U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Advanced Reactor Stakeholder Public Meeting

March 2, 2023

[Microsoft Teams Meeting]
Bridgeline: 301-576-2978
Conference ID: 417 405 578#

Advanced Reactor Program

Time	Agenda	Speaker
10:00 am – 10:15 am	Opening Remarks / Adv. Rx Integrated Schedule / Update on SCALE/MELCOR Advanced Reactor Source Term Demonstration Project	NRC
10:15 am – 10:55 am	Advanced Reactor Construction Oversight Program (ARCOP)	NRC
10:55 am – 11:30 am	Advance Contracting Requirement Under Section 302(b) of the Nuclear Waste Policy Act	NRC / DOE
11:30 pm – 12:00 pm	Micro-Reactor Deployment Policy Topics	NRC
12:00 pm – 1:00 pm	Lunch Break	All
1:00 pm – 1:30 pm	Transportation and Storage for Advanced Reactor Fuel and Transportable Micro-Reactors	NRC
1:30 pm – 1:45 pm	Guidance for Reviewing a Non-Power Liquid Fueled Molten Salt Reactor License Application	NRC
1:45 pm – 2:00 pm	Pre-Application Engagement on Materials Qualification Issues for Advanced Reactor Licensing	NRC
2:00 pm – 2:30 pm	Advanced Reactor Materials Interim Staff Guidance	NRC

Time	Agenda (continued)	Speaker
2:30 pm – 2:45 pm	Break	All
2:45 pm – 3:30 pm	Status of Two Draft Regulatory Guides on RIPB Seismic Design and Seismic Isolation for Commercial Nuclear Powerplants	NRC
3:30 pm – 3:35 pm	Future Meeting Planning and Concluding Remarks	NRC

Advanced Reactor Integrated Schedule of Activities

The updated Advanced Reactor Integrated Schedule

is publicly available on NRC Advanced Reactors website at:

https://www.nrc.gov/reactors/new-reactors/advanced/integrated-review-schedule.html

Advanced Reactor Integrated Schedule of Activities

- Strategy 1 — Knowledge, Skills, and Capability
- Strategy 2 — Computer Codes and Review Tools
- Strategy 3 — Guidance
- Strategy 4 — Consensus Codes and Standards
- Strategy 5 — Policy and Key Technical Issues
- Strategy 6 — Communication

Legend

- Concurrence (Division/Interoffice)
- Federal Register Publication
- Public Comment Period
- Draft Issuance of Deliverable
- Final Issuance of Deliverable
- ● EDO Concurrence Period
- Commission Review Period**
- ▼ ACRS SC/FC (Scheduled or Planned)
- External Stakeholder Interactions
- ↓ Public Meeting (Scheduled or Planned)

Version 2/27/23

Present Day

Strategy	Regulatory Activity	Commission Papers	Guidance	Rulema	NEIMA	Comple	2022												2023											
							Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
1	Development of non-Light Water Reactor (LWR) Training for Advanced Reactors (Adv. Rxs) (NEIMA Section 103(a)(5))					x																								
	FAST Reactor Technology				x	x																								
	High Temperature Gas-cooled Reactor (HTGR) Technology				x	x																								
	Molten Salt Reactor (MSR) Technology				x	x																								
	Competency Modeling to ensure adequate workforce skillset					x																								
	Identification and Assessment of Available Codes					x																								
	Development of Non-LWR Computer Models and Analytical Tools																													
	Reference plant model for Heat Pipe-Cooled Micro Reactor					x																								
	*Reference plant model for Sodium-Cooled Fast Reactor (update from version 1 to 2)****																													
	*Reference plant model for Molten-Salt-Cooled Pebble Bed Reactor (update from version 1 to 2)****					x																								
	Reference plant model for Monolith-type Micro-Reactor																													
	*Reference plant model for Gas-Cooled Pebble Bed Reactor (update from version 1 to 2)****					x																								
	*Reference plant model for Molten-Salt-Fueled Thermal Reactor (update from version 1 to 2)****																													
	Code Assessment Reports Volume 2 (Fuel Perf. Anaylsis)					x																								
	FAST code assessment for metallic fuel					x																								
	FAST code assessment for TRISO fuel					x																								
	Code Assessment Reports Volume 3 (Source Term Analysis)					x																								
	Non-LWR MELCOR (Source Term) Demonstration Project					x									↓															
	Reference SCALE/MELCOR plant model for Heat Pipe-																													

https://www.nrc.gov/reactors/new-reactors/advanced/integrated-review-schedule.html

Update on SCALE/MELCOR Advanced Reactor Source Term Demonstration Project

- Developed new SCALE and MELCOR modeling capabilities for five non-light water reactor designs (2021-2022)
  - Held workshops that included sample accident simulations
  - Workshop documentation is now available on NRC's advanced reactor source term website
- Held workshop on applying SCALE and MELCOR to the TRISO fuel cycle (February 28, 2023)
- Will develop and demonstrate targeted model improvements (2023)

SCALE/MELCOR non-LWR source term demonstration project	
•Heat-pipe reactor workshop on June 29, 2021 • Slides • Video Recording •SCALE report •MELCOR report	June 29, 2021
•High-temperature gas-cooled reactor workshop on July 20, 2021 • Slides • Video Recording •SCALE report •MELCOR report	July 20, 2021
•Fluoride-salt-cooled high-temperature reactor workshop on September 14, 2021 • Slides • Video Recording •SCALE report •MELCOR report	September 14, 2021
•Molten-salt-fueled reactor workshop on September 13, 2022 • Slides • Video Recording •SCALE report	September 13, 2022
•Sodium-cooled fast reactor workshop on September 20, 2022 • Slides • Video Recording	September 20, 2022

Advanced Reactor Construction Oversight Program

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

"… To develop the best oversight program possible that ensures safety and security, considers the diversity of technology and its risk profile, adapts to facility-specific insights, and leverages our collective experience, while remaining adaptable to respond to future opportunities and challenges.

- ARCOP Challenge



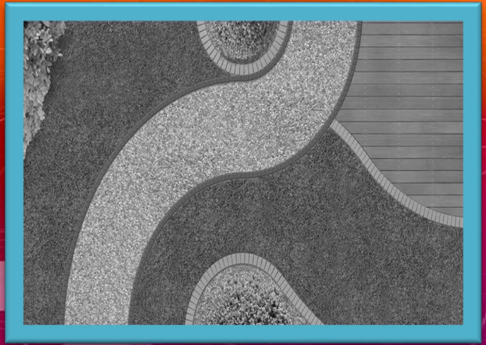
Objective

- WHAT: Provide reasonable assurance that advanced reactor plants are built and will operate in accordance with their licenses and applicable laws and regulations, thus adequately protecting the public and environment
- HOW: Leverage an oversight program that is comprehensive, scalable, innovative, risk-informed, performance based, and technology-inclusive

What is New in ARCOP?

