Enclosure 1 to NRC Staff Prepared White Paper
"Nth-of-a-Kind Micro-Reactor Licensing and Deployment Considerations"
September 2024 Draft – Released to Support ACRS Interaction

Standardization of Operational Programs for Nth-of-a-Kind Micro-Reactors

To facilitate efficient licensing of "nth-of-a-kind" (NOAK) micro-reactors, the staff has developed two options related to the approval of standardized operational programs. Option O1 would maintain the status quo where the U.S. Nuclear Regulatory Commission (NRC) staff may review measures proposed to satisfy operational programs through topical reports (TR), or through a design-centered review approach. For the design-centered review approach, the staff would review measures to satisfy operational requirements for the first micro-reactor application of a particular design and that review would be applied to subsequent micro-reactor applications of the same design that use the same approach as proposed in the first application. Option O2 would provide for the review of measures proposed to satisfy operational requirements (e.g., technical specifications (TS) and operational programs) in either a design certification (DC) or a manufacturing license (ML) application. This enclosure discusses the NRC staff's reasoning regarding review and approval of standardized operational programs as part of the DC or ML review.

Background

Approaches that rely on TRs or a design-centered review can be implemented under the current regulatory framework without change to Commission policy. However, current Commission policy restricts the operational requirements in the context of DC or ML application review to those that are material to the finding on the safety of the design, e.g., TS. Strategies that would allow an option for resolution of all operational matters during the DC or ML review to accord finality or enhanced regulatory stability to such requirements that are not material to the safety of the design would require a change to current Commission policy. Should the Commission change its policy to allow complete review and approval of all operational requirements in connection with a DC or ML application, maximal issue resolution could be achieved through rulemaking.

Historically, the Commission has approved operational requirements proposed in a DC application only if they are material to the safety of the design. The NRC decided to accord only limited finality to operational requirements even if approved in a DC review by applying the standards of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.109, "Backfitting," when it completed the advanced boiling water reactor (ABWR) and System 80+ DC reviews in the mid-1990s. The Commission approved this approach in the ABWR and System 80+ DC rules (Appendices A and B, respectively of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants"). In SECY-96-077, "Certification of Two Evolutionary

Designs," dated April 15, 1996 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML003708129), the NRC staff stated, in part, that not providing finality for TS and other operational programs "is not inconsistent with [10 CFR] Part 52's focus on design finality and it preserves NRC's flexibility to backfit future rules on operational matters such as steam generator tube plugging criteria even though such rules may affect the design incidentally." SECY-96-077 goes on to state that "[m]ost importantly, a provision has been included in Section 4 [of the ABWR design certification rule in 10 CFR Part 52, Appendix A] to provide that the final rules do not resolve any issues regarding conditions needed for safe operation (as opposed to safe design)." (Quoted in "Standard Design Certification for the U.S. [ABWR] Design," Final Rule, (Volume 62 of the *Federal Register* (FR), page 25800 (62 FR 25800)).) The Commission approved this approach for the ABWR and System 80+ DCs in the staff requirements memorandum (SRM) SRM-SECY-96-077, "Staff Requirements – SECY-97-077- Certification of Two Evolutionary Designs," dated December 6, 1996 (ML003754873).

Further, in the preamble to the ABWR Final Rule, the Commission specifically rejected a comment on the ABWR proposed rule that requested that all design control document (DCD) requirements be accorded finality, including operational-related and other non-hardware requirements. In rejecting that comment, the Commission stated:

> The Commission has determined that NEI's proposal to assign finality to operational requirements is unacceptable, because operational matters were not comprehensively reviewed and finalized for design certification (refer to Section III.F of this SOC [rule preamble]). Although the information in the DCD that is related to operational requirements was necessary to support the NRC's safety review of the standard designs, the review of this information was not sufficient to conclude that the operational requirements are fully resolved and ready to be assigned finality under § 52.63. Therefore, the Commission retained the former Section 4(c), but reworded this provision on operational requirements and placed it in Section VI.C of this appendix with the other provisions on finality (also refer to Section VIII.C of this appendix).

The Commission also excluded from finality information voluntarily provided in the Economic Simplified Boiling Water Reactor (ESBWR) DC application regarding operational requirements. In the preamble to "[ESBWR] Design Certification," Final Rule (79 FR 61944), the Commission stated:

> [GEH included] in the DCD details on two HFE [human factors engineering] elements (procedures and training) that are not used to determine the adequacy of the HFE design. In keeping with the established Commission policy of not approving operational program elements through design certification except where necessary to find design elements acceptable, the NRC is excluding these two HFE operational program elements in the ESBWR DCD from the scope of the design approved in the rule.

In contrast, an applicant for a combined license (COL) must describe all operational requirements specified in NRC regulation in the COL application. In the SRM for SECY-02-0067, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Operational Programs (Programmatic ITAAC)," dated September 11, 2002 (ML022540755), the Commission provided direction to the NRC staff that a COL applicant is not necessarily required to have ITAAC for an operational program with the exception of emergency planning. The Commission stated:

> [ITAAC] for a program should not be necessary if the program and its implementation are fully described in the application and found to be acceptable by the NRC at the COL stage. The burden is on the applicant to provide the necessary and sufficient programmatic information for approval of the COL without ITAAC.

The Commission defined "fully described" in the SRM for SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License Application Without Inspections, Tests, Analyses, and Acceptance Criteria," dated May 14, 2004 (ML041350440). The Commission stated:

> In this context, "fully described" should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow a reasonable assurance finding of acceptability.

The NRC staff further discussed the Commission's position on operational programs for COL applications in SECY-05-0197, "Review of Operational Programs in a Combined License Application and General Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," dated October 28, 2005 (ML052770257). In SECY-05-0197, the NRC staff defines operational programs for new nuclear power plants as programs that are required by regulation, are reviewed by the NRC staff for acceptability with the results documented in the safety evaluation report and will be verified for implementation by NRC inspectors. In the SECY, the NRC staff also provided a list of operational programs required by regulation. The Commission endorsed the NRC staff recommendations on operational programs in the SRM to SECY-05-0197, dated February 22, 2006 (ML060530316), and the list of operational programs required by regulations and the NRC staff review guidance for a COL application submitted under 10 CFR Part 52 can be found in "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (NUREG-0800, Standard Review Plan (SRP)) Section 13.4, "Operational Programs," dated March 2019 (ML18344A032).[1]

Design finality under 10 CFR 52.63, "Finality of Standard Design Certifications," is premised on an essentially complete design being provided for the DC review to justify a finding that the vendor's design, structures, systems and components (SSCs), justifications, etc., are sufficient and that additional or alternative design, SSCs, justifications, etc. are not necessary (see Section VI.A of the DC rules). "Essentially complete" operational requirements are not required for a DC application; however, 10 CFR Part 52, Subpart B, "Standard Design Certifications," requires that some information, such as a design quality assurance program and proposed TS, be provided. Under Section VIII.C of the DC rules, the NRC has also included additional

[1] The SRP, NUREG-0800, has been prepared to establish criteria that the NRC staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.

procedures for changes to those operational requirements that were completely reviewed and approved as documented in the safety evaluation report on the DC application (e.g., generic changes to operational requirements that do not require a design change would be subject to 10 CFR 50.109).

The requirements under 10 CFR Part 52, Subpart F, "Manufacturing Licenses," are similar to those for a DC, requiring information related to the design of the reactor but omitting requirements for information relating to operational requirements that are not material to the adequacy of the design. An applicant for a COL under 10 CFR Part 52 or a construction permit (CP) and subsequent operating license (OL) under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," can currently reference DCs or other NRC approvals. For a design certification application, the DC review also addresses COL or CP/OL action items, which identify matters outside the scope of a DC rule (DCR) that remain to be resolved in the COL or CP/OL proceeding. Further, an application that references a DCR must demonstrate that the site characteristics fall within the postulated site parameters specified in the DCR, satisfy the interface requirements set forth in the DCR, and demonstrate that any additional requirements and restrictions set forth in the DCR will be satisfied before the COL is issued. For a COL or CP/OL application referencing a DC, the applicant must address action items (referred to as license information in certain DCs). The application must include information to show that the requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC have been satisfied or that departures from these items are justified. These will be inspected, as needed, to verify compliance. This process could be used in the case of operational programs as well. Under the approach described in Option O2, an applicant could choose to pursue additional regulatory stability by submitting operational program(s) with the DC or ML application. In this case, the applicant would need to provide sufficient information to "fully describe" the program such that the NRC staff could determine whether the program description is adequate to satisfy the applicable NRC regulations. The scope of finality or additional regulatory stability that the NRC staff would provide by approving the program would depend on the detail and adequacy of the information provided by the applicant for the operational program(s) to be considered essentially complete.

The completeness of operational programs at the design stage in a DC or ML may also be dependent on a COL or CP/OL applicant having few site-specific or owner-specific features. Under the approach described in Option O2, the NRC staff could, upon a comprehensive review, determine that the information provided by the DC or ML applicant is sufficient to describe a program that meets the applicable NRC regulations, with the NRC staff verifying site-specific aspects during the COL or CP/OL review. The NRC staff also notes that guidance may still be warranted to address the many possibilities for novel deployment models of micro-reactors for both Option O1 and Option O2.

