOE	US Department of Energy
EPP	elastic-perfectly plastic
EPRI	Electric Power Research Institute
HALEU	high-assay low-enriched uranium
HTGR	high-temperature gas-cooled reactor
INL	Idaho National Laboratory
LWR	light-water reactors
MARVEL	Microreactor Applications Research Validation and Evaluation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRIC	National Reactor Innovation Center
OMB	Office of Management and Budget
ORNL	Oak Ridge National Laboratory
PWR	pressurized water reactor
QA	Quality Assurance
RIM	Reliability and Integrity Management
SDO	standards development organization
SME	subject-matter expert
SSC	structures, systems, and components
TRISO	tristructural-isotropic (fuel)

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Final Report on Critical Codes and Standards Gaps for Microreactors

1. INTRODUCTION

Codes and standards (C&S) have long provided the technical foundation for the design, construction, operation, and inspection of nuclear reactors and other applications of nuclear technologies. Developed through voluntary consensus processes involving utilities, vendors, technical experts, and government/regulatory representatives, C&S establish requirements and recommended practices for the characteristics, performance, safety, and use of nuclear systems, equipment, and materials. The current body of nuclear C&S largely reflects the technologies, materials, operating experience, and safety approaches associated with the large light-water reactors (LWRs) for which it was originally developed and subsequently refined [1].

Microreactors and other advanced reactors can continue to rely on many of these established standards. However, differences in reactor size, coolant and fuel technologies, operating temperatures, passive safety features, factory fabrication, transportability, remote or autonomous operation, and deployment models may fall outside the technical basis or intended scope of existing standards. These differences may necessitate actions to revise current standards or develop a completely new standard.

This report provides the final recommendations of a multi-phase effort sponsored by the U.S. Department of Energy (DOE) Microreactor Program to systematically identify, evaluate, and prioritize C&S gaps relevant to microreactor deployment, and to develop actionable recommendations for how DOE can support the resolution of those gaps.

In an early phase, a whitepaper was developed which provided background on the Nuclear Regulatory Commission (NRC) C&S endorsement process, an overview of the state of the microreactor industry, a characterization of microreactor technologies, and a preliminary review of publicly identified gaps in microreactor C&S [2]. A key finding of that review was that the industry has focused on identifying and addressing C&S gaps for advanced reactors in general, but there was a concern that there are gaps specific to microreactor designs that may not be addressed by the larger industry habit.

Subsequent phases of the project included a structured engagement with microreactor experts, including vendors, consultants, and other stakeholders to solicit input on existing challenges and gaps related to microreactor C&S. This was followed by structured engagement with experts knowledgeable in the activities of key standards development organizations (SDOs) to gain a perspective on where gaps in C&S may exist, or are being addressed, that apply more specifically to microreactor designs [2].

Recurring subjects were identified from these interactions and adjacent investigations that present a structure for assessment and recommendation for future actions to address the gaps. This report presents the final suggestions on where the DOE can support the resolution of these C&S gaps.

2. BACKGROUND

C&S developed through an appropriate consensus standards process provides the technical foundation for design of a product and its deployment for a product's safe and reliable commercial use and consumption. This is a key attribute for the development of new reactor designs and moving them from individual demonstration projects to repeatable commercial deployment. C&S translates research, testing, operating experience, and expert judgment into commonly accepted approaches for establishing requirements or complying with requirements established by a regulatory agency for design, materials, fabrication, construction, examination, testing, operation, and maintenance. This common technical basis promotes consistent safety practices and gives designers, suppliers, licensees, and regulators objective criteria for specifying, procuring, inspecting, and accepting nuclear equipment and activities.

The use and development of industry-consensus standards is also supported by federal law and policy. Section 12(d) of the National Technology Transfer and Advancement Act of 1995 directs federal agencies to use voluntary consensus standards in carrying out their activities unless doing so would be inconsistent with law or otherwise impractical. It also directs agencies to consult and participate in the development of standards when participation is in the public interest and compatible with agency missions and resources. Office of Management and Budget (OMB) Circular A-119 implements this policy by encouraging federal agencies to use consensus standards instead of government-unique requirements.

For the U.S. commercial nuclear power industry, numerous SDOs administer the development of voluntary consensus standards that are subsequently endorsed, or required by law, by the U.S. NRC, providing applicants and licensees with an accepted method for satisfying regulatory requirements. Although voluntary standards are not automatically legally binding, NRC endorsement gives applicants and reviewers a common technical basis and reduces uncertainty regarding an acceptable regulatory approach [1].

Many existing nuclear standards, however, were developed primarily for large LWRs. New designs may use high-temperature materials, graphite structures, liquid-metal or molten-salt coolants, tristructural-isotropic (TRISO) fuel, heat pipes, passive safety systems, digital controls, additive manufacturing, or transportable factory-produced modules. Existing standards may impose requirements or provide approaches to complying with regulatory requirements that are not relevant to these designs, while also lacking necessary rules, data, test methods, and acceptance criteria for new materials and operating environments. Unless these limitations are addressed, outdated or incomplete standards can create unnecessary cost, licensing uncertainty, inconsistent regulatory treatment, and barriers to deployment.

Effective C&S preserves nuclear safety while reducing duplication, uncertainty, schedule risk, and cost. Ensuring that C&S meets the needs of microreactor technologies is therefore not merely a technical-support activity, it is necessary to create the regulatory and industrial infrastructure for commercial deployment, allowing microreactors to progress from one-of-a-kind projects to repeatable commercial products.

3. GAP IDENTIFICATION

Gap identification included two primary perspectives: (1) organizations and personnel directly involved with the design and deployment of microreactors and (2) expertise associated with the activities of SDOs responsible for the development of C&S that support the design and deployment of microreactors. This combination of insights and perspectives was augmented by additional research into some of the specific topics in an effort to maximize the understanding of the challenges and possible resolutions communicated by the interviewees.

Assessment of designer interview data identified several technical subject areas that were believed to contain gaps in C&S that if addressed, can provide a more effective and efficient achievement of design requirements and better the deployment of microreactors. The methodology for conducting the interviews and assessing the data had an objective of identifying recurring challenges affecting microreactor C&S readiness and to distinguish between technical gaps, implementation challenges, and areas where additional guidance or coordination may be beneficial. Additional details on the methodologies can be found in the appropriate reports.

The final topics for inclusion in this final report were selected based on a belief that the potential gaps relate to more than one microreactor type (i.e., a broader applicability) and that recommended actions could be accomplished on a more near-term timeframe. The topic of quality assurance (QA) was added to the topics for discussion in this final report. This is based on interview data periodically referring to issues on QA and given the current attention on QA in multiple regulatory actions applicable to microreactors, it was included as a specific subject area of this report. The selected topics are listed here:

- C&S applicability
- Inspection and condition monitoring
- Materials qualification
- Reactor transportation
- Fuel characterization
- Unique deployment configurations
- QA
- Digital Systems

3.1. Codes and Standards Applicability

Assessment of the interview data and parallel research of industry activities concludes that existing nuclear and commercial C&S provide a substantial technical foundation for microreactor design, licensing, construction, operation, and transportation. The central challenge, as repeatedly noted is not solely a wholesale absence of standards, but uncertainty and inefficiency in applying frameworks developed largely for conventional, site-built, multidecade LWRs to microreactors that may be compact, factory-fabricated, transportable, highly integrated, remotely monitored, and designed for relatively short operating lives.