- Project-specific inspection scope
- Scalable inspection scope commensurate with performance
- Focus on QA performance in different construction areas
- Scalable inspection footprint including role of construction resident inspector
- Streamlined significant determination process commensurate with facility risk
- Performance assessment includes short term assessment for timely reaction to emergent issues
- Explore the use of 3rd party performance monitoring data

Key Considerations in ARCOP Development



Reactor Technology/ Attributes

- Different coolants, fuel, materials & design codes/standards
- Co-location with fuel facilities
- Wide range of sizes
- Enhanced safety margin/risk profile

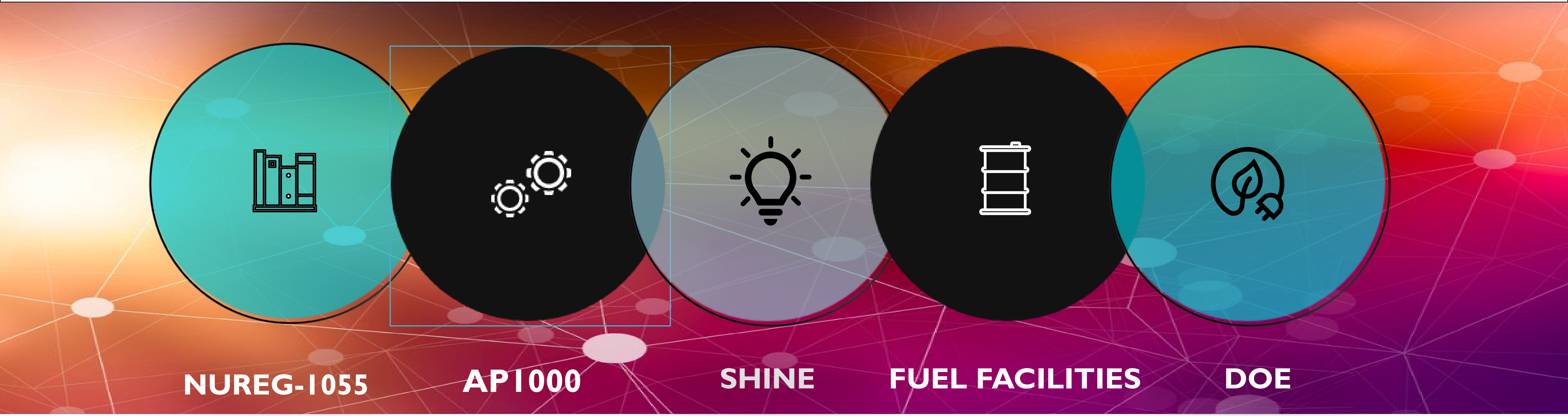
Manufacturing/ Construction Techniques

- Greater use of factory fabrication

Licensing Pathway

- Parts 50, 52 &53
- Different information may be available during construction
- Will ensure consistency of oversight

Building on Oversight Experience



NUREG-1055	AP1000	SHINE	FUEL FACILITIES	DOE
Greater focus on quality assurance	Enable flexibility, integration, scalability & hybrid capabilities	Value of hybrid and flexible inspection scope	Greater focus on design control, procedures & procurement	Insights from advanced reactor construction

Performance Monitoring Enhancements

Scope

Technology/facility-specific
Anchored to fundamental safety functions
Considers risk-insights for reactor & SSCs
QA + direct SSCs inspections
Operational readiness & security

Schedule

Flexible, matches construction/manufacturing pace
Supports inspection at different locations
Enables frequent performance assessment & scope adjustment

3rd Party Oversight

Gain additional data
Reduce redundancy
Leverage international inspections
Credit Authorized Inspection Agencies

Hybrid Inspection

Enables flexibility without sacrificing quality
Optimizes inspection conduct
Leverages technology

Resources

Project-specific
Commensurate with facility complexity and size
Reflective of FOAK vs proven technology

RTR Inspection Insights

Apply ARCOP to near-term RTR construction and refine based on experience

Construction Inspection Matrix (example)

Fundamental Safety Functions	Procurement of ASME Qualified Piping	Manufacturing of Reactor Vessel & Internals	Construction of Steel & Concrete Buildings
Reactivity Control	SSC1, SSC2 inspection family	SSC3, SSC4	N/A
Decay Heat Removal	SSC1, SSC5	SSC3, SSC4, SSC6	SSC7, SSC8
Radioactive Material Retention	SSC1, SSC2	SSC9, SSC10	SSC11, SSC12