The approach described in Option O2 would afford the NRC staff the flexibility to approve operational requirements proposed in a DC or ML application if the proposed requirements are adequately detailed such that the NRC staff can perform a comprehensive review and determine whether they satisfy NRC regulations. The NRC staff acknowledges that previous NRC policy decisions excluded operational requirements from the finality provided in 10 CFR 52.63, even if completely reviewed and approved, as explained above.

For the purposes of this enclosure, the NRC staff used the guidance in Section 13.4 of the SRP to inform the list of operational programs considered. The NRC staff notes that the operational programs referenced in this enclosure may not be all-inclusive or applicable with respect to every micro-reactor or advanced reactor design. The Advanced Reactor Content of Applications Project (ARCAP) guidance (ML23277A105) was developed to support near-term advanced reactor applicants and includes DANU-ISG-2022-01, "ARCAP Roadmap Interim Staff Guidance," issued March 2024 (ML23277A139). Appendix B of the document describes the regulations that are generally applicable to a non-light-water reactor (LWR) application for a CP and OL under 10 CFR Part 50 and DCs, COLs, and standard design approvals (SDA) under 10 CFR Part 52. The applicant should identify which operational programs are applicable to their design and deployment models.

Summary of the Operational Programs and NOAK Considerations

This section provides a summary of and describes NOAK considerations related to the following operational programs: Inservice Inspection, Inservice testing, Preservice Inspection, Preservice Testing, Motor-Operated Valve Program, Containment Leakage Rate Testing, Reactor Vessel Material Surveillance, Equipment Qualification, Fire Protection, Quality Assurance, Maintenance Rule, Reliability Assurance, Process Effluent Monitoring and Sampling Program, Radiation Program, Non-licensed Plant Staff Training, Reactor Operator Training, Operator Requalification Programs, Technical Specifications, Security and Material Control and Accounting, and Emergency Preparedness. The NOAK considerations include information related to standardizing the programs for micro-reactors and whether it might be possible to review and approve standardized operational programs in connection with a DC or ML application, through a topical report, or in a first-of-a-kind (FOAK) review ahead of submission of CP/OL or COL applications for NOAK micro-reactors.

<u>Inservice Inspection, Inservice Testing, and Preservice Inspection and Testing and Motor-Operated Valve Programs</u>

Current Framework

The purpose of the inservice inspection (ISI), inservice testing (IST), preservice inspection and testing, and motor-operating valve (MOV) programs is to assure the integrity and capability of SSCs that perform a safety or risk significant function.

The ISI program provides data on the condition of such SSCs necessary for a licensee to adequately manage deterioration and aging effects. This is done by periodically monitoring and tracking degradation (e.g., defects, corrosion, erosion) in welds and base metal (and/or graphite or composite inspection for non-LWRs) of components and component supports within the program's scope to determine their suitability for continued operation. The current ISI programs for LWRs include inspections of American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code) Class 1, 2, and 3 piping; safety-related pressure-retaining components; and component supports. Non-safety-related but safety-significant components are typically inspected as part of reliability assurance or maintenance programs. The staff review guidance for the ISI program can be found in NUREG-0800, Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing" (ML07055006), Section 5.4.2.2, "Steam Generator Program" (ML070380194), and Section 6.6, "Inservice Inspection and Testing of Class 2 and 3 Components" (ML070550071).

The purpose of the IST program is to periodically measure, assess, and track the performance of components within the program's scope. The NRC incorporates by reference the ASME *Operation and Maintenance of Nuclear Power Plants*, Division 1, Operations and Maintenance (OM) Code: Section IST (OM Code) in 10 CFR 50.55a, "Codes and standards," for IST programs in LWR nuclear power plants. The IST program is intended to verify the operational readiness of certain components within the scope of the program to perform their safety functions. The current IST programs for LWRs include components consisting of pumps, valves, and dynamic restraints (snubbers) that perform safety functions. The staff review guidance for the IST program can be found in SRP Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints" (ML16134A116), and Section 5.2.4.

The preservice testing and inspection program includes testing and the examination of certain components to assess their structural integrity and operational readiness prior to plant startup. Applicants or licensees use the test results to determine reference values for the acceptable performance of applicable components in the IST program. The staff review guidance for the preservice program can be found in both the ISI and IST program documents.

In the past, the Commission directly required implementation of a program to demonstrate the capability of MOVs to perform their safety functions because the previous ASME OM Code requirements for testing MOVs in nuclear power plants were not sufficient in light of the design of the valves and the conditions under which they must function. Since then, ASME has improved the testing requirements for MOVs to include diagnostic performance testing as part of the IST program specified in the ASME OM Code. For new reactors, the IST program includes diagnostic testing of MOVs to ensure that those components continue to be capable of performing their design-basis safety functions. SRP Section 3.9.6 provides staff guidance for review for MOVs as part of the IST program.

The NRC regulatory requirements for ISI and IST programs (which include preservice testing and MOV testing) are described in 10 CFR 50.55a, which incorporates by reference specified editions and addenda of the ASME BPV Code and OM Code. The ISI and IST programs are required to be implemented prior to initial plant startup.

As discussed in the ARCAP roadmap guidance, the requirements in 10 CFR 50.55a apply only to LWRs. That does not negate the need for applicants to comply with other regulations. With the increased use of probabilistic risk assessment (PRA) information in the design and regulation of nuclear power plants, the NRC staff anticipates that applications for future nuclear plants will include risk-informed programs. The NRC staff has developed guidance that describes the methods acceptable to the staff for the content of an application describing risk-informed ISI and IST programs for a non-LWR design. That guidance is described in DANU-ISG-2022-07, "Risk-informed ISI/IST Programs" (ML23277A145), which also references Regulatory Guide (RG) 1.246, "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). Additionally, DANU-ISG-2022-06, "Post-manufacturing and Construction Inspection, Testing and Analysis Program" (ML23277A144), includes guidance on the content of the portion of a non-LWR application associated with the development of a risk-informed post-construction (or post-manufacturing for an ML application) inspection, testing, and analysis program. In addition, as described in RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors"

(ML17325A611), the NRC has developed a set of non-LWR design criteria (Advanced Reactor Design Criteria (ARDCs)), based on the general design criteria (GDCs), for application to and insights for the development of principal design criteria for non-LWRs.

With respect to future IST Programs, ASME is finalizing the new OM-2 Code, “Code on Component Testing Requirements at Nuclear Facilities,” which provides IST provisions for new and advanced water-cooled and non-water-cooled reactors. The NRC staff is preparing a new RG to accept the ASME OM-2 Code with appropriate regulatory positions. ASME is also preparing a reformatted version of its QME-1 Standard, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” to allow its more efficient use for new and advanced water-cooled and non-water-cooled reactors. The NRC staff plans to revise RG 1.100, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants” (ML19312C677), to accept, with appropriate regulatory positions, the ASME Standard QME-1-2023 and the reformatted QME-1 version when available. Similar to the approach for ISI programs for non-water-cooled reactors described in RG 1.246, applicants may propose a license condition to implement the ASME OM-2 Code as accepted in the new RG for the IST activities for components in non-water-cooled reactors.

NOAK Strategy

Current regulations for a DC or ML do not require a detailed description of these inspection and testing programs. However, the regulations do not preclude an applicant from providing that information for a DC or ML. Depending on the COL application, some site-specific information may be required to be verified during the COL stage. The NRC regulations in 10 CFR 52.79(a)(11) require COL applicants to address the ASME BPV Code and ASME OM Code in accordance with 10 CFR 50.55a for water-cooled reactors. As stated in DANU-ISG-2022-07, the scope of a risk-informed inspection program for non-LWRs should be based on a plant-specific PRA. The applicant should identify which NRC regulations are applicable to its design.

<u>Containment Leakage Rate Testing Program</u>

Current Framework

The purposes of the containment leakage rate testing program, as defined by Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” to 10 CFR Part 50, are to assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the TS or associated bases; and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment. Several GDCs also require testing and inspection of containment, if applicable. Section 6.2.6, “Containment Leakage Testing,” dated March 2007 (ML070620007) of the SRP provides the staff guidance for the containment leakage testing program review. Section III of Appendix J to 10 CFR Part 50 requires leakage testing prior to initial fuel load.

As described in ARCAP DANU-ISG-2022-01, Appendix J to 10 CFR Part 50 is only applicable to LWRs; however, there are GDCs in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 related to containment testing, such as GDC 53, "Provisions for containment testing and inspection." As discussed in the DANU-ISG-2022-07, the GDC are generally applicable to and provide guidance in establishing the principal design criteria for reactors other than water-cooled reactors governed by the GDC. In addition, as described in RG 1.232, the NRC has developed the ARDCs as a set of non-LWR design criteria, based on the GDCs, for application to and insights for the development of principal design criteria for non-LWRs. The RG discusses ARDCs providing guidance related to containment testing and inspections for non-LWR design, if applicable.

NOAK Strategy

An applicant could provide a detailed testing program for containment or a containment-like structure, as applicable, in a DC or ML application. The NRC staff notes that some micro-reactors may be self-contained or have limited containment penetrations, which could reduce the scope of the program. Also, some designs may rely on other barriers and less on a containment or a containment-like structure. Staff encourage early engagement with the NRC to identify program applicability and treatment of SSCs in such cases.