Challenges or gaps with the applicability of C&S are not unique to any single SDO nor to any specific microreactor configuration or technology. As emphasized throughout the gap-assessment project documents, the regulatory framework and associated C&S have been historically LWR-focused, and microreactor attributes challenge many of the assumptions embedded within this framework. Many identified “gaps” therefore reflect issues of applicability, interpretation, implementation, or regulatory interface—issues that could be effectively addressed through targeted crosswalks, guidance documents, worked examples, and enhanced coordination among developers, SDOs, DOE, and the NRC.

A practical and efficient strategy is to rely on existing standards wherever they provide a technically defensible pathway, demonstrate their application through representative microreactor use cases, and pursue new data or targeted standards development only when a clearly defined deployment need cannot be resolved within existing provisions. This strategy has many facets and there can be significant benefit if a guide is developed to assist microreactor designers in ‘navigating’ the codes and standards that are available and how they may effectively applied.

Recommendation 1-1: Develop a guide or other tool that assists in the navigation of codes and standards to ensure applicability to microreactor-specific implementation.

Develop practical guidance to help microreactor developers understand how existing and developing codes and standards can be used to satisfy the requirements associated with different licensing pathways. Special emphasis can be given to the proposed Part 57 regulations. The guide can explain how relevant codes and standards fit together, when they should be applied during the design and licensing process, and what supporting information may be needed to demonstrate compliance. Other recommendations from this gap assessment may also benefit from integration into this recommended activity.

3.2. Inspection and Condition Monitoring

The ability to meet inservice inspection and condition monitoring requirements was a concern brought up during the interviews with designers. It was noted in Ref. [2] that the industry has made progress in developing techniques and standards for the performance of inspections and conducting condition monitoring. This progress is important because many microreactor concepts rely on compact, sealed, or highly integrated configurations that may not align with traditional inspection assumptions.

Additionally, the application of approaches developed for the LWRs do not apply to many microreactor designs because the degradation mechanisms are substantially different or the progress of degradation is less consequential for designs that have shorter operating windows that utilize replacement when timeframes are shorter.

Additional industry- and SDO-related engagement further confirmed the observation from the assessment of the interview data [2] that there may be a lack of awareness of what may be required for implementing a program for the inspection and condition monitoring to ensure that equipment will reliably perform their functions.

The primary SDO that provides standards for the conduct of inspection and condition monitoring is American Society of Mechanical Engineers (ASME). The regulatory framework for LWRs requires licensees to comply with 10 CFR 50.55a, *Codes and Standards*, which includes the requirement to develop and implement an Inservice Inspection Program that specifically meets the requirements of ASME Section XI, Division I [3].The advanced-reactor designs represented by microreactors does not have to comply with this regulation.

Advanced-reactor designs have to demonstrate that maintenance, inspection, and testing are performed to ensure that equipment important to the safe operation of the reactor will reliably perform their function. Division 2 of the 2019 Edition of ASME Section XI [4] has been recognized by the NRC [5] as a means for accomplishing this requirement. Since the first issuance of ASME Section XI, Division 2, in 2019, significant progress has been made in expanding the applicability of information across multiple reactor types as well as making improvements in the clarity and specificity of the code requirements.

As the potential use of ASME Section XI, Division 2, continues to increase, consideration of a workshop focused on the use of Division 2 is being made to provide information on what requirements associated with inspection and condition monitoring can be achieved through the use of ASME Section XI, Division 2, and what experience has been gained thus far and what technologies are being developed to support future construction and design. Focus on ensuring microreactor designers understand what is required for the conduct of inspections and condition monitoring is the primary recommendation from this assessment.

Recommendation 2-1: Provide support to industry in communication of requirements for and approaches to inspection and condition monitoring.

DOE, through the national laboratories and staff knowledgeable in ASME C&S, can support the communication and understanding of what standards are available and how they may be applied to new microreactors. This can be integrated with Recommendation 1-1 which includes the development of a code and standards Navigation Guide while other support can be coordinated with other industry groups like NEI and EPRI. These organizations work with the industry in deployment of technologies and conduct of workshops. As an example, EPRI is leading the development of a workshop for performance monitoring and the use of ASME RIM (as one primary subject). Aa workshop can help increase the understanding of what standards are available, and how they can be used, to develop inspection and condition monitoring approaches and programs. This can be augmented with the development of a document that accomplishes a similar communication to a boarder audience that may not be able to participate in such an industry workshop.

3.3. Materials Qualification

Materials qualification and elevated-temperature structural design for nuclear components are principally addressed through the ASME Boiler and Pressure Vessel Code (BPVC). Section III, Division 5 provides the primary construction and design framework for high-temperature reactor systems and includes established provisions for several materials relevant to microreactor structural and pressure-

boundary applications. For Class A elevated-temperature components, Subsection HBB establishes the design requirements, including design-by-analysis methods and associated material data [6].

For this assessment, design-by-analysis is treated as a structural design and acceptance pathway rather than as a stand-alone material-qualification process. The relevant materials-qualification question is whether the material, product form, weldment or joining method, and anticipated service conditions are adequately addressed by applicable code rules, material-property data, and qualification evidence. Where these conditions fall outside existing code coverage, additional data, analysis, or standards-development activity may be needed.

Interviews with microreactor developers and subject-matter experts (SMEs) indicate that some reported concerns may reflect limited awareness of changes incorporated into, or being developed in connection with, the 2025 Edition of ASME Section III, Division 5 [2, 7]. This is not meant to imply that there are no issues present for the qualification and benchmarking of materials for use in microreactor designs. Rather, it demonstrates the importance of distinguishing among: (1) Code-development needs already recognized by ASME, (2) implementation and applicability challenges associated with applying existing provisions to microreactor designs, and (3) material-property or qualification-data gaps for materials and service conditions not fully represented by the current code basis [2, 6-8]. This distinction helps determine whether the appropriate response is to improv industry guidance, enhance implementation support, perform additional research, or develop new or revised standards.

These subjects are primarily related to the use of metallic materials; however, there are similar and different challenges in the use of non-metallic materials such as graphite. An additional area of assessment and recommendations are added here to address what was found during parallel interviews with personnel involved with the application of graphite for microreactors and other information gathered from recent industry activities.

3.3.1. Explicit Code-Development Items

An example of limitations that are explicitly identified in the ASME code that was brought up during the assessment is HBB-I-1844, which states requirements for dissimilar-metal welds are “in the course of preparation.” HBB-II-6332 likewise identifies reference constitutive models for Type 304H stainless steel, Alloy 800H, and 2¼Cr–1Mo as under preparation, while reference models are provided for Type 316H stainless steel and 9Cr–1Mo–V [6]. These gaps can limit use of particular elevated-temperature design-by-analysis options for the affected materials or weld configurations.

Because these items are already recognized by ASME, they do not necessarily warrant a separate DOE standards project. A focused DOE role could be to determine whether national-laboratory data, model development, validation, or benchmark calculations could materially accelerate an active ASME effort when the issue is important to near-term deployment. This is consistent with the broader prioritization principle that DOE effort is most useful where it can close a demonstrated technical need without duplicating work already owned by an SDO [2].

3.3.2. Implementation and Applicability of Existing Provisions

Information from the interviews of microreactor designers identified application of ASME Section III, Division 5 design-by-analysis as a microreactor challenge and concluded that additional engagement with designers was needed to understand the specific difficulties encountered when applying the requirements. The same assessment identified deployment of materials in high-temperature environments as an area where designers could be engaged to determine how the latest code provisions can be applied.