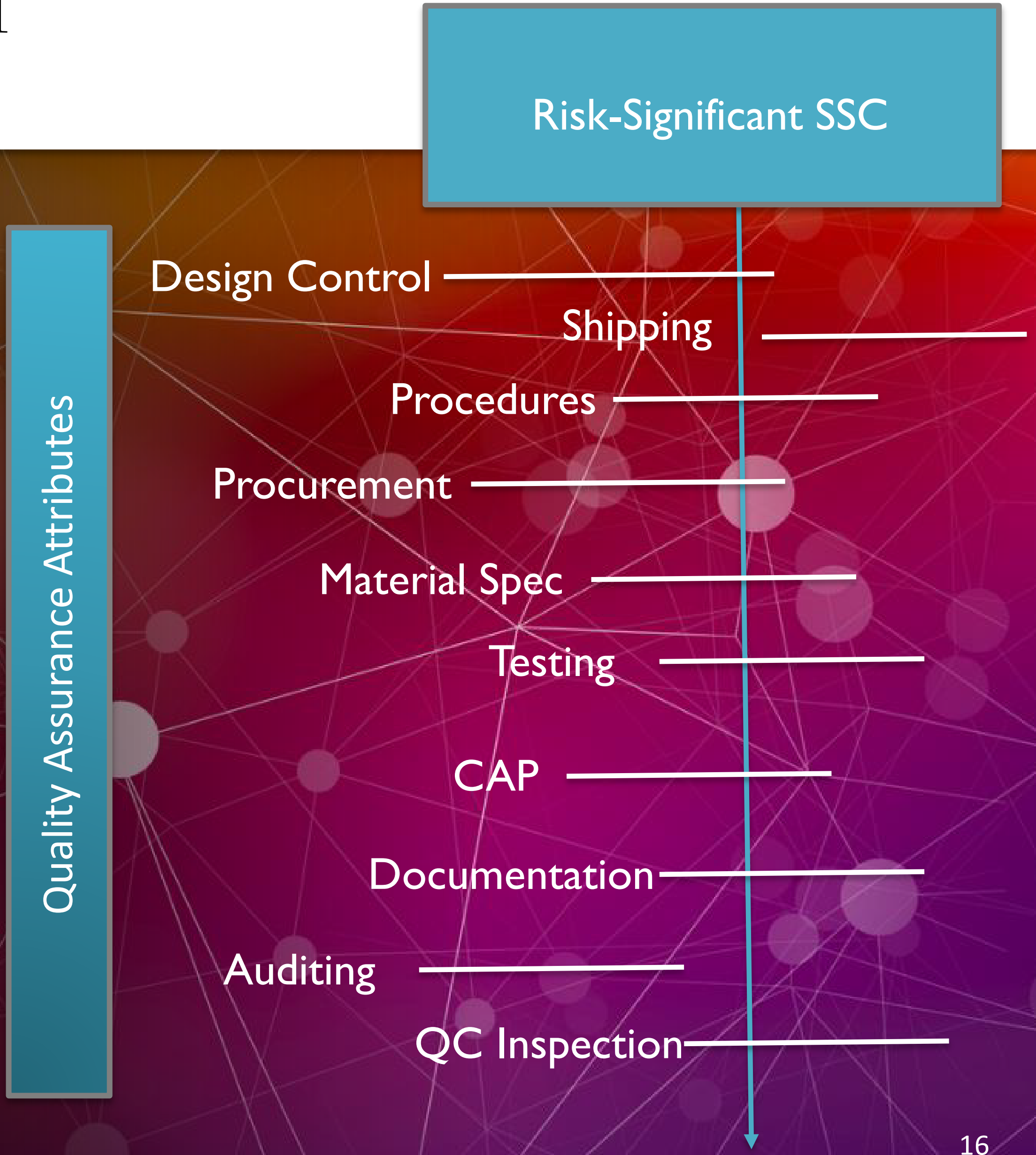
ARCOP Vertical Slice Inspection

01. In-depth inspection of QAP attributes associated with sampled SSCs.

02. Optimizes inspection strategies used for vendors and SHINE

03. Results inform assessment of construction area adequacy

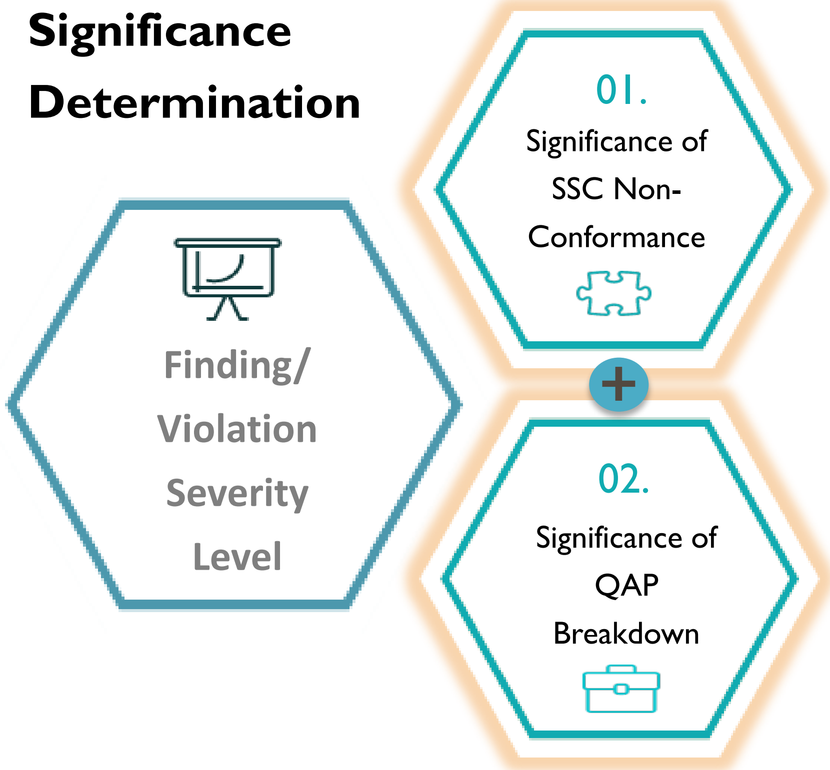
Choose risk significant SSC for inspection → Inspect SSC AND inspect applicable QAP attributes → Repeat for other SSCs in same construction area until reasonable assurance is attained for that area → Adjust baseline inspections as appropriate



Enforcement Enhancements

- Risk-informed
- Builds on well-established approaches
- Leverages general reactor safety criteria vs facility-specific quantitative risk assessment
- Significance determination effort commensurate with risk
- Appropriate level of detail to ensure clarity and consistency
- Quantitative SDP maybe used for risk profiles and system complexity approaching LLWRs

Significance Determination



Enforcement

SSC = Structure, System & Component
QAP = Quality Assurance Program

Performance Assessment Enhancements

What's new under ARCOP's performance assessment?

Strategic Areas

- Construction Quality
- Security Programs
- Operational Readiness

Cornerstones

- Quality of Suppliers' Activities
- Construction, Manufacturing, and Procurement
- Security Programs
- Operational Programs

Two-Tiered Approach

01. Licensee Assessment

- Based on severity level of findings/violations (similar to cROP)
- Informs supplemental & reactive inspections

02. QAP Assessment

- Assessment of QAP in each construction area
- Informs changes to baseline inspection plan
- Enables timely NRC and licensee response to performance deficiencies

ARCOP Development Timeline

2022
- Develop vision
- Develop information SECY
- Int./Ext. Communication

2023
- Address policy issues
- ARCOP guidance begins
- Update IMC 2550 for adv. RTRs
- SHINE OL

2024
- Hermes/ACU construction begins
- ARCOP guidance (con't)
- Possible LWA issued

2025
- ARCOP guidance issued
- Inspection organization est.
- Training

2026
- Construction begins (NLWR #1 & LWR-SMR #1)

Advance Contracting Requirement Under Section 302(b) of the Nuclear Waste Policy Act

Michael Kido

Office of the General Counsel, U.S. Department of Energy

March 2023

U.S. DEPARTMENT OF ENERGY

Nuclear Waste Policy Act of 1982 (as amended) (NWPA)

- Established the Federal responsibility for the disposal of spent nuclear fuel (SNF) and high-level radioactive waste (HLW).

- Assigned to DOE the responsibility of developing capabilities for disposal and, if necessary, consolidated interim storage (referred to in the NWPA as "monitored retrievable storage").

The Standard Contract - Background

- The “Standard Contract for Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste” (10 CFR Part 961) establishes the contractual terms and conditions under which DOE will make nuclear waste disposal services available to owners and generators of SNF and HLW (mostly nuclear utilities).

- The Standard Contract specifies the terms under which DOE will accept title to, transport and dispose of SNF and HLW from contract holders. It also provides for the payment of fees sufficient to offset DOE’s expenditures.

Section 302(b) - Background

- Section 302(b) of the NWPA lays out the advance contracting requirement for NRC license applicants.

- NRC cannot issue or renew a license to any person to use a “utilization or production facility under the authority of section 103 [Commercial Licenses] or 104 [Medical Therapy and Research and Development] of the Atomic Energy Act of 1954” unless such person has entered into a contract with DOE or DOE affirms in writing that such person is “actively and in good faith negotiating” with DOE for a contract.

DOE Office of Standard Contract Management

-. DOE's Office of Standard Contract Management manages these "Standard Contracts" and the Nuclear Waste Fund for DOE. The Office is housed within the Office of the General Counsel and continues DOE's core functions established by the NWPA pertaining to the Nuclear Waste Fund and the management of the Standard Contract.

- The Standard Contract and the Amendment to the Standard Contract for New Reactors are available at the following link under Applicable Documents on this office's website - https://www.energy.gov/gc/office-standard-contract-management.

Questions?

Contact Information

Michael Kido (DOE-OGC)
- Michael.Kido@hq.doe.gov

Cyrus Nezhad (DOE-OGC)
- Cyrus.Nezhad@hq.doe.gov

Connie Barton (Director, DOE Office of Standard Contract Management)
- Connie.Barton@hq.doe.gov

Advance Contracting Requirement Under Section 302(b) of the Nuclear Waste Policy Act – NRC Guidance

Joseph Sebrosky
NRR/DANU/UARP

Advanced Reactor Stakeholder Meeting
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U.S.NRC
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Nuclear Waste Policy Act of 1982, as amended

Section 302. Nuclear Waste Fund

(b) ADVANCE CONTRACTING REQUIREMENT-

(1)(A) The Commission shall not issue or renew a license to any person to use a utilization or production facility under the authority of section 103 or 104 of the Atomic Energy Act of 1954 (42 USC 2133, 2134) unless –

(i) such person has entered into a contract with the Secretary under this section; or

(ii) the Secretary affirms in writing that such person is actively and in good faith negotiating with the Secretary for a contract under this section.

Nuclear Waste Policy Act of 1982, as amended (continued)

(b) ADVANCE CONTRACTING REQUIREMENT [continued]-

(1)(B) The Commission, as it deems necessary or appropriate, may require as a precondition to the issuance or renewal of a license under section 103 or 104 of the Atomic Energy Act of 1954 (42 USC 2133, 2134) that the applicant for such license shall have entered into an agreement with the Secretary for the disposal of high-level radioactive waste and spent nuclear fuel that may result from the use of such license.

Source: NUREG-0980, Vol. 1, No. 10. ML13274A489.