<u>Reactor Vessel Material Surveillance Program</u>

Current Framework

The purpose of the reactor vessel material surveillance program, as required by 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50, is, in part, to monitor changes in properties of the reactor vessel which result from exposure to neutron irradiation and the thermal environment. As stated in the ARCAP roadmap, these 10 CFR Part 50 requirements are only applicable to LWR technologies. The regulations in 10 CFR 52.79(a)(13) also requires the reactor vessel surveillance program for COL applicants for LWR designs and references Appendix H to 10 CFR Part 50. Section 5.3.1, "Reactor Vessel Materials" (ML063190007), of the SRP provides the staff guidance for review of this operational program.

NOAK Strategy

An applicant could provide a detailed surveillance program for the reactor vessel at the design stage in a DC or ML, if applicable to the design. The design would assume certain operating parameters and a life cycle, which could then be utilized to define a standardized reactor vessel material surveillance program. Much of the required information may be known at the design stage for a DC or ML, such as material specifications used for the reactor vessel or housing, or special processes used for manufacture or fabrication of vessel components.

Equipment Qualification Program

Current Framework:

The purpose of the equipment qualification program is to ensure that all items of equipment that are important to safety (mechanical, electrical, and instrumentation and control equipment) can perform their design safety functions under all normal environmental conditions, anticipated operational occurrences, and accident and post-accident environmental conditions for each component's design life or a specified period. It includes all environmental conditions that may result from any normal mode of plant operation, anticipated operational occurrences, design-basis events (as defined in 10 CFR 50.49(b)(1)(ii)), post design-basis events, and containment tests.

Appendix A to 10 CFR Part 50 requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance for the safety functions to be performed. General Design Criterion 4, "Environmental and Dynamic Effects Design Bases," states, in part, that "[s]tructures, systems, and components 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, including loss-of-coolant accidents." General requirements associated with equipment qualification appear in GDC 1, "Quality Standards and Records," GDC 2, "Design Bases for Protection Against Natural Phenomena," and GDC 23, "Protection System Failure Modes," of Appendix A to 10 CFR Part 50. The regulations in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," require, in part, that the pertinent requirements of this appendix apply to all activities affecting the safety-related functions of the 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. These activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying. Further, 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states that "the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program… Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualifications testing of a prototype unit under the most adverse design conditions."

In 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," the NRC further established specific requirements for environmental qualification (EQ) of certain electric equipment important to safety located in a "harsh" environment. This regulation is also referenced in 10 CFR Part 52, Subsection C, for contents of applications for combined licenses. The implementation milestone for the EQ program is to have all qualification requirements met prior to the loading of fuel with SRP Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," dated March 2007 (ML063600397) and draft SRP Section 3.11 dated December 2017 (ML16343A167) providing staff guidance on reviewing the program to meet the requirements previously mentioned and other applicable regulations as appropriate.

Additionally, 10 CFR 50.55a(h), "Protection and safety systems," states that protection and safety systems must meet the requirements of the IEEE Std. 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations." The design-basis criteria identified in those include the range of transient and steady state environmental conditions during normal, abnormal, and accident conditions during which the equipment must perform its safety functions.

NOAK Strategy

The regulations and Atomic Energy Act of 1954, as amended (AEA), do not preclude an applicant from voluntarily providing the required information for the equipment qualification program at the DC or ML stage. However, 10 CFR 50.49 requires, in part, "each holder of or an applicant for an operating license issued under this part, or a combined license or manufacturing license issued under 10 CFR Part 52 of this chapter…shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section. For a manufacturing license, only electric equipment defined in paragraph (b) which is within the scope of the manufactured reactor must be included in the program."

For a DC applicant, in addition to providing the list of electrical equipment important to safety required by a DC for the EQ program, an applicant for a DC could also voluntarily provide a discussion related to performance specifications under conditions existing during and following design-basis accidents and the method for qualifying these SSCs, or other information required by 10 CFR 50.49 for a CP/OL or COL.

An applicant for a DC or ML could also choose to provide for NRC staff review and approval of qualification program information regarding other equipment, such as mechanical or instrument and control systems. NRC staff review and approval would be dependent on the detail and specificity of the information provided by the DC or ML applicant. The staff also notes that the amount of plant design standardization and site-specific information provided will impact the standardization of the equipment qualification program.

Fire Protection

Current Framework

The regulations in 10 CFR 50.48, "Fire protection," requires applicants for a power reactor CP, COL, DC, standard design approval, or ML, to have a fire protection plan that satisfies Criterion 3, "Fire protection," of Appendix A to 10 CFR Part 50. The fire protection plan must describe the overall fire protection plan for the facility, outline the programs for fire protection, automatic fire detection and suppression capability, and limitation of fire damage. The fire protection plan must also describe specific features necessary to implement the program, such as administrative controls (e.g., policies and procedures) and personnel requirements for fire prevention and manual fire suppression activities, and the means to limit fire damage to SSCs important to safety, including those that are safety-related, so that the capability to safely shutdown the plant is ensured. In general, nuclear power plant SSCs "important to safety" are those required to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The regulations in 10 CFR 50.48 discuss the requirement to have a fire protection plan that satisfies Criterion 3. It also specifies the requirement to specify the means to limit fire damage to SSCs important to safety so that the capability to shut down the plant safely is ensured.

The regulations in 10 CFR 50.48(a) require a fire protection program (FPP) to protect all equipment important to safety. The post-fire safe-shutdown analysis should demonstrate compliance with 10 CFR 50.48(a)(2) to ensure that the fire protection features provided for SSCs important to safe shutdown are capable of limiting fire damage to systems required to achieve and maintain post-fire safe-shutdown conditions. The NRC does not prescribe a specific method for meeting regulatory requirements governing post-fire safe-shutdown capability. The goals of the post-fire safe-shutdown capability are to clearly define a safe and stable end state, identify the minimal set of SSCs credited to reach the safe state, and provide a list of safe-state SSCs and their locations. An analysis of the capability to safely shutdown the plant after a fire evaluates the effects of a fire in the fire areas of the plant and identifies a safe shutdown success path that is free of fire damage. Procedures for effecting a safe shutdown should reflect the results and conclusions of the safe shutdown analysis.

Fire protection for nuclear power plants uses the concept of defense-in-depth to achieve the required degree of reactor safety. This concept integrates administrative controls, fire protection systems, and safe shutdown capability to achieve the following objectives:

- Preventing fires from starting.
- Rapidly detecting, controlling, and extinguishing those fires that do occur, thereby limiting fire damage.
- Provide an adequate level of fire protection for SSCs important to safety, so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant.

The defense-in-depth approach uses the design and operation of nuclear power plants in a manner that prevents and mitigates accidents that release radiation or hazardous materials. The key is to create multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is relied upon exclusively. The defense-in-depth approach includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures.

The concept of defense-in-depth is an important element of the NRC's safety philosophy. It is embodied in the requirements, and an important element of NRC's regulatory decision-making process. Since the beginning of licensing nuclear power plants, the concept of defense-in-depth has been an integral part of the regulatory framework regardless of whether the term defense-in-depth was used.

Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants" (ML23214A287), describes an approach that is acceptable to staff for licensees to meet the regulatory requirements of 10 CFR 50.48(a). Section 9.5.1.1, "Fire Protection Program" (ML090510170), of the SRP provides guidance for staff review of the fire protection program.

The guidance in DANU-ISG-2022-09, "Risk-Informed, Performance-Based Fire Protection Program (for Operations)" (ML23277A147), includes one way in which non-LWR applicants and NRC staff reviewers can ensure compliance with NRC regulations. In 10 CFR 50.48, the NRC requires each holder of an OL or COL to have a fire protection plan (also referred to in DANU-ISG-2022-09 as a program description) that satisfies Criterion 3, of Appendix A, to 10 CFR Part 50. However, the GDC in 10 CFR Part 50, Appendix A, are considered only guidance for applicants applying for licenses for non-LWRs. These applicants must propose principal design criteria (PDC) in accordance with 10 CFR 50.34, "Contents of applications; technical information," or 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report."

NOAK Strategy

The regulations in 10 CFR Part 52 do not require a complete FPP be provided at the design stage for a DC or an ML, but the regulations do require information regarding a description and analysis of the fire protection design features for the standard plant and for the reactor for a DC and ML, respectively.

Staff note that site-specific information may not be available to provide at the design stage for a DC or ML application. For instance, per DANU-ISG-2022-09, the program should address the administrative controls and personnel requirements for manual fire suppression activities, to include offsite manual firefighting resources. This information, if applicable, may not be known until a site is selected. Additionally, information traditionally contained in pre-fire plans, such as staging of fire equipment and assembly locations, may also be site-specific. However, a generic fire program could be reviewed with a DC or ML application with COL action items. The staff could then verify that information about the program is included during the COL application review.

Staff also note that the FPP requirements will be heavily dependent on the micro-reactor plant design, and the applicant should identify which requirements are applicable for that specific design. An applicant could determine at the design phase that a certain requirement is not applicable due to that specific, standardized plant design, which could then potentially further streamline the COL review for a NOAK reactor. Staff encourage early preapplication engagement with the NRC to identify applicable regulations and requirements.