The recent SME interviews provide important context for this gap. One interviewee characterized the principal ASME-space issue in many cases as one of “clarity” in applying what already exists, noted that the 2025 Edition made significant changes, and observed that smaller developers may have less visibility

into ongoing Section III developments than organizations with long-standing participation in ASME activities. The interviews therefore support workshops, case studies, and representative examples that help developers understand existing C&S available to them as a means of complying with relevant regulations. The Oak Ridge National Laboratory (ORNL) report provides a concise statement that captures the broader conclusion: implementation uncertainty can be more significant than the complete absence of standards, and near-term progress may be achieved through guidance, worked examples, and clarification of existing pathways [7].

The code sets some clear limits on when certain analysis methods can be used. For example, HBB-I-1800 does not allow the elastic-perfectly plastic EPP analysis for very flexible or highly nonlinear structures, like canopy or omega-type seals. HBB-I-1815 also requires that the structure's movements and strains remain small. In microreactors, this could affect components such as flexible seals, bellows, or long slender support members, but more typical parts—like vessels, heat-pipe tubes, core barrels, and support plates—are usually not restricted just because they are small.

There are other inelastic analysis options—such as strain limits, creep-fatigue, and functional deformation—but these methods still assume small deformations, and some rules like load-controlled stress checks are not fully developed. Because of this, components outside the EPP limits may not always have a complete Division 5 alternative, especially when both geometric and material nonlinearities are involved

The applicability of the available material-property data can also be design-specific. ASME Section III, Division 5 provides elevated-temperature material data within defined material-, temperature-, time-, and analysis-dependent limits [6]. For microreactor applications, the relevant challenge may be determining whether the proposed material and service conditions fall within the existing code data envelope or whether supplemental qualification information is needed.

3.3.3. Material and Environmental Qualification Data

The strongest data gap concerns service conditions for which the structural design basis does not fully represent material degradation in the actual reactor environment. HBB-3121 states that the tests underlying the design fatigue curves did not include corrosive environments that could accelerate fatigue failure. HBB-3124 separately recognizes that environmental effects, particularly fast-neutron irradiation, can change material properties and degrade fracture resistance. HBB-Y-4200 further states that the Division 5 rules do not explicitly address degradation of properties caused by service conditions or environment and that coolant- and irradiation-induced degradation must be properly characterized when it negatively affects material properties and derived allowable [6].

The interviews support the same interpretation with specific note that the elevated-temperature material qualification basis addresses only selected failure mechanisms and specifically raised radiation degradation and corrosion as additional considerations that designers must manage, including through condition monitoring or Reliability and Integrity Management (RIM) where appropriate.

DOE funded an activity to explore a means to deploy the use of a material to the reactor design as an alternative to deploying a material that has been qualified through the ASME process. This was termed Accelerated Material Deployment (AMD), and the assessment and recommended actions were documented under Ref. [8]. Possible application of this process is identified as part of a scope of work in FY27 for applying an alternative material applied to a reactor design instead of applying an ASME qualified material

The outcome of the activity was a proposal for a risk-informed, performance-based pathway for deploying materials in advanced reactors when the material is not fully covered by the ASME BPVC or lacks qualification data for the intended temperature, environment, or application. Rather than waiting for the full body of long-term testing normally required for code acceptance, an applicant could use 10 CFR 50.43(e)(1) to establish reasonable assurance of safety through an application-specific

combination of available test data, relevant operating experience, engineering analysis, and validated modeling. The approach requires sufficient baseline information to understand material behavior and credible degradation mechanisms; it is not intended to permit deployment when the available information is inadequate to establish a defensible safety basis.

Uncertainty in the use of the material would be managed through a robust RIM program consistent with ASME Section XI, Division 2. The RIM program would establish structures, systems, and components (SSC) reliability targets, monitor relevant degradation through qualified examinations, continuous monitoring, or surveillance specimens, and verify that actual performance remains within predicted limits. The licensing basis should define performance thresholds, reporting requirements, and predetermined responses to unexpected degradation—including increased monitoring, model revision, repair, replacement, or operating restrictions. The option to use this approach is provided as an alternative to the identification of materials and executing a multi-year qualification process.

3.3.4. Graphite Materials

Previous reports for this gap assessment project identified graphite as a material with limited code applicability, but little details were provided on the potential gaps associated with qualifying graphite for use in microreactors. Additional investigation was carried out to identify gaps with the qualification and use of graphite in reactor designs.

Graphite qualification is a high-priority materials challenge for high-temperature gas-cooled reactors (HTGRs) and graphite-moderated microreactors. The qualification and licensing of graphite components for advanced reactors is being pursued through the Advanced Reactor Technologies–Gas-Cooled Reactor (ART-GCR) Graphite Qualification Program, conducted through the DOE's Office of Nuclear Energy. The activities of this program are not necessarily qualifying particular commercial graphite grades for reactor vendors but generates shared data, methods, models, and the support of standards development; the reactor designer remains responsible for demonstrating grade-specific qualification under the applicable ASME Code and licensing basis.

Industry currently faces significant barriers to acquiring and using historical data. Much of the information is proprietary, held by foreign organizations, or stored in obsolete formats. Older data may have been generated using non–American Society for Testing and Materials (ASTM) methods or for graphite grades and reactor environments that differ from current applications. Consequently, industry must do more than obtain the data: it must demonstrate that the material, test methods, exposure conditions, and supporting records are sufficiently comparable and reliable for the intended use. A governed database could preserve the information while clearly identifying which data are suitable for qualification, modeling, bounding evaluations, or contextual use only.

A consolidated graphite database is needed because no current nuclear graphite grade has a complete dataset covering the range of material variability, irradiation exposure, temperature, oxidation, mechanical loading, and other conditions anticipated over a reactor's operating life. Qualification programs are lengthy and expensive, while valuable information is distributed among DOE laboratories, international research programs, graphite suppliers, reactor developers, and operating records from earlier graphite-moderated reactors. Bringing these data together under common metadata, quality-ranking, and access rules would reduce duplicative testing, improve comparisons among graphite grades, support predictive models, and provide a stronger technical basis for material qualification, component monitoring, and regulatory review. A recommendation associated with the development of this database is not included as a result of this gap assessment because this is an activity that extends beyond microreactors and should already be included under other programs. Additionally, it is not C&S centric so falls outside the scope of recommendations from this assessment. However, inclusion in this report highlights the priority and need for development and need for such a database.

A primary source for this additional investigation into gaps associated with C&S applied for the design and qualification of graphite materials comes from Ref. [9]. This report stresses the importance of the availability of graphite data, and it also identifies that life-cycle performance monitoring is an important issue because there is not enough empirical data to predict decades of graphite behavior for every new grade, coolant, temperature, dose, and geometry, the report argues for moving from an exclusively prescriptive “prove everything before operation” philosophy toward a balanced performance-monitoring approach. This aligns with the subject area of inspection and condition monitoring of this gap assessment report (Section 3.2) where the referenced graphite report specifically suggests that ASME Section XI, Division 2, becomes an integral part of the ASME qualification philosophy rather than simply an application after operations begin.

This same report [9] identifies that no graphite grade being considered for next-generation reactors has completed the full ASME/ASTM qualification process, largely because producing and testing enough material to qualify a grade the traditional way takes years. Recognizing this challenge, ASME has already begun allowing the use of historical data—test results collected by manufacturers as part of their normal quality programs—to help fill these gaps, as long as the data clearly comes from the same graphite grade and is backed by solid documentation (re-enforcing the suggestion that historical data is made more availability). It further highlights that industry is making progress on the use of smaller, targeted tests to confirm that new material behaves as expected, rather than repeating extensive testing for every grade. Studies support the idea that this kind of streamlined approach can work without sacrificing safety or reliability which may provide an opportunity to improve the qualification process for use of graphite materials.