Generic Letter No. 83-07 – The Nuclear Waste Policy Act of 1982

TO ALL POWER AND NON-POWER REACTOR LICENSEES, APPLICANTS FOR AN OPERATING LICENSE AND HOLDERS OF CONSTRUCTION PERMITS

Gentlemen:

SUBJECT: THE NUCLEAR WASTE POLICY ACT OF 1982 (Generic Letter No. 83-07)

On January 7, 1983, the Nuclear Waste Policy Act was enacted. The purpose of this letter is to ensure that you are aware of a provision of the Act (Section 302(b)) (copy enclosed) that requires licensed owners or generators of spent nuclear fuel or high-level waste to have a contract with the Secretary of Energy, by June 30, 1983, for the disposal of such waste.

This mandate applies to all facilities licensed under Sections 103 and 104 of the Atomic Energy Act of 1954. If a facility is to be licensed or have its license renewed before June 30, 1983, licensing is contingent on the existence of either a contract with the Secretary or a written affirmation by the Secretary that the owner/generator is actively and in good faith negotiating with the Secretary. For facilities to be licensed or have a license renewed after June 30, 1983, licensing is contingent on the existence of a contract.

- GL 83-07 is dated February 16, 1983
- Addressed to "ALL POWER AND NON-POWER REACTOR LICENSEES, APPLICANTS FOR AN OPERATING LICENSE AND HOLDERS OF CONSTRUCTION PERMITS"

NUREG-1537, Part 1, Rev. 1, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content”

- Published February 1996.
- Section 1.7, “Compliance With the Nuclear Waste Policy Act of 1982”

  The applicant should briefly discuss how it meets the requirements of Section 302(b)(1)(B) of the *Nuclear Waste Policy Act* of 1982 for disposal of high-level radioactive wastes and spent nuclear fuel. This discussion should include the contract arranged with DOE for return of the material. A copy of the cover letter for the contract between the applicant and DOE should be included in an appendix to the [safety analysis report].

Combined License Example

- In Section 1.5.2 of each safety evaluation report on an AP1000 combined license (COL) application, the staff evaluates compliance with Section 302(b) of the Nuclear Waste Policy Act
  - Example SER found at: https://www.nrc.gov/docs/ML1227/ML12271A045.pdf
- Similar staff evaluations can be found in Section 1.4.2 of the ESBWR COL safety evaluation (see https://www.nrc.gov/docs/ML1419/ML14198A557.pdf) and Section 1.5S.3 of the ABWR COL safety evaluation (see: https://www.nrc.gov/docs/ML1527/ML15271A126.pdf)

Questions?

Micro-Reactor Licensing and Deployment Topics

Advanced Reactor Stakeholders Meeting
March 2, 2023

William Kennedy
Amy Cubbage
Advanced Reactor Policy Branch
U.S. Nuclear Regulatory Commission

Introduction

- Goals of this presentation
- SECY-20-0093 summary
- NRC draft white paper on micro-reactor licensing strategies
- Licensing and deployment topics for factory-fabricated transportable micro-reactors
- Discussion items

Goals of this Presentation

- Inform stakeholders of the micro-reactor licensing and deployment topics currently being considered by the NRC staff for factory fabricated transportable micro-reactors
- Hear feedback from stakeholders, including other topics for consideration and thoughts on prioritization

SECY-20-0093 Summary

- SECY-20-0093[1] laid out several issues related to micro-reactor licensing and deployment, including information on the current regulations, applicability to micro-reactors, stakeholder perspectives, and NRC staff considerations
- Some issues are being addressed in ongoing rulemakings and guidance development, and some are topics for consideration for factory-fabricated transportable micro-reactors as described later in this presentation

[1] SECY-20-0093: Policy and Licensing Considerations Related to Micro-Reactors (https://www.nrc.gov/docs/ML2025/ML20254A363.html)

SECY-20-0093 Summary

- Security Requirements
- Emergency Preparedness
- **Staffing, Training, and Qualification Requirements**
- **Autonomous and Remote Operations**
- Regulatory Oversight
- Aircraft Impact Assessment
- Annual Fee Structure
- **Manufacturing Licenses and Transportation**
- Population-Related Siting Considerations
- Environmental Considerations

Micro-reactor Licensing Strategies

- NRC issued a draft white paper titled, “Micro-reactors Licensing Strategies,” to facilitate the development of optional strategies to streamline the licensing of micro-reactors (https://www.nrc.gov/docs/ML2132/ML21328A189.pdf)
  - Enhanced standardization of the design and operational programs
  - Manufacturing license may provide flexibility for design and fabrication in a factory and reduce site-specific inspections and verifications
  - Use of “bounding values” for external hazards and site characteristics could reduce NRC staff review effort
  - Generic Environmental Impact Statement for Advanced Nuclear Reactors (ANR GEIS) rulemaking

Licensing and Deployment Topics – Factory-Fabricated and Transportable Micro-Reactors

- The NRC staff is continuing to develop topics related to licensing and deployment of factory-fabricated transportable micro-reactors to identify policy issues and options to address them
- **Loading fuel at a manufacturing facility**

  Developers may propose loading fuel into reactors at the manufacturing facility either during or after the manufacturing process.
- **Qualifications for personnel handling fuel at a manufacturing facility**

  Loading fuel at a manufacturing facility would also require appropriately-qualified personnel to handle the fuel.
- **Timelines for ITAAC closure, hearings, and 52.103(g) findings**

  The process for beginning operation under combined licenses includes several steps with extended timeframes, such as ITAAC closure, the associated 52.103(g) finding, and the ITAAC hearing process (including the AEA 189a.(1)(B) requirement to provide notice of an opportunity for hearing at least 180 days before scheduled fuel load).

Licensing and Deployment Topics – Factory-Fabricated and Transportable Micro-Reactors

- **Licensing replacement of reactor modules**

  Deployment scenarios may involve delivering fueled micro-reactor modules to the power plant site and replacing the modules with some periodicity.

- **Low Power Physics Testing at a Manufacturing Facility**

  Developers may seek to load fuel and conduct low power physics testing at the manufacturing facility.

- **Transportation of fueled reactor modules**

  Reactor modules that are loaded with fresh, irradiated, or spent fuel might be transported between the manufacturing facility, operating power plant site, and a facility for refurbishing or decommissioning reactor modules.

Licensing and Deployment Topics – Factory-Fabricated and Transportable Micro-Reactors

- **Remote and autonomous operations**

  Micro-reactor developers might include capabilities for remote or autonomous operation and monitoring, including cybersecurity features, and propose not having on-site reactor operators.

- **Irradiated fuel and spent fuel**

  The definition of spent fuel (10 CFR Parts 71 and 72) includes criteria that fuel has been withdrawn from a nuclear reactor following irradiation and has undergone at least one year's decay since being used as a source of energy in a power reactor. Depending on how long it has been since the final reactor shutdown of a micro-reactor, different regulations may apply to the storage and transport of the reactor fuel or the fueled micro-reactor module.

- **Decommissioning process/funding assurance**

  Decommissioning transportable micro-reactors may involve independent regulated decommissioning of power plant sites as well as the reactor modules upon removal. Facility licensing and decommissioning licensing requirements may apply to developers who seek to use a centralized facility to decommission reactor modules away from power plant sites.

Additional Topics for Longer-Term Consideration

- **Mobile micro-reactors**

  The NRC staff is aware that deployment of mobile micro-reactors is of interest to some developers.

- **Maritime or space applications**

  The NRC staff is aware that maritime and space applications of micro-reactors may be of interest to developers.