<u>Quality Assurance, Maintenance Rule and Reliability Assurance Programs</u>

Current Framework for the Quality Assurance Program

The quality assurance (QA) program comprises all those planned and systematic actions necessary to provide adequate confidence that an SSC will perform satisfactorily in service. Quality assurance also includes quality control, which comprises those QA 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. The QA requirements are set forth in Appendix B to 10 CFR Part 50 and must be implemented for activities affecting safety-related plant equipment.

Multiple regulations require an applicant for an OL to have a QA program. As stated, in 10 CFR 50.54, "Conditions of licenses," each nuclear power plant subject to the QA criteria in Appendix B of 10 CFR Part 50 shall implement a QA program under 10 CFR 50.34(b)(6)(ii) or 10 CFR 52.79 of this chapter, and that a holder of a COL shall implement the QA program for operation 30 days prior to the scheduled date for the initial loading for fuel. Additionally, Criterion 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50 requires that a QA program be established and implemented.

The regulations in 10 CFR Part 52 also detail requirements for DC and ML applicants to provide QA descriptions. A DC applicant is required to provide a QA program description applied to the design of the SSCs. An ML applicant is required provide a description of QA program applied to the design, and to be applied to the manufacture of, the structures, systems, and components of the reactor. A quality assurance program description (QAPD) submitted by a DC applicant may be a QA TR or part of a safety analysis report. The ML and DC requirements do not require the operational QA program be provided.

A QAPD submitted by a COL applicant applies to all phases of a facility's life, including design, construction, and operation, and may be submitted in two phases. The first phase could apply to design and construction QA activities and the second phase could apply to operational QA activities. Per Section C.I.17.5.3 of RG 1.206, Revision 0, "Combined License Applications for Nuclear Power Plants" (ML070630027), COL applicants may use an existing QAPD which the NRC approved before the COL was issued for current use, provided that they identify and justify alternatives to, or differences from, the SRP in effect 6 months prior to the docket date of the application of a new facility.

Section 17.1, "Quality Assurance During the Design and Construction Phases," issued July 1981 (ML052350349), Section 17.2, "Quality Assurance During the Operations Phase," issued July 1981 (ML052350361), Section 17.3, "Quality Assurance Program Description," issued August 1990 (ML052350376), and Section 17.5, "Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants," issued August 2015 (ML15037A441), of the SRP provide staff guidance on review of the QA program related to different standards, but all ensuring compliance with 10 CFR Part 50 Appendix B. The ARCAP roadmap guidance also provides guidance for staff and non-LWR design applicants regarding QA.

Current Framework for the Maintenance Rule Program

The purpose of the maintenance rule (MR) program is to help assure proper plant maintenance and enhanced plant safety. Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (ML18220B281), states, "As discussed in the Statements of Consideration for the Maintenance Rule, there is a clear link between effective maintenance and safety when considering such factors as the number of transients and challenges to safety systems, and the associated need for operability, availability, and reliability of safety equipment. In addition, good maintenance is also important to ensure that failure of other than safety-related SSCs that could initiate or adversely affect a transient or accident is minimized."

The MR program is based on requirements of 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants." For a COL review, the description of the operational program and proposed implementation milestones for the MR program are reviewed in accordance with 10 CFR 50.65. The implementation milestones are plant-specific except that 10 CFR 50.65 will require that the program be fully implemented before the 10 CFR 52.103(g) finding (pre-fuel load).

Section 17.6, "Maintenance Rule," issued July 2014 (ML14099A044), of the SRP provides guidance for staff to review the MR operational program. The guidance in Section 17.6 also notes that in meeting the MR requirements, the applicant may incorporate by reference, an

NRC-approved generic final safety analysis report (FSAR) template that provides a complete generic MR program description for in developing the COL FSAR (e.g., Nuclear Energy Institute (NEI) document NEI-07-02A, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52," Revision 3 (September 2007)).

Current Framework for the Reliability Assurance Program

The reliability assurance program (RAP) applies to those SSCs, both safety-related and non-safety-related, identified as risk significant (or significant contributors to plant safety) as determined by using a combination of probabilistic, deterministic, or other methods of analysis used to identify and quantify risk. The SSCs within the scope of the RAP (referred to as "RAP SSCs") are identified by using a combination of probabilistic, deterministic, and other methods of analysis to identify and quantify risk, including PRA, severe accident evaluation, assessment of industry operating experience, and expert panel deliberation. The purpose of the RAP is to provide reasonable assurance of the following:

- A plant is designed, constructed, and operated in a manner that is consistent with the risk insights and key assumptions (e.g., SSC design, reliability, and availability) from the probabilistic, deterministic, and other methods of analysis used to identify and quantify risk.
- The RAP SSCs do not degrade to an unacceptable level of reliability, availability, or condition during plant operations.
- The frequency of transients that challenge the RAP SSCs is minimized.
- The RAP SSCs will function reliably when challenged.

The RAP is implemented in two stages. The first stage is the design reliability assurance program (D-RAP), which encompasses reliability assurance activities conducted before initial fuel load. The objective of the D-RAP is to ensure that the plant is designed and constructed in a manner that is consistent with the risk insights and key assumptions (e.g., SSC design, reliability, and availability) from the probabilistic, deterministic, and other methods of analysis used to identify and quantify risk. The key features of the D-RAP include the following:

- Programmatic controls that ensure the risk insights and key assumptions are consistent with the plant design and construction. These controls address organization responsibilities, design control activities, procedures and instructions, records, corrective action and assessment plans, and that the list of RAP SSCs is appropriately developed, maintained, and communicated to the appropriate organizations.
- QA programs related to design and construction activities to provide control over activities affecting the quality of the RAP SSCs.

The second stage comprises activities that occur during the operations phase (after initial fuel load). The objective of the RAP during the operations phase of the plant's license is to ensure that the reliability and availability of RAP SSCs are maintained commensurate with their risk significance. The RAP during the operations phase can be implemented through regulatory requirements for SSCs, including the areas of: (1) the MR program established through 10 CFR 50.65, (2) the QA program for safety-related SSCs established through Appendix B to 10 CFR Part 50, (3) QA controls for non-safety-related RAP SSCs established in accordance with Part V of SRP Section 17.5, and (4) the inservice inspection, inservice testing, surveillance testing, and maintenance programs.

Section 17.4, “Reliability Assurance Program,” issued May 2014 (ML13296A435), of the SRP provides staff guidance to review the RAP program. As previously stated, SECY-04-0032 and SECY-05-0197 discussed operational programs. SECY-18-0093, “Recommended Change to Verification of the Design Reliability Assurance Program,” dated September 20, 2018 (ML18192B471), further discussed D-RAP and discontinuation of the use of ITAAC for that operational program. The Commission approved the staff’s recommendation in the SRM for SECY-18-0093, dated August 7, 2019 (ML19219A944). Although the RAP guidance does not prohibit review and approval of the program for a CP/OL, the guidance is currently geared toward a COL applicant.

NOAK Strategy

The staff review and approval would be heavily dependent on the quality of information and level of detail provided by the applicant. Standardization of the programs would depend on considerations such as the standardization of design, deployment and organization, and the level of detail and quality of information provided by the applicant.

As discussed above, the regulations require a QAPD to be submitted per application. For instance, the NRC staff review a DC applicant’s QAPD for design and construction (to the point that construction detail is developed by the DC). The COL applicant would also have design responsibilities related to the review of the design for their application, in addition to the QA information required for operation. An applicant who is the same entity for both the DC and CP/OL or COL could potentially more easily generate standardized QA programs. For instance, Appendix B to 10 CFR Part 50 requires that the “authority and duties of persons and organizations performing activities affecting the safety-related functions of structures, systems, and components shall be clearly established and delineated in writing.” This information and clear delineation of duties for the operational QA program may or may not be known at the design phase but may be more likely to be known if the same entity owns all QAPDs. Additionally, the NRC staff notes the clear delineation of program ownership and responsibilities is important to ensure that the QA requirements across all phases of the facility’s life are understood. Applicants pursuing a standardized program could also consider how the design QA requirements for a DC and/or ML are incorporated into the CP/OL or COL, and which additional operational QA requirements are required for a CP/OL or COL. A standardized operational QA program should take into account early consideration of program implementation.

The scope of these programs may also be driven by the life cycle, design and safety categorization of the plant SSCs. For instance, if there are few structures that meet the requirements for inclusion in the MR and very little maintenance is expected for the plant (e.g., a micro-reactor with a shortened deployment cycle), the applicant could potentially justify a more limited program scope.

Process and Effluent Monitoring and Sampling Program

Current Framework

The purpose of the process and effluent monitoring and sampling program is for a licensee to control, monitor, perform radiological evaluations of all releases, document and report all radiological effluents discharged into the environment. The principal regulatory basis for requiring the effluent and environmental monitoring at nuclear power plants is contained in multiple regulations, including GDCs 60, 61, and 64 of Appendix A of 10 CFR Part 50, 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents – nuclear power reactors," and 10 CFR 50.36a, "Technical specifications on effluents from nuclear power reactors." The regulations in 10 CFR Part 20, "Standards for Protection Against Radiation," also provide dose limits and further requires that licensees comply with the Environmental Protection Agency's generally applicable environmental radiation standards of 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," for facilities that are part of the fuel cycle. Appendix I to 10 CFR Part 50, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," also includes guidance on design objectives which may be used by applicants to meet provisions in 10 CFR 50.34a.