3.3.5. Recommendations/Actions

Vendor interviewees indicated that there are challenges with the use of materials in many designs because the desired material is not qualified for the specific application or there are challenges with how it is to be applied, thus forcing the use of a small set of qualified materials being used in commercial nuclear reactors today. Few details on the materials and associated limitations were provided by the interviewees, but it was subsequently identified in the first assessment report on gaps on the application of C&S in microreactors [2] that there has been significant progress on advancing the application of materials to a broader set of operating conditions and some restrictions may be resolved. Programs, such as the DOE Advanced Materials and Manufacturing Technologies (AMMT) Program, are pursuing several additional material applications. Therefore, recommendations for future action are not focused on the type of materials and the qualification for use of these materials. Recommendations are focused on ensuring that confirming these other actions have the priority and the necessary data to pursue completion.

Highlighting that the industry continues to address material related needs is not a microreactor specific need and applies to numerous advanced reactor technologies. As a result, the recommendations may have a lower priority as a result of not being an issue that affects only microreactors not all microreactors.

Recommendation 3-1: Consolidate and extend existing material qualification and CGD experience into a reusable Microreactor procurement package.

This is a recommendation to consolidate lessons from AMMT material-qualification activities, MARVEL (Microreactor Applications Research Validation and Evaluation) commercial-grade dedication (CGD) experience, and applicable NRC and DOE guidance into a reusable procurement and acceptance package for one cross-cutting microreactor material application. For a selected material (exclude materials or components required to comply directly with ASME Section III construction rules where CGD is not permitted), define the material’s service conditions and safety function; applicable C&S; material properties and critical characteristics; supplier process, traceability, and change-control requirements; acceptance testing and sampling; and the CGD basis.

This recommendation is not presented as establishing a new qualification or CGD pathway. AMMT is already demonstrating gap-to-code pathways, and MARVEL has already demonstrated CGD for microreactor materials and components. This gap is integration and transferability, converting those project-specific experiences into a standardized package, possibly under a current or new standard, or as an industry topical report, that another developer could use.

Recommendation 3-2: Re-invest in the Accelerated Material Deployment approach and determine a possible case study and development of formal industry guidance.

Industry can foster the next phases of the AMD method described under INL/RPT-23-74258 [8] with the application of the framework to a specific material and reactor application that demonstrates how the evidence required by 10 CFR 50.43(e)(1), the material-performance model, RIM reliability targets, monitoring methods, action thresholds, license conditions, and regulatory reporting would operate in an actual licensing case. A subsequent phase to the case example would be the coordination with industry is to then take the case example information and develop more detailed guidance either through a standard or through an industry topical report that can subsequently be endorsed by the NRC and implemented more broadly by applicants.

Recommendation 3-3: Develop and codify a more expeditious ASME/ASTM qualification pathway for graphite materials.

To streamline nuclear-grade graphite qualification without compromising rigor, coordinated action can be pursued from three perspectives: ASME can strengthen and clarify the HHA-III-5000 historical-data provisions to give designers a practical, standardized method for determining when existing data can be credited toward qualification—avoiding the time and expense of a full multi-lot campaign; ASTM can further develop D7219's criteria for demonstrating material pedigree, sampling-plan adequacy, and test-method equivalency (including older ASTM methods and qualified non-ASTM/international methods), extending the standard's recent additions requiring QA and sampling documentation to establish that historical and current material share the same graphite grade; and industry can generate the technical basis needed to justify applicability of historical/manufacturer data—covering grade/process pedigree, adequate QA, equivalent test methods, and relevant temperature/fluence/environment conditions—reserving new testing only for closing the specific gaps that remain after this basis is established.

3.4. Reactor Transportation

Transportability is integral to many microreactor deployment concepts. Some concepts are factory-assembled but transported unfueled, then fueled and ultimately defueled or decommissioned at the operating site while other concepts rely more directly on mobility where the reactor is factory-assembled, fueled, and tested for transport to the operating location. In the former case, transportation is important but does not necessarily represent a new code and standards problem while the latter is influenced by the applicable packaging, safety basis, design criteria and inspection program for transport of fissile material, through all phases of transport. This shows that transportability cannot be assessed generically.

Interview data supports this assertion with transportation SMEs indicating that the applicability of C&S is driven primarily by the radioactive contents, fissile inventory, package type, activity, heat generation, criticality hazard, and transport mode rather than by the reactor technology itself. The ORNL report [7] concluded that existing transportation regulations, guidance, and consensus standards provide a substantial and generally technology-neutral foundation and that challenges with transportation may not simply be characterized as an absence of transportation regulations or standards and more as a lack of integration between transportation and reactor safety and design decisions.

Data from interviews with personnel experienced in the subject of reactor transportation identified several NRC Regulatory Guides and existing ASME pressure-vessel construction framework as examples that can be used in a Part 71 package-safety basis. This shifts the principal microreactor question from “what new transport standard is missing?”, for example, to “how should the existing package, structural,

inspection, and operating requirements be applied to an integrated reactor configuration?" The final assessment of gaps and suggested actions are presented based on the different concepts and deployment models.

3.4.1. Reactor Hardware and Fuel Transported Separately

When reactor hardware and fuel[a] are transported separately, the transportation pathway is comparatively mature. Hardware that is not itself an "irradiated" radioactive-material package that can be shipped under applicable commercial and hazardous-material requirements, while the fuel is evaluated under the existing Part 71 framework according to its contents and hazards [10]. However, a practical concern is whether certified packages provide the payload, geometry, enrichment, and fuel-form coverage needed at the scale and schedule of advanced-reactor deployment [7, 11].

High-assay low-enriched uranium (HALEU) and advanced fuel forms illustrate this distinction: the Part 71 pathway exists, but package payload, geometry, enrichment, and fuel-form capacity may still constrain deployment. NRC and DOE guidance therefore treat package availability and criticality evaluation as enabling needs rather than evidence that a new transport framework is required [12, 13]. Attention on gaps related to transportability will therefore focus on other concepts and deployment models.

3.4.2. Factory-Fueled Microreactor Transported Before Operation

Factory-fueled transport before operation presents a more significant implementation gap because the reactor, fuel, package design, factory acceptance process, and site-receipt process become closely coupled. The interview and associated assessments identify this factory-to-site interface as an area where implementation clarity can reduce redesign and deployment uncertainty. NRC SECY-24-0008 further confirms that factory fuel loading, operational testing, and features used to preclude criticality are active regulatory considerations for factory-fabricated microreactors [14].

Information obtained during the SME interviews adds an important distinction between a fueled reactor that has not operated and one that has undergone nuclear testing before shipment. Even limited operation can change the radionuclide inventory, decay heat, shielding need, containment basis, and package classification. Accordingly, the transportation safety basis cannot treat all "fresh" factory-fueled reactors as equivalent. This is primarily a design-integration and implementation issue: the existing Part 71 framework still applies, but the applicant must define the actual contents and demonstrate several attributes for the transportation package [10, 14-16].