Next Steps

- Stakeholder engagement
- Identify policy issues
- Consider options to address the issues
  - Guidance development
  - Rulemaking
- Draft White Paper to further stakeholder input
- Engage Commission as appropriate

Discussion Items

- Are there scenarios of interest that are not captured in this presentation?
- What do stakeholders see as the highest priority topics to address?
- Which regulatory topics pose the greatest risks to micro-reactor deployment?
- Other feedback or questions

Lunch Break

Meeting will resume at 1:00 pm EST

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 417 405 578#

Transportation and Storage for Advanced Reactor Fuel and Transportable Microreactors

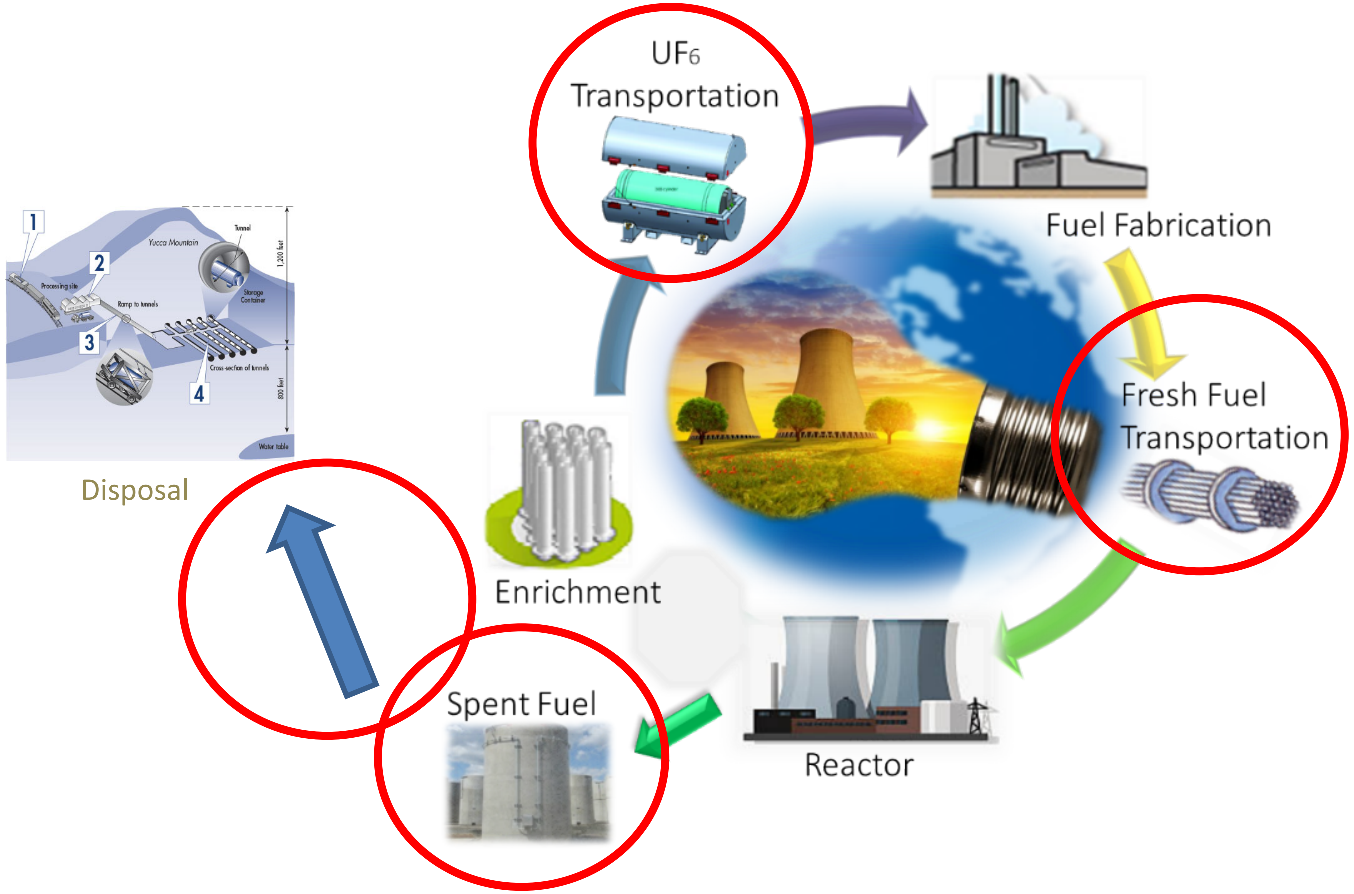
Advanced Reactor Stakeholder Meeting
March 2, 2023

Bernard White
Storage and Transportation Licensing Branch
Division of Fuel Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission

Introduction

- NRC is ready to review transport packages and spent fuel storage applications
  - Transportation package certification (10 CFR Part 71)
  - Spent fuel storage installations (10 CFR Part 72)
- NRC regulatory framework in 10 CFR Part 71 allows for the review of for advanced reactor fuel and transportable microreactors
- NRC approved transportation packages and storage systems for TRISO and metallic fuels.

The Fuel Cycle



DFM Resources

Thorough and timely reviews of advanced reactor package applications is a high priority for the NRC, and our reviews will ensure that new technologies may be used safely

Early and frequent communication is key

Preparation

- Training for NRC staff provides insights on significant safety features of specific designs and technologies
- Technical reports addressing potential challenges assist staff in risk informing their reviews
  - Review of Operating Experience for Transportation of Fresh (Unirradiated) Advanced Reactor Fuel Types (ML20184A151)
  - Potential Challenges With Transportation Of Fresh (Unirradiated) Advanced Reactor Fuel Types (ML20209A541)
- Meetings with advanced reactor vendors provide staff with knowledge on specific designs and technologies

Preparation

- NRC welcomes pre-application engagements to support an efficient review of new applications and amendments (LIC-FM-1, Overview & Expectations of the Certification and Licensing Process)
- Early engagement helps NRC to understand future needs and inform its budget
- NEI Letter dated December 15, 2020
- Preapplication engagement ensures applicants and regulator have shared understanding of
  - the applicable requirements
  - review approach and
  - whether data gaps exist (e.g., testing) that need to be addressed.

Conclusion

- NRC is proactively expanding our knowledge of advanced reactors and their fuels
- Early engagement supports:
  - a common understanding of the regulatory issues associated with advanced reactor fuel designs and technology
  - Timely and efficient reviews
  - NMSS and partners have sufficient resources
- NRC review and oversight ensure safe use of transportation packages in the public domain

Bernard White,

Sr. Project Manager

Bernard.White@nrc.gov

301-415-6577

Yoira Diaz-Sanabria, Chief

Storage and Transportation Licensing Branch

Yoira.Diaz-Sanabria@nrc.gov

301-415-8064

Guidance regarding Non-Power Liquid Fueled Molten Salt Reactor License Applications

Advanced Reactor Stakeholder Meeting
March 2, 2023

William B. Kennedy
Senior Project Manager
Advanced Reactor Policy Branch
Division of Advanced Reactors and Non-Power Production and Utilization Facilities
U.S. Nuclear Regulatory Commission

Contents

- Background
- Overview of the Oak Ridge National Laboratory (ORNL) report
- NRC staff endorsement of Appendix A
- Appendix B of the ORNL report
- Future plans
- Information resources

Background

- Under contract with NRC, ORNL developed a report titled, “Proposed Guidance for Preparing and Reviewing a Molten Salt Non-Power Reactor Application” (ORNL/TM-2020/1478)
- The NRC staff made the report available on the NRC public website in Summer 2020 (https://www.nrc.gov/docs/ML2021/ML20219A771.pdf)

Overview of the ORNL Report

- An information resource for stakeholders interested in licensing of non-power MSRs
- Based on NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors”
- Focuses on the technical information needed to apply NUREG-1537 to the review of a non-power liquid fueled MSR license application

Overview of the ORNL Report

- Main body describes the work to prepare the report
- Appendix A, “Part 1, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power MSRs: Format and Content”
- Appendix B, “Part 2, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power MSRs: Standard Review Plan”