This program consists of four component programs: such as the radiological effluent technical specifications/standard radiological effluent controls (RETS/SREC), the offsite dose calculation manual, radiological environmental monitoring program, and process control program (PCP), all of which are required by fuel load.

The generic template in NEI 07-09A, "Generic FSAR Template Guidance for Offsite Dose Calculation Manual Program Description," Revision 0, March 2009 (ML091050234), fully describes, at the functional level, elements of the process and effluent monitoring and sampling programs required by 10 CFR Part 50, Appendix I, and 10 CFR 52.79(a)(16). The document NEI 07-10A, "Generic FSAR Template Guidance for Process Control Program (PCP)," Revision 0, March 2009 (ML091460627), is also one element of the process and effluent monitoring and sampling program. While the NRC staff has not endorsed either template, it has approved both templates via safety evaluation, which is similar to an approved topical report.

Section 11.4, "Solid Waste Management System," dated January 2016 (ML15029A174), and Section 11.5, "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems," dated January 2016 (ML15029A182), of the SRP provide NRC staff review guidance. The ARCAP roadmap and Chapter 9, DANU-ISG-2022-03, "Control of Routine Plant Radioactive Effluents, Plant Contamination and Solid Waste," of the ARCAP guidance (ML23277A141) provide additional discussion on process and effluent monitoring and sampling programs for non-LWRs.

NOAK Strategy

For a DC and ML, 10 CFR 52.47, "Contents of applications; technical information," and 10 CFR 52.157, "Contents of applications; technical information in final safety analysis report," respectively, require that information be provided with respect to the design of equipment to maintain control over radioactive materials in gaseous and liquid effluents during normal operations described in 10 CFR 50.34a(e), and the kinds and quantities of radioactive materials expected to be produced in the operations and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR Part 20. However, the detailed operational program is not required.

An applicant could voluntarily submit information for staff review during the design stage to support review of this operational program. Such information could include, if applicable, specific instruments, design features to reduce radioactivity levels in wastes, and sampling procedures. The NRC staff notes that site-specific information, such as sampling locations, could be conditioned in the license and verified by NRC staff upon implementation.

Radiation Protection Program (Including Minimization of Contamination)

Current Framework

The purpose of the radiation protection (RP) program is to protect people and the environment from unnecessary exposure to radiation as a result of civilian uses of nuclear materials. An as low as is reasonably achievable (ALARA) program is included as part of the RP program. The base requirements for the RP and ALARA programs are found in 10 CFR 20.1101, "Radiation protection programs," which requires in paragraph (a), that licensees develop, document, and implement a radiation protection program commensurate with the scope of 10 CFR Part 20 and in paragraph (b) that licensees use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA.

To ensure that the RP requirements in 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations," and 10 CFR Part 20 are met, the information that is typically presented in a design application includes a discussion of how RP practices are incorporated into operational programs and plans and design decisions; a general description of the radiation source terms; radiation protection design features, including a description of plant shielding, ventilation systems, and area radiation and airborne radioactivity monitoring instrumentation; designation of radiation areas; a dose assessment for operating and construction personnel; and a discussion of the design of the health physics facilities. The application should also include proposed PDC that address the design, fabrication, construction, testing, and quality of the structures, systems, and components necessary to control occupational exposure to within regulatory limits. The GDC 19, 61, 63, and 64, in Appendix A to 10 CFR Part 50, also provide requirements with respect to LWR design to protect workers and the public from radiation exposure. For COL reviews, the description of the operational program and proposed implementation milestone for the RP programs are reviewed in accordance with 10 CFR 20.1101 and, to the extent that it is not described in other sections, the leakage control program required by 10 CFR 50.34(f)(2)(xxvi).

The document NEI 07-03A, "Generic FSAR Template Guidance for Radiation Program Description," Revision 0, May 2009 (ML091490684), is a complete generic RP program description for use with COL applications. It provides guidance for the phased implementation of the RP program. While the NRC staff has not endorsed NEI 07-3A, it has approved the NEI 07-03A RP program template via safety evaluation and NEI 07-03A is similar to an approved topical report. An applicant may also refer to NEI 07-08A, "Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures are as Low as is Reasonably

Achievable (ALARA)," Revision 0, October 2009 (ML093220178), which provides a complete generic ALARA program description for use in developing COL applications. While the NRC staff has not endorsed NEI 07-08A, it has approved the NEI 07-08A generic program description template via safety evaluation.

Current staff guidance for the RP programs can be found in Section 12.1, "Assuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable" dated September 2013 (ML13151A061), and Section 12.5, "Operational Radiation Protection Program," dated September 2013 (ML13155A232), of the SRP. Chapter 10 of DANU-ISG-2022-04, "Control of Occupational Dose," dated March 2024 (ML23277A142), also provides guidance.

In addition to the RP program requirements in 10 CFR 20.1101, 10 CFR 20.1406, "Minimization of contamination," requires that facility design and procedures are in place to minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. The regulations in 10 CFR 20.1406 also require that licensees, to the extent practical, conduct operations to minimize the introduction of residual radioactivity into the site, including the subsurface, in accordance with the existing RP requirements in Subpart B, "Radiation Protection Programs," of 10 CFR Part 20 and the radiological criteria for license termination in Subpart E, "Radiological Criteria for License Termination," of 10 CFR Part 20.

An application for a DC, SDA, and ML must describe the design features associated with minimizing contamination of the facility and the environment, facilitating eventual decommissioning, and minimizing the generation of radioactive waste in accordance with 10 CFR 20.1406(b). The programmatic and procedural aspects of 10 CFR 20.1406 are not required to be provided until the COL or OL application is filed. The minimization of contamination program and/or procedures are usually included as part of the RP program. The licensee implements the program and/or procedures for minimization of contamination at the time of initial fuel load.

Regulatory Guide 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning" (ML080500187), provides guidance for meeting the requirements of 10 CFR 20.1406. The document NEI 08-08A, "Generic FSAR Template Guidance for Life Cycle Minimization of Contamination," Revision 0, October 2009 (ML093220530), provides a complete generic program description for use in developing a 10 CFR 20.1406 program for a COL application.

While NEI 07-03A, NEI 07-08A, and NEI 08-08A are not applicable to the review and issuance of construction permits and operating licenses under 10 CFR Part 50, these documents may be useful as templates to develop programs applicable to these applications.

NOAK Strategy

Although a detailed RP program is required for a CP/OL or COL applicant, the DC, and ML regulations require some discussion of radiation protection information, such as the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR Part 20. The regulations and AEA do not preclude an applicant from voluntarily providing additional RP program information (including 10 CFR 20.1406 program and/or procedural information) at the design stage for a DC or ML. Staff review and approval would be heavily dependent on the detail and specificity of the information provided by the applicant.

The NRC staff note that site-specific information could be conditioned in the DC or ML and verified by NRC staff upon implementation. For instance, under the current guidance in Section C.I.12 of RG 1.206, "Radiation Protection," dated June 2007 (ML070630016), prior to initial fuel receipt, the RP program should detail the RP organization and the facilities available to support receipt, storage and control of nonexempt radioactive sources. This information may be specific to a site or COL applicant and may therefore not be known at the design phase. However, the DC or ML applicant could provide a recommended organization or facilities that a COL applicant could reference and justify any site or owner-specific differences.

While most of the RP program may not require site-specific information, some aspects of the minimization of contamination programs and/or procedures are more likely to require site-specific information, such as the appropriate groundwater monitoring provisions or potential leakage paths to the environment at a specific site.

The size of the RP program and amount of RP work required for micro-reactors would be expected to be reduced compared to the current operating power reactor fleet. Specific details of the RP program for a particular micro-reactor application, including on the necessary size and scope of the radiation program, would depend on the specific application.

Non-licensed Plant Staff Training, Reactor Operator Training and Requalification Programs

Current Framework for Non-licensed Plant Staff

The purpose of the non-licensed plant staff training program is to provide qualified personnel to operate and maintain the facility in a safe and efficient manner, as well as to keep the facility in compliance with its license, TS, and applicable regulations. The regulations in 10 CFR 50.120, "Training and Qualification of Nuclear Power Plant Personnel," require, in part, that a training program must be derived from a systems approach to training and must provide for the training and qualification of the following categories of nuclear power plant personnel: non-licensed operator, shift supervisor, shift technical advisor, instrument and control technician, electrical maintenance personnel, mechanical maintenance personnel, radiological protection technician, chemistry technician, and engineering support personnel. This training program is required by 10 CFR 50.120 to be in effect 18 months prior to fuel load. At present, all commercial nuclear plants in the U.S. have obtained training program accreditation via the National Academy for Nuclear Training as a means of complying with this requirement.

Detailed information for the non-licensed plant staff training program is not required by a DC or ML applicant. However, for an OL or COL application the staff review is focused on the applicant's detailed program, which should include the initial training, periodic retraining, and qualifications that are required for non-licensed plant staff. Power reactor facility licensees can commit to meeting RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," Revision 4 (ML19101A395) which endorses American National Standards Institute/American Nuclear Society (ANSI/ANS)-3.1-2014, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants," as a means of complying with the personnel qualification requirements. Applicants can also commit to meeting the guidelines of NEI 06-13A, "Template for an Industry

Training Program Description," Revision 2, March 2009 (ML090910554), as a means of addressing training requirements.