Another implementation issue is the package containment boundary and its leak-test strategy. Leakage-test methods for Type B packages have been established [17]; however, interviewees with knowledge in transportation noted that applying conventional package leak-testing practice to an integrated reactor could be more complex because the transport configuration may contain multiple penetrations, control interfaces, internal fluid paths, and activated or plated-out material. This does not indicate that a new leak-testing standard is necessarily required. Rather, representative configurations can be used to determine whether existing methods, such as those contained in ANSI N14.5 are adequate as written or reveal a standards-development need.

3.4.3. Irradiated Microreactor Transported After Operation

For end-of-life transportation, two cases should be distinguished. If used fuel can be removed from the reactor module, the fuel can follow an established irradiated-fuel or radioactive-material transportation pathway using an approved package, with applicable physical-protection requirements addressed separately. If irradiated fuel remains integral to the reactor module, Part 71 still provides the governing safety framework, but the package and contents configuration may be unlike conventional

[a] For this assessment, it is assumed that the microreactor will be fueled with uranium that has not been previously irradiated.

spent-fuel shipments and will require a design-specific demonstration of source term, cooldown period, decay heat, shielding, containment, criticality, structural response, inspection, and security [15, 18].

SME interviewees suggested that decay heat and radiation shielding may be more important practical drivers than the absence of a governing regulation. Cooling time before shipment can materially affect both heat removal and shielding mass. The SMEs also identified 10 CFR 71.41(c) as an existing mechanism through which an applicant may propose environmental or test conditions different from the prescribed conditions when equivalent safety can be demonstrated and the proposed rule "Modernizing Package Certification Requirements" that amends Part 71 includes a new section 10 CFR 71.41(e) that allows for the use of a risk-informed transportation package approval process.. Therefore, concepts such as controlled standoff or exclusion distances could be evaluated as applicant-specific alternatives under the existing regulatory framework before concluding that the dose limits themselves must be revised [10, 16].

Routing, security, identification of safety compensatory measures and receipt responsibilities also remain important for irradiated modules, but existing requirements already provide a substantial foundation. However, the security implications of potential compensatory measures are not well explored. The relevant near-term may be to define how those requirements interact with a large integrated reactor shipment and who is responsible for each acceptance and inspection step across the reactor site, carrier, and receiving facility. [7, 16, 18, 19].

3.4.4. Other Scenario-Dependent Issues

Maritime transportation and deployment by air are possible transportation modes. The regulatory basis for these models is not a blank slate. IAEA SSR-6 remains the package-safety foundation across transport modes and several organizations have requirements and codes that apply to the transport of radioactive goods, including waste.

From the perspective of maritime transport, a principal question can be related to how a microreactor shipment fits within the existing package-plus-vessel framework, rather than whether a maritime transportation framework exists [20-22]. Recent federal and industry activities show that this implementation question is already being explored. The National Reactor Innovation Center (NRIC) has supported a multi-year maritime nuclear program addressing commercial maritime configurations, regulatory and readiness barriers, demonstration pathways, and a risk-informed transportation-package approval approach for transportable nuclear power plants in maritime shipment [22, 23]. Rather than performing parallel research with regard to maritime transport, gaps and associated recommendations for closure were not investigated any further.

Interviews with the vendors [2] indicated that the lack of a U.S. Type CF package category as a potential limitation for air deployment. Interviews with the transportation SME re-enforce that statement. U.S. regulations do contain specific requirements for fissile-material packages transported by air under 10 CFR 71.55(f), but a gap may be the availability of a practical, certifiable air-transport package and approval pathway for the mass, fissile inventory, activity, and configuration of the particular microreactor deployment scenario, especially after irradiation [2, 10, 16, 20]. Because defining the gap and importance is dependent on how the vendor is to deploy the reactor, the applicability may be limited to just a few vendors so recommending actions for gap closure is not included in this report.

3.4.5. Recommendations/Actions

Assessment of the interview data shows that transportability should be addressed as a life-cycle-integration challenge with specific technical gaps rather than as a stand-alone regulatory problem. Near-term work can build on the requirements identification activity that is documented under PNNL-38760, *Microreactor Incident-Free Transportation Radiation Dose Assessment,* December 2025 [24] by establishing representative shipment scenarios and clearly differentiate unfueled deployment from fueled or irradiated whole-unit transport. In parallel, technical efforts can continue to focus on ensuring

certified-package availability for HALEU and TRISO, improving TRISO impact and shock/vibration data, assessing repeated-transport and high-temperature material performance, and developing viable alternatives to conventional leakage testing when the reactor itself constitutes part of the package. The overall aim is not to replace the existing transportation framework, but to make it consistently applicable at microreactor scale without repeated first-of-a-kind interpretation or erosion of the defense-in-depth provided by package qualification and acceptance testing.

Elements of this are already being advanced—particularly the regulatory pathway for whole-unit transport, the development of HALEU/TRISO transportation packages, and improved shock/vibration evaluation. Although much of the progress is federally sponsored, it directly supports commercial reactor designs and transport solutions, making it more accurate to characterize the current state as "industry-enabling work involving commercial participants" rather than fully industry-led development.

Recommendation 4-1: Develop a microreactor transportation applicability and responsibility crosswalk.

DOE—working through the national laboratories and in coordination with NRC, DOT/PHMSA, package designers, and microreactor developers—should develop a crosswalk that links representative microreactor transportation states to the applicable regulatory and standards framework for highway, rail, and ship/barge transport. At a minimum, the crosswalk should differentiate:

1. Shipment of hardware and fuel separately
2. Factory-fueled microreactor shipment with no pre-transport nuclear operation[b]
3. Factory-fueled microreactor shipment after low-power or operational testing
4. Shipment of an irradiated microreactor following operation
5. Maritime or multimodal shipment of factory-fueled or irradiated modules, including port and vessel interfaces. Coordination with the US Coast Guard and IMO should be considered.

For each transportation state, the crosswalk should identify the required package classification (e.g., Type A fissile material, Type B fissile material), source term, presence of other hazardous materials (such as sodium), criticality controls, structural code pathway, leak-test method, radiation and thermal limits, security and routing requirements, Level VI inspection applicability (as relevant), pre-shipment acceptance steps, post-transport checks, multimodal transfer needs, and responsible organizations.

Developing this crosswalk directly addresses the recurring conclusion that the main gap is not missing regulations, but a lack of clarity in how to apply the existing framework.

Recommendation 4-2: Evaluate leak-testing and containment performance for integrated microreactor transport configurations.

The technical evaluation should focus on verifying leak-testing and containment performance for integrated microreactor transport configurations that involve multiple penetrations or internal boundaries, ensuring that representative system configurations are used to determine whether existing ANSI N14.5 methods remain adequate, require adaptation, or indicate a need to revise a current or develop a new standard.

[b] A factory-fueled microreactor may contain a neutron startup source.

3.5. Fuel Characterization

Industry and national laboratory feedback identified a standards gap associated with the specialized characterization methods used for TRISO fuel-manufacturing quality assurance. ASME NQA-1 provides an established framework for controlling quality-affecting activities, including data handling, personnel qualification, calibration, traceability, and records [25]. However, many TRISO-specific measurements continue to rely on laboratory procedures and historically developed best practices rather than consensus test methods. Many of these practices were developed or refined through the ART-GCR Fuel Development and Qualification Program managed through the DOE's Office of Nuclear Energy and subsequently transferred to fuel manufacturers [26]. Interviewees emphasized that, although these practices are technically established, no comprehensive industry-consensus framework identifies which characterization methods should be applied at each manufacturing stage, the performance or acceptance purpose of each measurement, or the appropriate provisions for sampling, property distributions, measurement uncertainty, and reporting.