Overview of the ORNL Report

- Covers various topics, including:
  - The facility
  - Site characteristics
  - Design of structures, systems, and components
  - Molten salt reactor description
  - Molten salt reactor cooling systems
  - Engineered safety features
  - Instrumentation and control systems
  - Electrical power systems

Overview of the ORNL Report

- Covers various topics, including:
  - Auxiliary systems
  - Experimental facilities and utilization
  - Radiation protection program and waste management
  - Conduct of operations
  - Accident analyses
  - Technical specifications
  - Other license considerations

Overview of the ORNL Report

- Refers to existing guidance in NUREG-1537 and interim staff guidance augmenting NUREG-1537 for other topics:
  - Financial qualifications
  - Decommissioning
  - Environmental review

NRC Staff Endorsement of Appendix A

- By letter dated November 18, 2020, the NRC staff endorsed Appendix A of the ORNL report as guidance, subject to certain clarifications, for preparing license applications for non-power liquid fueled MSRs under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” Section 50.21(c). (https://www.nrc.gov/docs/ML2025/ML20251A008.pdf)
- Helps applicants provide the information required by 10 CFR 50.34, “Contents of applications; technical information,” and other regulations

Appendix B of the ORNL Report

- Appendix B provides a standard review plan tailored to liquid fueled molten salt reactor technology, including:
  - Areas of review
  - Acceptance criteria
  - Review procedures
  - Evaluation findings
  - Technical rationale

Future Plans

- The NRC staff is considering whether to endorse Appendix B as guidance in the near term
- In the longer term, the NRC staff plans to incorporate the ORNL report, as appropriate, in a new volume of NUREG-1537 covering non-power liquid fueled MSRs

Information Resources

- NRC's public website on advanced reactors https://www.nrc.gov/reactors/new-reactors/advanced.html
- "Endorsement of Appendix A to Oak Ridge National Laboratory Report Titled, "Proposed Guidance for Preparing and Reviewing a Molten Salt Non-Power Reactor Application," as Guidance for Preparing Applications for the Licensing of Non-Power Liquid Fueled Molten Salt Reactors" (ADAMS Accession No. ML20251A008)

Information Resources

- "Proposed Guidance for Preparing and Reviewing a Molten Salt Non-Power Reactor Application" (ORNL/TM-2020/1478) (ADAMS Accession No. ML20219A771)
- NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Contents" (ADAMS Accession No. ML042430055)
- NUREG-1537, Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria" (ADAMS Accession No. ML042430048)

Questions?

Contact me by e-mail at William.Kennedy@nrc.gov

or by telephone at

(301) 415-2313

Pre-Application Engagement on Materials Qualification Issues for Advanced Reactor Licensing

Meg Audrain
Office of Nuclear Reactor Regulation
March 2, 2023
Advanced Reactor Stakeholder Meeting

Agenda

- Why have pre-application engagements?
- Code Requirements
- Environmental Testing
- Design Envelope
- Non-Code Qualified Materials

Why Have Early Engagement?

- Encouraged for all materials used in safety related and risk-significant applications
- Important to ensure NRC staff and applicants have a common understanding on data requirements for these materials
- More efficient for applicants and NRC staff to do this in pre-application space to ensure timely application reviews.
  - Significant lead time for materials testing could lead to delays if not addressed early

ASME Code Requirements

- NRC staff anticipates most applicants will qualify materials and designs to ASME Section III, Division 5, because many of the proposed designs operate at temperatures or in environments where existing Codes endorsed by the NRC or incorporated by reference do not apply.
- Applicants should demonstrate how their design complies with Div 5, as conditioned in RG 1.87
- Applicants should justify deviations from Div 5 and demonstrate why the proposed deviations are acceptable. The use of alternative codes of construction should include a delta analysis.

Environmental Testing

- Div 5 rules do not cover "…deterioration that may occur in service as a result of radiation effects, corrosion, erosion, thermal embrittlement, or instability of the material" but states that these effects shall be taken into account for design or service life
- NRC's forthcoming Materials in Advanced Reactors ISG provides guidance for staff reviews in this area. Applicants should consider this information as they develop qualification, monitoring and surveillance programs
- Environmental testing data is potentially time consuming to gather and results could impact design or component lifetimes

Environmental Testing

- Used to develop corrosion or degradation rates for specific reactor environments
- Needed to understand environmental effects and their impacts on mechanical and thermal behavior
- Needed to set appropriate limits on coolant purity
  - Not explicitly addressed like it is for LWRs
  - No coolant purity standards exist for non-LWR environments
- Needed to determine if transient could potentially be end of life event

Data Supports Design Envelope

- Should show that any data used, historic or planned, is directly applicable to plant design and environment
- Data should support design for operating and accident conditions
- Confirm that any standards referenced in Div 5 were used or provide a delta analysis for standards that were used (e.g., QA programs)

Use of non-Code Qualified Materials

- For use of non-Code Qualified materials, the NRC will review material qualification data
  - ensure material and mechanical properties support intended functions
  - environmental testing still needed
- Applicants should demonstrate that graphite will be qualified as per Div 5. In addition, any deviations from Code should be addressed

Conclusions

- Early engagement is important to support timely application reviews
- NRC wants to ensure a common understanding on data qualification and any potential testing requirements during pre-application
- Beneficial to both NRC staff and applicants

Questions?

Interim Staff Guidance on Materials Compatibility in Advanced Reactor Environments

Meg Audrain
Office of Nuclear Reactor Regulation
March 2, 2023
Advanced Reactor Stakeholder Meeting

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Agenda

- Public Comment Period
- NRC Stakeholders
- Applicability and Purpose of ISG
- Regulatory Framework
- Qualification and Performance Monitoring
- Technical Content
- Conclusions and Questions

Public Comment Period

- Draft ISG, Material Compatibility for Non-Light Water Reactors, DANU-ISG-2023-01 (ML22203A175)
  - FRN will be published in early March 2023
- 60-day public comment period: early March – early May 2023
- Submit comments to be considered by staff. Only written comments will be formally addressed in the final ISG.
  - www.regulations.gov; Docket ID **NRC-2022-0215**

Why Develop the ISG?

- Staff expects that most applicants will demonstrate their materials meet ASME Section III, Division 5 (Div 5), “High Temperature Reactors”
- Div 5 rules do not cover environmental combability; however, it states that these effects shall be taken into account for design or service life of structures, systems and components (SSCs)
- Currently no staff guidance on how to review materials qualification, performance monitoring methods, and surveillance for non-LWRs
- Staff guidance will ensure consistency and clarity for reviewing applications
  - Identify information related to materials qualification that the NRC staff should consider in their reviews
  - Guide the staff in identifying where monitoring and surveillance programs may be appropriate

Applicability

- Applicable to NRC staff reviews of non-LWR designs that propose to use materials allowed under Div 5
  - Power and non-power reactors
  - Part 50 - construction permit and operating license
  - Part 52 - design certification, combined license, standard design approval, or manufacturing license

Non-LWR environment

- Non-LWR environments may have unique material corrosion, degradation mechanisms, and irradiation effects
- Studies have identified the gaps in knowledge that exist for some of these coolant types and the impact on the materials being considered in the construction and operation of these non-LWR nuclear power plants
- Because of the state of knowledge and long test times, there is a strong emphasis on using mitigation strategies, performance monitoring, and surveillance programs to ensure SSCs continue to satisfy the design criteria

Current Regulatory Framework

- Under 10 CFR 50.34(a)(3)(i), 10 CFR 52.47(a)(3)(i), 10 CFR 52.79a(4)(i), applicants must include principal design criteria (PDC) for the facility
- For non-LWRs, Regulatory Guide (RG) 1.232, "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors," issued March 2018, provides proposed guidance for the development of principal design criteria for non-LWR reactors
- Several design criteria relate to materials qualification for structural materials and state the importance of environmental compatibility, inspection, materials surveillance and functional testing