Section 13.2.2, "Non-Licensed Plant Staff Training," Revision 4 (ML15006A129), of the SRP and DANU-ISG-2022-05, "Organization and Human-System Considerations" (ML23277A143) provide staff review guidance for these programs.

Current Framework for Reactor Operator Training and Requalification Programs

Similar to the non-licensed plant staff training program, the purpose of the reactor operator training and requalification programs is to provide licensed operators and senior operators to operate the facility in a safe and efficient manner, as well as to keep the facility in compliance with its license, TS, and applicable regulations. The requirements in 10 CFR Part 55, "Operators' Licenses," set criteria for the issuance of licenses to operators and senior operators of utilization facilities.

Detailed information describing these programs is not required by a DC or ML applicant. However, for an OL or COL review, these training programs should contain adequate format, attributes, and level of detail to demonstrate that the requirements of 10 CFR Part 55 will be met. This information should be submitted 18 months prior to fuel load. Applications should also address the use of a simulator and have plans for establishing either a plant-referenced or Commission-approved simulator by a timeframe that will support the licensing of operators. Similar to the training programs for non-licensed plant staff, applicants may also make commitments to meet the guidelines of RG 1.8 and NEI 06-13A and seek accreditation from the National Academy of Nuclear Training as a means of meeting the training related requirements of 10 CFR Part 55. An OL or COL applicant should also describe the licensed operator requalification program as required in 10 CFR 50.54(i-1) and 10 CFR 55.59, "Requalification," including the first anticipated requalification period set in 10 CFR 55.59(a)(1). The requalification program is required to be in effect within three months after issuance of an OL or the date the Commission makes the finding under 10 CFR 52.103(g).

Section 13.2.1, "Reactor Operator Requalification Program; Reactor Operator Training," Revision 4 (ML15006A035), of the SRP and DANU-ISG-2022-05 provide additional guidance for reactor operator training requirements for LWR and non-LWR applicants.

NOAK Strategy

For a NOAK application, it is assumed that development of a design-specific knowledge and abilities list, which is necessary to develop licensing examinations, as well as the approval of any modifications to the operator licensing and examination process (including establishing the technical justification of any related exemption requests, such as an exemption request from the 18-month requirement in 10 CFR 50.120), would have already occurred in conjunction with the first of a kind (FOAK) licensing process or have been developed during the DC or ML stage. Site-specific information required may, but will not necessarily, include crew complement (which may change, for instance, due to the number of units on site), response activities and training such as for fire protection, and site-specific systems. The NRC staff notes that the NRC previously endorsed NEI 06-13A as an option for addressing training programs. As such, a similar process could also be used for the NOAK framework.

Technical Specifications

Current Framework

The TS establish requirements for items such as safety limits, limiting safety system settings, limiting control settings, limiting conditions for operation (LCO), surveillance requirements, design features, and administrative controls. Section 182a. of the AEA requires applicants for nuclear power plant operating licenses to provide TS. The regulations in 10 CFR 50.36, "Technical specifications," and 10 CFR 50.36a require that each applicant for a license authorizing operation of a production or utilization facility include in its application proposed TS. This regulation also requires the TS to include LCOs and defines LCOs as the lowest functional capability or performance levels of equipment required for safe operation of the facility. The regulation requires that when an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor, or follow any remedial actions permitted by the TS until the condition can be met.

Standard TS (STS) are published as NUREG-series publications. The STS are modified through the NRC staff approval of "Travelers" typically submitted by the industry's Technical Specifications Task Force. The purpose of the traveler program is to minimize industry and NRC time and effort by providing a streamlined review and approval of STS changes. The NRC encourages licensees to upgrade their TSs consistent with the criteria in the policy statement and conforming, to the practical extent, to the latest revision with incorporated Travelers of the improved STS. Following the NRC staff's approval of the traveler, licensees may submit a license amendment request to adopt the traveler.

DANU-ISG-2022-08, "Risk-Informed Technical Specifications" (ML23277A146) describes methods acceptable to the NRC staff for an applicant to prepare proposed TS using a risk-informed evaluation process. At the CP, DC, or ML application stage, some numerical values, graphs, and other data are not as complete as necessary for plant operation because determination of specific numerical values is pending future decisions by the OL or COL applicant on selection and procurement of hardware after issuance of the CP, DC, or ML. A DC application may describe COL action items related to the generic TS to be denoted by square brackets in the proposed generic TS and associated bases with appropriate guidance to COL applicants for completing COL action items. At the OL or COL application stage, as-procured or site-specific information (denoted by brackets in the reference DC (i.e., generic DCD) or ML TS) must be replaced with the final operational information, which must be in conformance with the FSAR. For a COL application referencing a DC, this information is in the plant-specific DCD.

For a DC application, the applicant should provide generic TSs to confirm that they will preserve the validity of the plant design, as described in the DCD, by ensuring that the plant will be operated (1) within the required conditions bounded by the DCD and (2) with operable equipment that is essential to prevent postulated design-basis events or mitigate their consequences. For an ML application, the applicant should propose TSs in a similar manner to those provided in a DC application.

NOAK Strategy

The DC and ML regulations require that an applicant provide proposed TS prepared in accordance with 10 CFR 50.36 and 10 CFR 50.36a. As stated above, the generic TSs are provided to ensure that the validity of the plant design is preserved.

There are some provisions that may be more likely to not be known at the design stage for a DC or ML, such as information on operating organization and management, and site location. A DC or ML applicant could provide a proposed program with few site-specific gaps that a future COL applicant could adopt if they pursued standardized TS. For the site location, staff could verify that information is provided in the TS, if needed, during the CP/OL or COL review, as the applicant should be selecting a site within the bounds of the design. There are also optional programs in the STS for which a DC or ML applicant could provide a proposed program that a CP/OL or COL applicant may or may not want to pursue (e.g., risk-informed completion time program and setpoint control program). Lastly, the NRC staff note that there have been years of operating experience with TSs and that modeling TS on the operating fleet STS (e.g., format and content), as appropriate, could make the staff's review more efficient for the NOAK framework.

<u>Security</u>

The licensee security programs deal with threats, thefts, and sabotage relating to special nuclear material, high-level radioactive wastes, nuclear facilities, and other radioactive materials and activities that the NRC regulates. The security programs that apply to nuclear power plant applicants and licensees are specified in 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage."

Current Framework for Physical Security

Current staff review guidance associated with physical security is provided in Section 13.6, "Physical Security" (ML18344A041), of the SRP which establishes criteria that the NRC staff uses in evaluating whether an applicant or licensee meets NRC regulations to construct and operate nuclear power plants. Section 13.6 points to other sections of the SRP, as appropriate. For instance, a DC application contains certain security elements as outlined in Section 13.6.2, "Physical Security – Review of Physical Security System Designs – Standard Design Certification and Operating Reactor Licensing Applications" (ML14140A210), of the SRP. An applicant may be relieved of containing these security elements in their application if one of more of the following criteria apply: (a) Physical security systems or functions will not be located within the nuclear island or structures, or (b) Physical security systems will not be integral to the construction of the nuclear island and structures (e.g., independent of building structure, portable, etc.). A nuclear island may not exist for micro-reactors due to the reactor design. The physical security program is required prior to fuel receipt. The ARCAP roadmap also provides a discussion on security plans.

Current Framework for Cybersecurity

The cybersecurity program is an integral part of each current nuclear power plant licensee's physical protection program. A licensee's physical security programs must comply with the performance objectives and requirements set forth in 10 CFR 73.55, and 10 CFR 73.55(b)(8) requires licensees to establish, maintain, and implement a cybersecurity program in accordance with 10 CFR 73.54, "Protection of digital computer and communication systems and networks." The current power reactor cybersecurity requirements in 10 CFR 73.54 require applicants for an OL under the provisions of 10 CFR Part 50 and holders of a COL under the provisions of 10 CFR Part 52 to address the security of digital computer and communication systems and networks in their cybersecurity plans. This program is required prior to receipt of fuel on site.

Under the existing cybersecurity framework, a licensee's cybersecurity program must provide reasonable assurance that digital computer and communication systems and networks associated with safety-related functions, functions important to safety, security, and emergency preparedness are adequately protected against cyberattacks, up to and including the design-basis threat as described in 10 CFR 73.1, "Purpose and scope." While the cybersecurity requirements in the existing regulation were developed for the current fleet of large LWRs and do not specifically address standardization of advanced reactors such as micro-reactors, the performance-based nature of the regulation allows considerations for standardization of the cybersecurity program. Licensees and applicants are encouraged to use existing NRC or NEI guidance which contain templates for developing a cybersecurity plan to comply with the licensing requirements of 10 CFR 73.54. Section 13.6.6, "Cyber Security Plan" (ML102630477), of the SRP and the ARCAP roadmap also provide discussion on cybersecurity plans.

Current Framework for Access Authorization

The existing access authorization (AA) requirements in 10 CFR 73.55, 10 CFR 73.56, "Personnel access authorization requirements for nuclear power plants," and 10 CFR 73.57, "Requirements for criminal history records checks of individuals granted unescorted access to a nuclear power facility, a non-power reactor, or access to Safeguards Information," provide reasonable assurance that individuals subject to access authorization programs are trustworthy and reliable, so as not to constitute an unreasonable risk to public health and safety or the common defense and security, including the potential to commit radiological sabotage, regardless of the reactor technology.