A review of the interview data also identified opportunities to reduce the burden associated with legacy characterization practices. Some existing methods involve slow analysis, hazardous waste generation, destructive sampling, interrupted production runs, or limited ability to characterize statistically meaningful property distributions.

Research on this subject identified that work is already underway to transition some TRISO-specific laboratory practices toward consensus standards, and this existing activity can be leveraged rather than a separate activity pursued through the DOE that is duplicative. ASTM Committee C26.05 is currently developing a standardized approach for the preparation of TRISO particles for quantitative uranium recovery and subsequent uranium assay [27]. The technical basis is supported by recent ORNL research and international benchmarking.

The interviews also cautioned against retaining inherited requirements that are redundant, tied to obsolete fabrication methods, or not demonstrably connected to fuel performance. Consequently, near-term work should not attempt to standardize every historical method or specification requirement. NUREG-2246 also emphasizes identification of key fuel-manufacturing parameters and establishment of a fuel-performance envelope to inform testing requirements [28]. A broader performance-to-test applicability matrix or a comprehensive inventory of characterization methods could eventually support these objectives, but such activities would require substantial coordination among fuel vendors, laboratories, reactor developers, regulators, and standards-development organizations and are therefore not proposed here as near-term microreactor-specific DOE actions. Near-term work should instead focus on a limited, implementable codes-and-standards action that organizes and communicates common data and reporting expectations while leveraging ongoing ASTM work

The interview data and additional research suggest coordinated actions that organize existing knowledge and identify where modernization would provide measurable manufacturing, licensing, or deployment benefits.

The need for consistent characterization, reporting, sampling, uncertainty treatment, and traceability arises primarily from the TRISO fuel form and its manufacturing and qualification basis which extends the applicability to other advanced reactors using TRISO fuel. Additionally, these gaps will not apply to all of the microreactor types and as a result have lower priority or be addressed by other activities within the industry. Therefore, only a single recommendation is provided.

Recommendation 5-1: Establish common minimum quality and reporting requirements.

Develop a common characterization data and reporting template. The template should address attributes to establish a fuel and sample pedigree and applicable QA controls. This action would not establish universal acceptance limits or replace current quality program controls such as those established under ASME NQA-1 [25]. Rather, it would provide consistent fuel-specific implementation and allow

results from different laboratories and manufacturers to be understood and compared. It would also provide a consistent record structure for NRC review and production acceptance. The resulting template could be presented as an industry guidance document issued for trial use or, if intended to support a licensing position, through an industry topical report. After sufficient use and consensus, the content could be advanced through an appropriate standards-development organization as a consensus guide or standard. This staged pathway keeps the near-term action focused while maintaining a direct connection to code and standards development.

3.6. Unique Deployment Configurations

Recent microreactor C&S assessments identify civil/structural/seismic requirements, modular construction, and repeatable deployment across multiple sites as important subjects where changes to current, or development of new C&S may be of benefit. Existing American Concrete Institute (ACI), American Institute of Steel Construction (AISC), American Society of Civil Engineers (ASCE), American Nuclear Society (ANS), and ASME C&S already address many underlying technical functions, but those functions should be distinguished. For example, ANS-2.27 addresses investigations and data acquisition needed for seismic hazard assessment, ANS-2.29 addresses probabilistic seismic hazard analysis (PSHA), ASCE/SEI 4 and 43 address seismic analysis and performance-based seismic design of nuclear structures, and ACI 349, AISC N690, and ASME NQA-1 address structural design/construction and quality requirements. However, there is still an implementation and integration gap: whether the current endorsed standards and their interfaces adequately support factory fabrication, qualified modular construction, and repeated deployment of a standardized design at multiple sites[7].

The regulatory framework for the deployment of microreactors is also evolving in a direction that is intended to support standardized and repeated deployment. Proposed Part 57 expressly includes mechanisms for common application information, multiple reactors, multiple sites, manufacturing licenses, generic finality, Site Parameter Envelopes, and large designated areas. Draft NUREG-2271 provides corresponding application and staff-review guidance. As of the status date of this gap assessment report, however, Part 57 remains a proposed rule, and NUREG-2271 remains draft guidance; therefore, recommended activities should focus on developing reusable technical methods and evidence that could support implementation rather than assuming that the regulatory framework itself is already final [29, 30] and recommendations can be re-evaluated or updated after Part 57 is finalized.

The interview data, previous reports, and additional research identified potential gaps and opportunities that are being communicated here under three related areas: alignment of modular and precast construction with existing nuclear civil/structural standards; efficient site-characterization and site-to-envelope conformance for repeated deployment; and integration of multiple standards and licensing interfaces into a repeatable deployment process.

3.6.1. Modular Construction Alignment

ACI CODE-349 [31] provides requirements for nuclear safety-related concrete structures, while ANSI/AISC N690 [32] provides requirements for safety-related steel structures and includes the structural basis used for modular steel and steel-plate composite systems. The current NRC-endorsed concrete design basis under RG 1.142 relies on ACI 349, which contains restrictions on the use of precast concrete structural systems for seismic/lateral-force resistance unless an alternative qualification basis is demonstrated. While ACI 318 and ACI 319 are generally applicable for non-nuclear or non-safety related precast components, there is a gap that results from the precast usage ban under ACI 349. In other words, specific issues such as proprietary connection systems, inspection evidence, and post-installation acceptance activities are fully addressed for a proposed microreactor deployment concept. It is further confirmed by analyzing the existing C&S that the conventional precast concrete and steel-plate composite modular systems are established applications.

The nuclear industry has been making recent significant progress in the modernization of standards and increased applicability to advanced-reactor technologies, which include microreactors. ASCE-92, (Seismic Analysis and Design; not yet released), is one such standard that intends to improve alignment between commercial (non-nuclear) construction QA practices and Appendix B so that commercial constructors can be used for nuclear construction; however, the final draft of ASCE-92 is still under preparation, and the efforts should be monitored and leveraged rather than duplicated [33].

3.6.2. Site Characterization, Site Parameter Envelopes, and Data/Model Reuse

Site-specific seismic, geotechnical, hydrologic, meteorological, and demographic information remains necessary for demonstrating that an actual deployment location is suitable. This is not unique to microreactors. The microreactor-specific concern is economic and programmatic: when a standardized design is deployed many times, repeating a full site-by-site technical basis can become disproportionate to the size and schedule of each project. Proposed 10 CFR 57.18, together with Chapters 2, 3, and 17 and Appendix F of NUREG-2271, provide mechanisms for reusing common design information, defining a Site Parameter Envelope, and using area-wide or existing information where adequately justified [29, 30].

The American Nuclear Society standards ANS-2.27 [34] and ANS-2.29 [35] provide comprehensive guidance for site investigations and probabilistic seismic hazard analysis and lay out the requirements for collecting and evaluating the data necessary to assess seismic risks at nuclear facility sites, while ASCE/SEI 4 [36] and 43 [37] provide methods for seismic analysis and performance-based seismic design of nuclear SSC.

A cohesive implementation approach should clarify how various sources of data—including regional, adjacent-site, historical, federal, or commercial datasets—are evaluated and qualified for reuse in site preparation. A process for defining how a site demonstrates conformance to an approved Site Parameter Envelope could be developed, enabling an efficient and repeatable deployment while preserving site-specific safety and regulatory requirements.