Qualification and Performance Monitoring - Terminology

- Materials qualification
  - Testing conducted in an environment simulating the anticipated operating environment for the reactor, including chemical environment, temperatures, and irradiation
- Performance monitoring
  - Inspections or examinations to confirm adequate performance and to identify unacceptable degradation
  - May also include aging management programs or post-service evaluations
- Surveillance programs
  - Examination of test coupons and components removed from the reactor over the licensed operating period

Qualification and Performance Monitoring

- An SSC's performance will be demonstrated through a combination of materials qualification programs, performance monitoring, and surveillance programs, which collectively provide assurance that a component will meet the design requirements over its intended design life in the applicable environment
- The scope of materials qualification and monitoring programs should include safety-related component materials, safety-significant component material, and as needed, non-safety related component materials whose failure could impact critical design functions
- Testing should be conducted to determine if materials properties and allowable stresses meet applicable codes and standards or other design requirements

Qualification and Performance Monitoring

- Availability of data on performance in a specific operating environment will inform the review to ensure an SSC will maintain its intended function
  - Little data – could require robust performance monitoring and surveillance programs
  - Large amount of data or significant design margin - may require less rigorous performance monitoring and surveillance programs
- Performance monitoring and surveillance programs could be needed for SSCs that are not planned to undergo periodic inspections and/or functional testing

Technical Content of ISG

- The ISG separates degradation issues into generically applicable issues and technology specific issues
- Three technology specific appendices
  - Molten salt reactors, liquid metal reactors, and HTGRs
- Represents current state of knowledge - as additional operating experience and laboratory testing become available, treatment of issues may change, and new issues may be identified.

General Degradation Mechanisms

- Corrosion
- Creep and creep Fatigue
- Environmentally assisted cracking
- Flow induced degradation (abrasion, erosion, cavitation)
- Flow induced vibration
- Gaskets and Seal chemical compatibility
- Irradiation effects
- Stress relaxation cracking
- Thermal emissivity, thermal aging, thermal fatigue and transients
- Wear/fretting

General Materials Issues

- Advanced manufacturing technologies
- Lubricants
- Ceramic insulation
- Weld design and fabrication
- SiC/SiC composites
- SA-508/533 Bainitic Steel for RPVs

Molten Salt Reactor Appendix

- Graphite compatibility
- Materials considerations (degradation, cracking, corrosion)
- Salt composition
- Tritium production

Liquid Metal Reactor Appendix

Sodium-cooled fast reactors

- Caustic stress-corrosion cracking
- Exothermic reactivity with water
- Sodium purity effects on corrosion
- Combining ferritic steels and austenitic steels (galvanic corrosion)
- Liquid metal embrittlement

Lead-cooled fast reactors

- High temperature corrosion
- Effect of flow velocity
- Liquid metal embrittlement
- Nonmetallic materials
- Oxygen control

High Temperature Gas Cooled Reactor Appendix

- Creep-rupture strength
- Emissivity
- Graphite
- Graphite dust
- Helium impurities
- Metallic materials qualification considerations

Conclusions

- NRC staff developed an ISG to guide staff on reviewing applications using materials allowed under Div 5
- ISG has been issued for public comment
  - Comment period – March to May 2023
- NRC staff encourages stakeholders to provide feedback on contents of ISG through this process

Questions?

Break

Meeting will resume at 2:45 pm EST

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 417 405 578#

Periodic Advanced Reactor Stakeholder Meeting: Status of Draft Regulatory Guide 1410 and 1307, Including Responses to NEI Comments

Dr. John Stamatakos
Institute Scientist at Southwest Research Institute
March 2, 2023

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Overview

- Changes since publication of the Pre-decisional guides
  - Current versions address both Framework A and Framework B, consistent with the most recent version of 10 CFR Part 53
  - Three options apply to both frameworks
- Discuss NEI comments and responses (four main groups)
  - Comments that were addressed/incorporated in the current drafts
  - Comments used for planning Appendix B and to revised RIL 2102-04/NUREG for Option 3.
  - ASCE 7 related
  - Part 50/52 related
- Future plans and summary

Changes Since Publication of Both The Pre-decisional Guides

- Added discussions in sections A and B for both Framework A and Framework B, consistent with the most recent version of 10 CFR Part 53 (prior draft was only for Framework A)

- Modified all three options to address both frameworks

- Incorporated many review comments and suggestions, including those from NEI.

Table 1 Relationships among Regulations, Guidance, and Seismic Design Options for Framework A

Use of Risk Analysis/Insights	Regulations	Basis and Key Guidance	Applicable Proposed Seismic Option for Design
SPRA or PRA-based SMA to establish safety margin for the design	§ 53.500 § 53.510 § 53.520 § 53.415 § 53.480 § 53.210 § 53.220 § 53.230 § 53.450	• RG 1.233 option to use current regulations and design guidance. • DC/COL-ISG-020	**Option 1** SDC-5 and LS-D with a minimum 0.1g foundation level DBGM consistent with current LWR practice. SPRAs/SMAs are used to demonstrate adequate margin beyond the design.
Integrated SPRA (e.g., for LMP framework endorsed in RG 1.233)	§ 53.500 § 53.510 § 53.520 § 53.415 § 53.450 § 53.480 § 53.210 § 53.220 § 53.230 § 53.470	• RG 1.233 with a voluntary option to use LMP framework. • DG-1413	**Option 2** Follow LMP framework using PRAs as integrated tools in the design. Select SDC and LS for SSCs classified as SR or NSRSS, and for others as needed to meet the safety criteria.
Risk analysis required	§ 53.500 § 53.510 § 53.520 § 53.415 § 53.450 § 53.480 § 53.210 § 53.220 § 53.230 § 53.470	• DG-1413	**Option 3** Use PRA either in a fully integrated manner to select SDCs and LSs or to demonstrate adequacy of the selection of SDCs, LSs, and resulting design. Select SDCs and LSs for SSCs classified as SR or NSRSS, and for others as needed to meet safety criteria.

Table 1 and Table 2 (next slide) are from Draft RG 1410 but are applicable to both RGs.

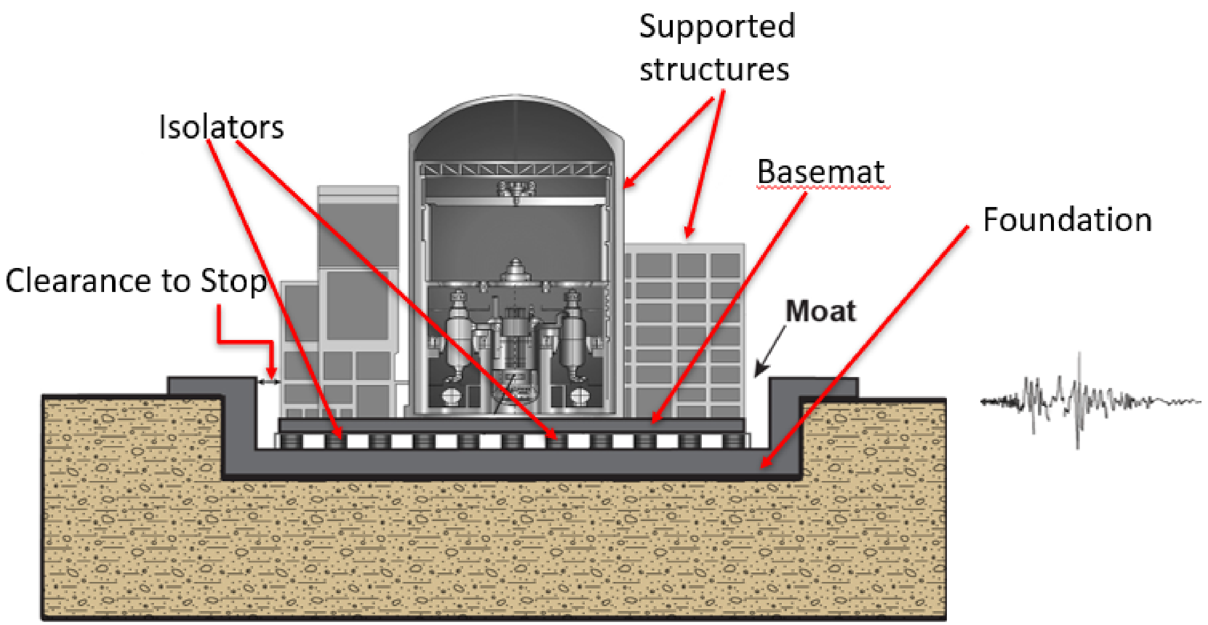
Table 2 Relationships among Regulations, Guidance, and Seismic Design Options for Framework B