An applicant or licensee should consider the guidance in Section 13.6.4, "Access Authorization Operational Program" (ML15226A009), of the SRP for determining AA measures prior to fuel entering the protected area, when preparing an application for a COL or early site permit (ESP) under 10 CFR Part 52, or a CP under 10 CFR Part 50 while transitioning into their operational program. These licensing documents for the AA program should provide reasonable assurance until nuclear fuel is in the protected area and the requirements under 10 CFR 73.56 are applicable for maintaining high assurance that the individuals subject to this section are trustworthy and reliable, such that they do not constitute an unreasonable risk to public health and safety or the common defense and security, including the potential to commit radiological sabotage.

Current Framework for Fitness For Duty

The performance objectives of 10 CFR Part 26, "Fitness For Duty Programs," provide reasonable assurance that: (1) individuals are trustworthy and reliable as demonstrated by the avoidance of substance abuse; (2) individuals are not under the influence of any substance, legal or illegal, or mentally or physically impaired from any cause, which in any way adversely affects their ability to safely and competently perform their duties; (3) measures are established and implemented for the early detection of individuals who are not fit to perform their duties; (4) the construction site is free from the presence and effects of illegal drugs and alcohol; and (5) the workplaces are free from the presence and effects of illegal drugs and alcohol.

An applicant or licensee should consider the guidance in Section 13.7, "Fitness For Duty – Introduction" (ML15111A091), of the SRP for determining the programmatic requirements that apply to a nuclear power reactor when preparing an application for a COL or ESP under 10 CFR Part 52, or a CP/OL under 10 CFR Part 50. Additional fitness for duty (FFD) programmatic requirements apply as a reactor under construction progresses to completion and fueling. The type of reactor licensed under 10 CFR Part 50 or 10 CFR Part 52 would not impact the FFD program requirements applicable during construction or reactor operations. The ARCAP roadmap guidance also provides a discussion on the FFD program.

NOAK Strategy

Design standardization at the DC or ML stage could enable the deployment of micro-reactors of a common design with a streamlined NRC review and approval of operational programs (i.e., targeted to site-specific features) at the CP/OL or COL stage. For an applicant to standardize its security program at the DC or ML level, additional information from that required in 10 CFR Part 52 Subpart B and Subpart F would need to be voluntarily submitted by an applicant. Currently, a DC applicant may submit security information as part of the application according to the guidance provided in Section 13.6.2 of the SRP, but there is no similar guidance that currently exists for a ML applicant to provide security information to the NRC as part of the application. Additionally, security plans (i.e., physical security plan, training and qualification plan, safeguards contingency plan, cybersecurity plan) are not required to be submitted for review until the OL or COL phase of the licensing process.

The level of approval that can be achieved in a DC or ML is entirely dependent on what is sought by the applicant for the DC or ML and the completeness and quality of information that the applicant is able to provide at the time of requesting NRC approval. Staff note that currently, the format and content of the security plans may conform to the most recent revision of the NRC-accepted NEI 03-12, "Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, [and Independent Spent Fuel Storage Installation Security Program]." As stated in Section 13.6.1, "Physical Security - Combined License and Operating Reactors" (ML17291B265), of the SRP bracketed text identified in NEI 03-12 is intended to act as a placeholder for each applicant to address and provide additional details of specific proposed physical protection measures, to account for site-specific conditions and to ensure understanding of how the licensee intends to meet certain Commission requirements. Applicants could use this template to inform types of site-specific information that might be needed and provide that up front during the design phase. Additionally, if an applicant is referencing a FOAK CP/OL or COL micro-reactor application, the more information that is provided, reviewed, and approved, the greater the potential for streamlining follow on NOAK micro-reactor applications.

Staff note that several geographic factors may impact the security posture of a facility and thus the standardization of the program. For example, a single micro-reactor versus multiple micro-reactors in the same localized area may change the risk profile of the identified site. If security is initially designed at a specific site for a single micro-reactor, adding additional micro-reactors may alter the radiological consequence and ultimately the risk profile. Pending an analysis from the applicant, the results have the potential to require additional security measures at the determined location.

A micro-reactor geographically located in a remote location may have a low population zone over a substantial distance, while a micro-reactor located in a more densely populated location may not have this option. The size of the low population zone could impact the security measures required to maintain adequate protection. Additionally, the geographic location could impact the response capability of local law enforcement or other offsite armed responders based upon the number of responders available and the time it would take the responders to react to a security event. As proposed in SECY-22-0072, "Proposed Rule: Alternative Physical Security Requirements for Advanced Reactors" (ML21334A003), and SECY-23-0021, "Proposed Rule: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors" (Part 53) (ML21162A093), if the applicant states in their security plans that they commit to relying on local law enforcement or other offsite armed responders as the sole outfit for response to a security event, the response capability may be dependent on the location of the site.

The design of the security for a micro-reactor must factor in these considerations but it could be possible to develop standardized security using geographic bounding conditions that applicants must meet based on their specific situation in order to utilize a standardized security plan template. To achieve this objective, the security application content will need to be based on a geographic model that addresses these concerns, which would minimize the amount of site-specific information in an application. The NRC review and approval of security plan templates and features for a given geographical location could be accomplished to the maximum extent practicable during consideration of a DC or ML application. For example, a micro-reactor applicant could submit a security plan template for a standard design to the NRC for review, which would specify the bounding conditions about the geographic location (i.e., the security needed based on radiological consequence for a micro-reactor in a remote location vs. a densely populated location, and the ability to rely on local law enforcement as a reliable asset for the interdiction and neutralization function). Once the NRC has endorsed the template and any associated guidance, the application for any given deployment site would have a security plan based on the template and finalized with site-specific information.

Regarding cybersecurity, staff notes that historically plants had distinct network topologies. If a NOAK micro-reactor applicant's defensive computer security architecture (DCSA) deviates from the previously NRC-approved DCSA through a DC, ML, or a FOAK COL or CP/OL proceeding, it may present a decrease in the efficiency of the NOAK licensing by requiring an additional NRC staff review. For example, a FOAK DCSA has a boundary data communication device that separates and protects critical digital assets of a higher security level from a lower security level; changing the same boundary data communication device in the DCSA of a NOAK may warrant an analysis, to assess the impact of the change to the overall effectiveness of the defense-in-depth protective strategy. Staff notes that preapplication engagement activities would facilitate early discussions between the staff and developers to gain a better understanding of any site-specific conditions that may influence changes in the DCSA.

The NRC staff is cognizant that evolving technology and emerging threats may also present a challenge to efficient NOAK licensing that may require an additional analysis and NRC staff review. For instance, if a COL applicant chooses to deviate from previously approved technology, the applicant should provide sufficient information and analyses to demonstrate that the implementation is safe and does not decrease the overall effectiveness of the defense-in-depth protective strategy. Staff expects concepts such as security by design applied to FOAK micro-reactors to facilitate more efficient licensing of NOAK micro-reactors especially in the cybersecurity program. As mentioned, applicants may also use templates to comply with the regulations. The templates could be considered as a method of standardizing. It is likely that the standardization and protection against the disruption or malicious control for micro-reactors will rely heavily on DCSA, as described in the NRC and NEI guidance, and fully understanding the architecture of a FOAK micro-reactor will be essential in providing insights for standardizing a NOAK.

Material Control and Accounting

Current Framework

The NRC's regulations require the licensee to maintain a nuclear material control and accounting (MC&A) program that tracks and verifies special nuclear material (SNM) that is on site. The MC&A regulations ensure that the information collected by the licensee about SNM is accurate, authentic, and sufficiently detailed to enable a licensee to 1) maintain current knowledge of its SNM and 2) manage its program for securing and protecting SNM. The MC&A program, together with physical protection of facilities and information security requirements, make up the primary elements of the NRC's SNM safeguards program. The MC&A component of the larger safeguards program helps ensure that SNM within a licensed facility is not stolen or otherwise diverted from the facility. A licensee's MC&A program should: provide accurate, complete, up-to-date, and reliable information about quantities and precise locations of the facility's SNM; maintain control over the SNM to ensure continuity of knowledge, thus enhancing the ability to detect unauthorized removal; detect loss, attempted or actual theft of SNM; investigate and resolve any indications of loss or attempted theft of SNM; and maintain sufficient information to aid in recovery of missing SNM if loss or theft has actually occurred.

The regulations in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," provide the requirements for facilities that are authorized for or are seeking a license to possess and use more than a critical mass of SNM.[2] All licensees authorized to possess SNM in quantities greater than 1 gram must meet the applicable regulations of 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material." Subpart B, "General Reporting and Recordkeeping Requirements," of 10 CFR Part 74 contains general reporting and recordkeeping requirements for licensees authorized to possess 1 gram or more of SNM. Specific control and accounting requirements are provided in subparts C, D, and E[3] of 10 CFR Part 74 for certain licensees based on the category of the SNM.

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2 In 10 CFR 70.4, "Definitions," *critical mass of special nuclear material* means SNM in a quantity exceeding 700 grams of contained uranium-235; 520 grams of uranium-233; 450 grams of plutonium; 1500 grams of contained uranium-235, if no uranium enriched to more than 4 percent by weight of uranium-235 is present; 450 grams of any combination thereof; or one-half such quantities if massive moderators or reflectors made of graphite, heavy water, or beryllium may be present.