3.6.3. Repeatable Deployment and Cross-Standard Integration

One challenge for the repeatable deployment of microreactors is the absence of a single integrated implementation framework linking the different pieces required for repeated deployment. Existing standards address individual elements—for example, site characterization under the ANS-2 series; seismic analysis and design under ASCE/SEI 4 and 43; concrete construction under ACI 349 [31]; steel structures under AISC N690 [32]; and QA under ASME NQA-1 [25]. Proposed 10 CFR 57 also allows applicants to identify generally recognized consensus C&S and justify their applicability, adequacy, sufficiency, supplements, or modifications [30].

This gap is therefore not simply “which standard applies,” but how the standards are assembled into a consistent, reusable deployment package. Draft NUREG-2271 [29] provides application and review guidance for code selection and for multiple-site and large-area licensing, but it does not establish a universal evaluation rubric, an accepted hierarchy of standards, or a generic technical test that automatically applies to later applicants. This suggests that an integrated process connecting the Site Parameter Envelope, site-to-envelope conformance, modular construction and installation controls, QA/configuration management, and documentation of site-specific departures could be of benefit.

This framing is also consistent with the ORNL civil/structural/seismic assessment, which concluded that the principal challenge is alignment and implementation of existing standards for modular construction, factory fabrication, transportability, and repeatable multi-site deployment—not a complete lack of applicable civil and seismic standards [7].

3.6.4. Recommendations/Actions

Two activities are suggested for consideration in the near-term. The first relates to how existing and evolving codes, standards, and licensing provisions are applied together for modular and repeatable deployment. The other involves a distinct technical question concerning reuse of site-hazard information, analytical models, and Site Parameter Envelope concepts across multiple deployments [1, 6].

Recommendation 6-1: Establish a practical code and standards implementation pathway for modular and multi-site deployment.

This action focuses on closing the implementation gap between the existing civil/structural, seismic, construction, QA, and licensing requirements and the repeatable factory-to-site deployment model envisioned for microreactors. Existing standards already address many individual elements of modular construction, especially for non-safety related components, while proposed Part 57 and draft NUREG-2271 provide mechanisms for common information, multi-site applications, Site Parameter Envelopes, manufacturing licenses, and generic finality. What remains unclear is which requirements can be applied directly, where implementation guidance or adaptation is needed, and whether any interfaces remain insufficiently addressed for repeatable modular deployment. Action under this subject area may be a good option to integrate into Recommendation 1-1 on development of a guide to navigate codes and standards applicable to microreactors.

Recommendation 6-2: Establish and demonstrate a defensible methodology for reuse of site-hazard data and analytical models across multiple deployments.

This action focuses on reducing unnecessary repetition in site characterization and hazard analysis while preserving the site-specific evidence needed to demonstrate safe deployment. Existing standards and proposed Part 57 mechanisms allow standardized design information and bounding site parameters to be used, but a sufficiently clear and reusable methodology has not yet been demonstrated for determining which seismic, geotechnical, hydrologic, meteorological, or what site information can be carried forward from one deployment to another. The principal output should be a worked multi-site deployment methodology showing what information can remain generic, what must remain site-specific, and under what conditions re-use is technically defensible. Limiting the demonstration to a small number of representative site/hazard profiles would keep the effort achievable within a short timeframe while producing a method applicable to more than one standardized reactor technology.

3.7. Quality Assurance

The interview summaries identified a mismatch between traditional nuclear QA (NQA) requirements and the practical realities of microreactor supply chains. Applying NQA-1 and ASME Section III requirements throughout the entire supply chain can be costly and difficult to scale, particularly for graphite and specialty-material suppliers that may decline nuclear work rather than establish nuclear-specific QA programs. This results in the potential for reducing the pool of qualified suppliers, increasing costs, and causing schedule delays. This can be interpreted to mean that developers need a clear and defensible pathway for crediting alternative standards and established commercial programs for the design and procurement of microreactors.

Microreactors may benefit from the flexible QA framework proposed in 10 CFR 57. The purpose of Part 57 is to develop a dedicated licensing pathway for factory-fabricated microreactors, and other low-consequence reactors. The proposed rule would not require compliance with Appendix B or mandate the use of ASME NQA-1.This flexibility means microreactor developers can select industry-approved QA programs rather than being constrained by QA systems designed for large LWRs.

Proactively leveraging this flexibility, microreactor applicants can reduce unnecessary compliance costs, expand their supplier base, and streamline design-to-deployment timelines—all while maintaining a defensible, safety-focused QA program aligned with modern regulatory expectations. SDOs and industry

groups such as NEI have recognized this potential through efforts to increase flexibility within existing standards and develop new ones. The DOE-NE Regulatory Development Program is also developing a Part 57 License Application Accelerator, whose scope could potentially incorporate an alternative QA-program pathway.

Recommendation 7-1: Develop a pathway for use of alternative QA Program requirements.

With the presence of multiple approaches to taking advantage of the flexibility afforded by Part 57 as it applies to development of a QA program, developing a guide or other means providing a pathway for adoption of alternatives for microreactor developers for adoption of QA requirements and development of a QA program.

3.8. Digital Systems

Microreactor developers are increasingly incorporating high levels of automation, remote operation, digital engineering tools, advanced diagnostics, wireless technologies, and digital-twin capabilities into their designs. These trends introduce challenges related to control system architecture, human-system interfaces, failure modes, software QA, configuration management, verification and validation, and the credibility and uncertainty treatment of continuously updated models. The assessment of developer interviews and cross-cutting assessments highlighted that digital systems and digital controls introduce challenges, particularly where autonomy, reduced staffing, and software-intensive functions intersect with licensing expectations.

One theme from the interviews is that many IEEE standards referenced in nuclear regulatory guidance are broadly technology-neutral and remain applicable to microreactor designs. These existing standards provide a foundation for digital instrumentation and control systems, software life-cycle assurance, and communications systems, helping anchor new digital capabilities within familiar and accepted frameworks. While additional guidance may likely be needed to support the deployment of microreactors, the subjects can be broad and not exclusive to only microreactors. This broader applicability opens the door for support by a larger advanced reactor technologies community and not necessarily resolved through development of a standard. Therefore, no recommendations for action regarding codes and standards are included.

4. SUMMARY OF RECOMMENDATIONS AND IDENTIFIED PRIORITY

Eleven recommendations were identified from the final assessment of possible gaps in codes and standards:

- **Code Applicability Recommendation 1-1**: Develop a guide or other tool that assists in the navigation of codes and standards to ensure applicability to microreactor-specific implementation
- **Condition Monitoring Recommendation 2-1**: Provide support to industry in communication of requirements for and approaches to inspection and condition monitoring
- **Material Qualification Recommendation 3-1**: Consolidate and Extend Existing Material Qualification and CGD Experience into a Reusable Microreactor Procurement Package
- **Material Qualification Recommendation 3-2**: Re-invest in the Accelerated Material Deployment approach and determine a possible case study and development of formal industry guidance
- **Material Qualification Recommendation 3-3:** Develop and codify a more expeditious ASME/ASTM qualification pathway for graphite materials
- **Reactor Transportation Recommendation 4-1**: Develop a microreactor transportation applicability and responsibility crosswalk
- **Reactor Transportation Recommendation 4-2**: Evaluate leak-testing and containment performance for integrated microreactor transport configurations
- **Fuel Characterization Recommendation 5-1**: Establish common minimum quality and reporting requirements
- **Deployment Configuration Recommendation 6-1**: Establish a practical code and standards implementation pathway for modular and multi-site deployment
- **Deployment Configuration Recommendation 6-2**: Establish and demonstrate a defensible methodology for reuse of site-hazard data and analytical models across multiple deployments
- **Quality Assurance Recommendation 7-1**: Develop a roadmap or formal pathway for use of alternative QA Program requirements

A method of prioritization was presented in the Idaho National Laboratory (INL) report [2]. The subjects presented and prioritized in that report were carried through, for the most part, to the subsequent assessment of SME interviews documented in ORNL report [7] and to this final gap assessment report. The process for prioritization is not repeated here and explanation on the process can be found in the INL report. Results of the prioritization process are shown in Table 1, Priority of final recommendations.