3 Subpart C – Special Nuclear Material of Low Strategic Significance, also known as Category III; Subpart D – Special Nuclear Material of Moderate Strategic Significance, also known as Category II; and Subpart E – Formula Quantities of Strategic Special Nuclear Material, also known as Category I.

Facilities licensed under 10 CFR Part 50 or 10 CFR Part 52 are required to account for and control their SNM as part of a condition of obtaining a license. In general, for both 10 CFR Part 50 and 10 CFR Part 52 applicants and licensees, 10 CFR Part 74 Subpart B (excluding 10 CFR 74.17, "Special nuclear material physical inventory summary report") contains the applicable MC&A requirements. Regulatory Guide 5.29, "Special Nuclear Material Control and Accounting Systems for Nuclear Power Plants" (ML13051A421), provides ANSI N15.8-2009, "Material Control Systems - Special Nuclear Material Control and Accounting Systems for Nuclear Power Plants," as an acceptable approach for complying with the NRC's regulations regarding MC&A requirements in Subpart B of 10 CFR Part 74 at nuclear power plants (NPP). However, ANSI N15.8-2009 is intended for LWRs that utilize low-enriched uranium oxide fuel. Applicants for other types of reactors utilizing designs and fuels that may differ from that described in ANSI N15.8-2009 may be subject to additional MC&A requirements.

NOAK Strategy

Preliminarily, if a micro-reactor maintains SNM at category II levels or below in discrete items, either in fuel assemblies or a packaged core, then these micro-reactors would be subject to the same regulatory framework as NPPs. MC&A requirements are applicable for these reactors at the fixed site in which they are operated and includes documenting the transfer of SNM during shipment. No detailed fundamental nuclear material accounting plan would need to be submitted; however, a plan to implement the 10 CFR Part 74 Subpart B requirements should be described in the license application and would need to be in place prior to receipt of material. If the micro-reactor design can account for SNM through item-counting, then item count and identification check may be sufficient for physical inventory if continuity of knowledge of the manufacturers or shippers' values for SNM is maintained, as per the guidance in ANSI-15.8-2009.

MC&A plans are not necessarily site-specific. For NPPs (and micro-reactors) a description of how the facility will meet the requirements of 10 CFR Part 74 Subpart B is generally submitted in the application for an OL. This can, in principle, be standardized across any specific design of micro-reactors, and submitted as part of the design approval in either a DC or ML. Of note, a plant at which micro-reactors are fabricated under an ML is subject to the requirements of 10 CFR Part 70 and 10 CFR Part 74 only if the licensee also holds a 10 CFR Part 70 license authorizing possession of SNM in the requisite quantity.

<u>Emergency Planning</u>

Current Framework

The purpose of emergency preparedness (EP) is to ensure adequate protective measures can and will be taken in the event of a radiological emergency. EP is not a design feature of a particular technology or reactor design; rather, it is an operational program that provides defense-in-depth through effective planning to reduce radiation dose to the public beyond what is accomplished by design and overall reactor safety. The NRC's EP regulations are not design specific; they are intentionally design inclusive. Emergency plans must demonstrate compliance with required planning standards. These planning standards are similar across reactor facility types, as the operational aspects of EP are well established in regulation and in practice across all hazards. Emergency planning functions are capabilities or resources to prepare for, and respond to, a radiological emergency. The broad language of planning standards offers flexibility to address plant-specific and site-specific planning considerations in the planning functions.

The requirements in 10 CFR 50.47, “Emergency Plans,” and Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” provide planning standards which were originally developed for the hazards and emergency planning needs for large LWRs. The requirements in 10 CFR 50.160, “Emergency preparedness for small modular reactors, non-light-water reactors, and non-power production or utilization facilities,” provide alternative risk-informed, performance-based planning standards for small modular reactor or non-LWR applicants. Micro-reactor applicants may choose to use 10 CFR 50.160 for the efficient licensing and deployment of new micro-reactor technologies. However, micro-reactor applicants could choose to use the planning standards in 10 CFR 50.47 and Appendix E to establish an appropriate level of planning, including use of the provisions in 10 CFR 50.33(g)(1) for determining the size of the emergency planning zone on a case-by-case basis for reactors with authorized power levels of less than 250 MWt. The regulations in 10 CFR 50.47(a) describe the findings that must be made by the NRC before issuance of an OL under 10 CFR Part 50, or a COL or ESP under 10 CFR Part 52. In addition, under 10 CFR Part 52, an applicant for an ESP may provide complete and integrated plans or major portions of emergency plans for review. The NRC issues an OL or COL if it determines that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency and all other applicable requirements are satisfied.

NOAK Strategy

Many aspects of a micro-reactor EP program could be reviewed in a DC or ML application. The review consists of evaluating whether the proposed emergency planning functions provide the resources or capabilities necessary to demonstrate compliance with the required planning standards. The ability to review and approve emergency plans at the DC/ML stage is subject to the quality and completeness of information submitted by the applicant in regard to elements of the plan. Approval of any portion of the emergency plan at the DC or ML stage does not make EP part of the design and subject to design standards. Approved portions of the emergency plan could be incorporated into a license application but remain part of the EP operational program.

The staff has identified three primary considerations in addressing the standardization of EP for NOAK reactors.

<u>Site-Specific Considerations to Standardize EP Approaches</u>

Commensurate with the radiological risks and hazards of the facility, many of the emergency planning functions for the emergency plan could be standardized and reviewed at the DC/ML stage. In this context “standardized” refers to the standardization of methods and means used to accomplish a planning function or required element of the emergency plan. To the extent practicable, approval could be granted to standardized planning functions that are not site-specific. Examples could include onsite emergency response organization staffing, emergency action levels, accident assessment capabilities, and emergency response facilities and equipment that would be standard for the design and operation of the facility at any site. Some planning functions require site-specific planning elements that cannot be either standardized or established until a specific site is identified, such as site-specific memorandums or agreements with state, local, tribal governments and evacuation time estimates (if needed).

In cases where the emergency planning zone (EPZ) extends beyond the site boundary and additional offsite planning elements are required, the plans to meet those requirements would be site-specific and would involve Federal Emergency Management Agency (FEMA) findings and determinations that the plans are adequate and capable of being implemented. These FEMA findings and determinations cannot be addressed as part of the DC or ML application and therefore would need to be included in the ESP/COL/OL. If the applicant demonstrates that the plume exposure pathway EPZ does not extend beyond the site boundary then some site-specific requirements may not be applicable.

Some elements of an emergency plan could be approved as part of a DC or ML with placeholders for site-specific information. For example, identification of principal response organizations and assistance resources could be included as ITAAC or license conditions and subject to verification. The ITAAC closure verification confirmation process for 10 CFR Part 52 license applications or use of license conditions for 10 CFR Part 50 license applications as part of an OL could be used as tools to approve an emergency plan in connection with a DC or ML application.

Efficiently Addressing EPZ Sizing Determinations

The established planning basis for EP is an evaluation of the consequences from a spectrum of accidents to scope the planning efforts for the distance, time, and materials released.
The planning distance, or EPZ, establishes the physical bounds for which emergency planning is in place to assure predetermined, prompt protective measures can be taken. In addition, the detailed planning within the EPZ provides a basis for expansion of the response efforts beyond the EPZ, should it become necessary during an actual emergency.

The staff anticipates that many NOAK micro-reactor facilities will be planned with the goal of having a site-boundary EPZ, or no EPZ. A site-boundary EPZ, or no EPZ, could be approved at the DC or ML stage if sufficient and complete information is provided for the analysis used in the EPZ determination. Such an analysis would likely make use of bounding assumptions for site-specific factors like meteorology and consequences of event scenarios like seismic and security events. Efficiency would be gained by using information developed for other parts of the licensing basis to inform the spectrum of accidents and provide source terms and consequence analyses; this includes considerations of licensing basis events, acts of sabotage, and credit for mitigation capabilities. An applicant would need to demonstrate in the subsequent COL application that the NOAK facility remains bounded by the consequences and assumptions approved for the design at the DC/ML stage. An applicant may also seek approval of a methodology to determine EPZ sizing, such as through a topical report that an applicant could use in a site-specific COL application to improve efficiency. That could streamline determination of the site-specific plume exposure pathway EPZ size and NRC approval of site-specific COL applications incorporating a DC/ML design.

Deployment of Multiple NOAK Reactors at the Same Site

There are additional considerations with potential scenarios in regard to multiple NOAK reactors being deployed at the same site in a phased manner. A designer/manufacturer could address in a DC or ML application the deployment of multiple reactors at a common site and explicitly state that number of reactors. The bounding safety analysis and hazard analysis would need to account for the radiological risks and hazards associated with any additional reactors desired in a NOAK application for that site. In addition, efficiencies in using existing emergency planning functions to meet the planning standards could be gained for multi-reactor sites. Common resources and capabilities like emergency response facilities and equipment could be shared, similar to the approach taken by many large LWRs operating on the same site.

The DC or ML application would therefore need to specifically consider and analyze relevant multiple reactor scenarios. A COL applicant that requests additional reactors beyond that assessed in the standardized emergency plan approved in the DC or ML stage would be considered a departure due to the potential for differing site-specific hazard conditions. This would impact the regulatory efficiency of NOAK reactor licensing timeliness for EP approvals of multiple reactor deployment scenarios.