A simple approach would be to suggest action on only high priority recommendations or a combination of high and medium priority. However, an additional consideration is applied and that is the ability to commence the activity in the near-term. This is important because codes and standards are developed through a consensus process that usually requires additional time; thereby increasing the importance of getting a quick start.

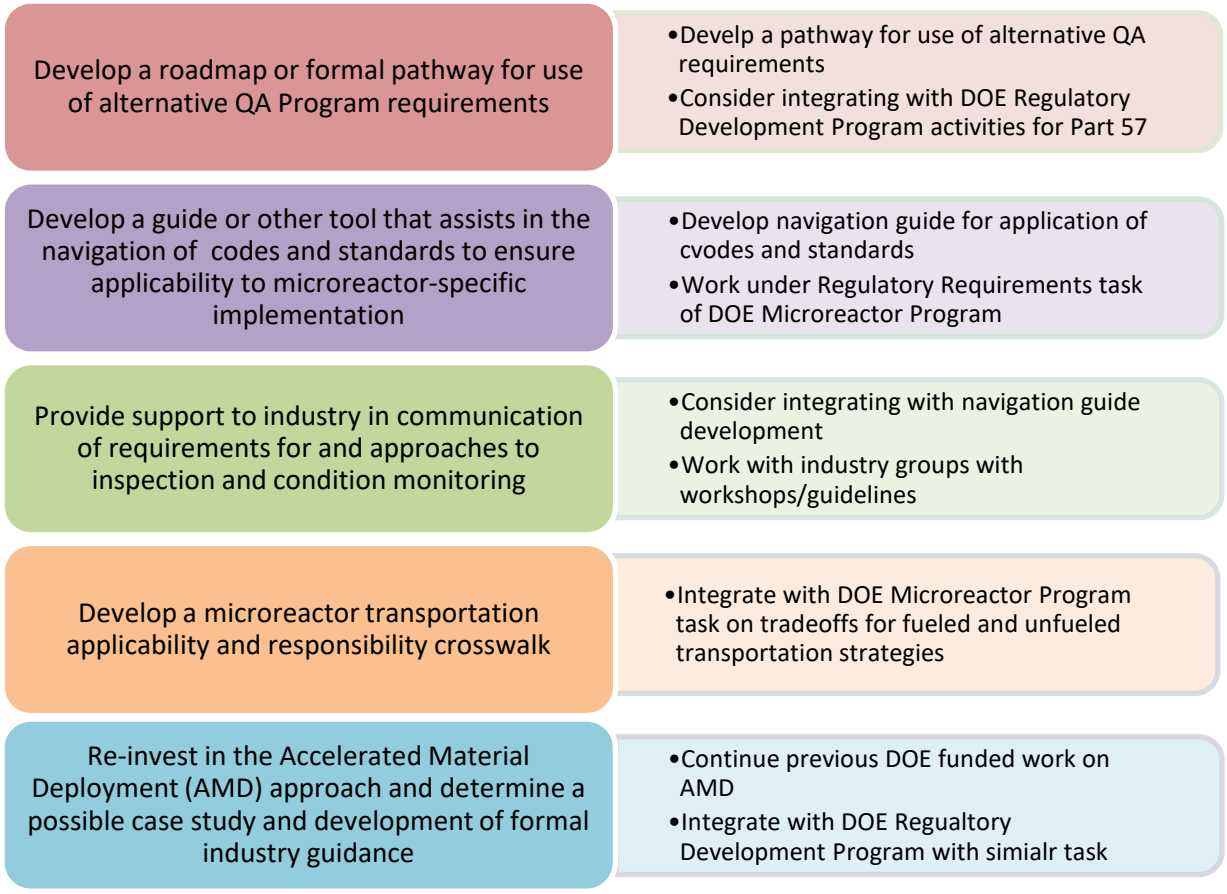
Table 1. Priority of final recommendations.

	Applicability	Deployment Criticality	Resolution without DOE	Value of DOE Investment	Priority
Code Applicability Rec. 1-1	High	Medium	Low	High	Medium
Condition Monitoring Rec. 2-1	High	High	Low	Medium	Medium
Materials Qualification Rec. 3-1	Medium	Medium	Low	Medium	Low
Materials Qualification Rec. 3-2	Medium	Medium	Medium	High	Medium
Materials Qualification Rec. 3-3	Medium	Low	High	Medium	Low
Reactor Transportation Rec. 4-1	High	Medium	Medium	Medium	Medium
Reactor Transportation Rec. 4-2	High	High	Medium	High	High
Fuel Characterization Rec. 5-1	Medium	Low	Medium	Medium	Low
Deployment Configurations Rec. 6-1	High	High	Medium	Low	Low
Deployment Configurations Rec. 6-2	Medium	Medium	Medium	High	Medium
Quality Assurance Rec. 7-1	High	High	Low	Medium	High

Each recommendation was reviewed to determine if activities may already be planned for the upcoming fiscal year (FY27) and whether the scope of the project already incorporates much of the recommendation or there may be an opportunity to influence the scope of the activity to include the recommendation. This increases the chance of successfully addressing the recommendation without having to wait an additional year for finding approval.

Figure 1 provides a summary of recommendations that are anticipated to be moving forward in FY27. Absence from this figure is not intended to imply that the other recommendations will not see action. Changes are occurring rapidly for microreactor design and deployment, the recommendations can be considered as new activities are funded.

Figure 1: Projects for FY27 implementation



5. CONCLUSION

The assessment confirms that microreactor deployment is influenced less by unresolved technical barriers and more by uncertainties in applying legacy nuclear C&S to new reactor configurations. Across multiple subject areas, including code applicability, inspection and condition monitoring, materials qualification, transportation, fuel characterization, deployment configurations, and QA, stakeholder input consistently identified implementation challenges, information/data gaps, and areas where existing standards do not fully address microreactor-specific attributes. These challenges arise from assumptions embedded in standards developed for large LWRs, resulting in inefficiencies, conservatism, and increased regulatory uncertainty even where technical bases are strong. The findings from this assessment provide a structured understanding of these issues and clarify where targeted actions can support timely and repeatable microreactor deployment.

The results highlight the need for development of a guide or means of assisting microreactor developers in understanding how C&S relate and possible pathways for application. Assessment of planned FY27 activities in either the DOE Microreactor Program, which funded this program, or other DOE programs identified opportunities for executing several recommendations. Activities that integrate with FY27 activities include:

- Develop a roadmap or formal pathway for use of alternative QA Program requirements
- Develop a guide or other tool that assists in the navigation of codes and standards to ensure applicability to microreactor-specific implementation
- Provide support to industry in communication of requirements for and approaches to inspection and condition monitoring
- Develop a microreactor transportation applicability and responsibility crosswalk
- Re-invest in the Accelerated Material Deployment (AMD) approach and determine a possible case study and development of formal industry guidance

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