Use of Risk Analysis/Insights	Regulations	Basis and Key Guidance	Applicable Proposed Seismic Option for Design
SPRA or PRA-based SMA to establish safety margin for the design	§ 53.3525 § 53.4730(a)(4) § 53.4730(a)(14)	• DC/COL-ISG-020	**Option 1** SDC-5 and LS-D with a minimum 0.1g foundation level DBGM consistent with current LWR practice. SPRAs/SMAs are used to demonstrate adequate margin beyond the design.
Integrated SPRA (e.g., for LMP framework endorsed in RG 1.233)	§ 53.3525 § 53.4733 § 53.4730(a)(5)(ii), (iii), and (iv) § 53.4730(a)(34)(i) § 53.4730(a)(1)(vi)(A)	• RG 1.233 with a voluntary option to use LMP framework. • DG-1413	**Option 2** Follow LMP framework using PRAs as an integrated part of the design process. Select SDCs and LSs for SR SSCs and other safety-/risk-significant SSCs as needed to meet the safety criteria.
Risk analysis required	§ 53.3525 § 53.4733 § 53.4730(a)(5)(ii), (iii), and (iv) §53.4730(a)(34)(i) § 53.4730(a)(1)(vi)(A)	• DG-1413	**Option 3** Use PRA either in a fully integrated manner to select SDCs and LSs or to demonstrate adequacy of the selection of SDCs, LSs, and resulting design. Select SDCs and LSs for SR SSCs and other safety-/risk-significant SSCs as needed to meet the safety criteria.

Table 1 and Table 2 (next slide) are from Draft RG 1410 but are applicable to both RGs.

Pre-decisional Draft RG 1307

– Technical considerations:
- Use the same technical approach as described in Pre-decisional Draft RG 1407 (3 options)
- Focus on addressing SI specific criteria for each of the 3 options
- Guidance relies on ASCE 43-19 and ASCE 4-16 as well as available literature



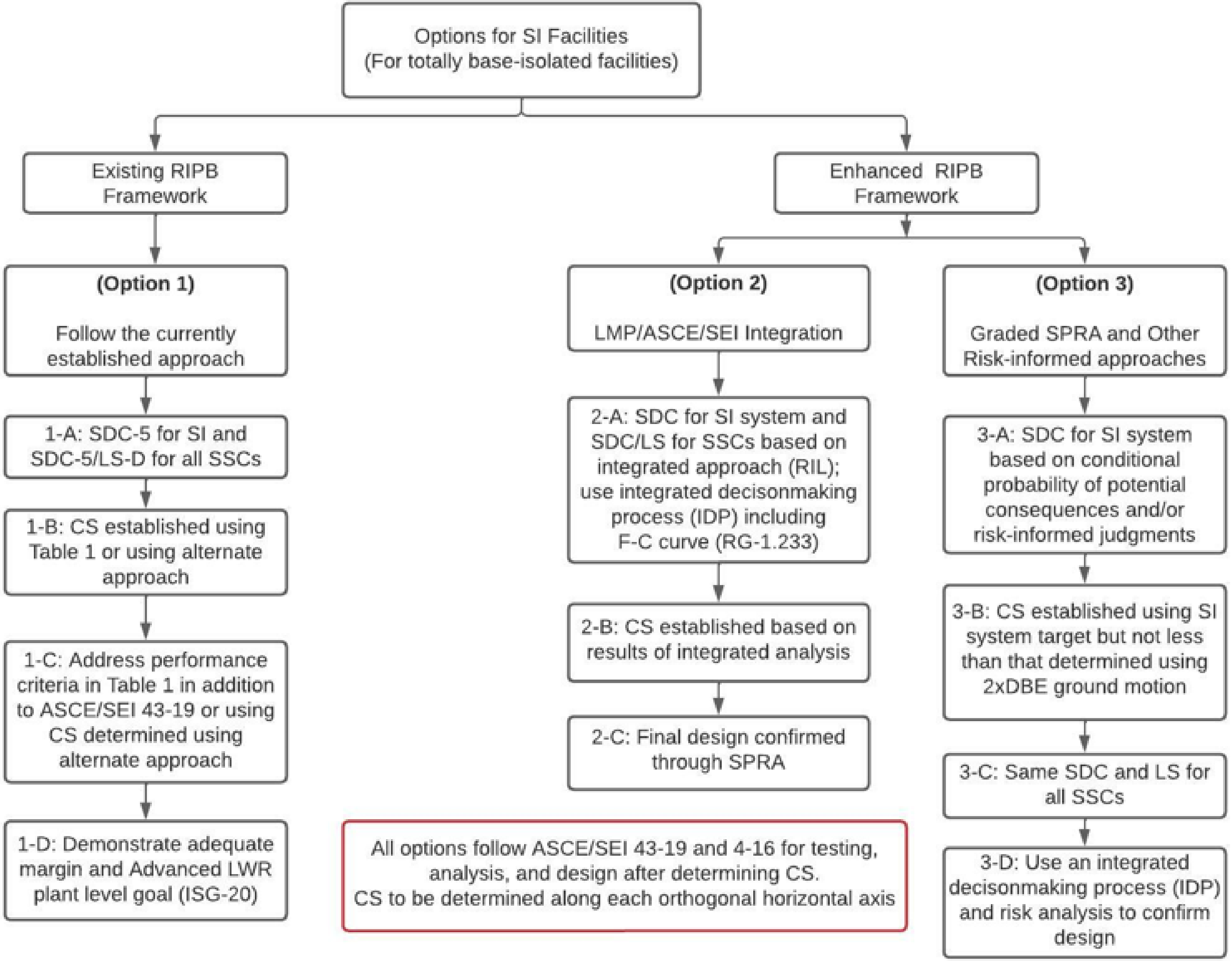


Figure 2 Three options for seismically isolated facilities

Revised flowchart from Draft RG 1307

Color Coding to Categorize Responses in NEI Table

We organized the NEI comments in a table and then categorized them as follows:

Description	Comment #s (Part 1)	Comment #s (Part 2)
To be incorporated in the next revision of the RGs	1, 3, 5, 7, 9, 12, 13	1, 2, 3, 5, 7, 12, 14
To be addressed in next revision of RIL/ NUREG	4, 6, 14, 18, 19	9, 10, 11, 13
We feel outside the scope of the RGs	2, 8, 10, 11, 15, 16, 17	4, 6, 8

Example: Comments that Were Incorporated in the Revised RGs

3	B. Proposed Options	The DG provides an example for Option 2, but not for Option 3. Option 3 may be the most desirable, but it is also the one that will be hardest to reach consensus on without further guidance.	Consider providing a similar example for Option 3.	Develop one or more example applications of Option 3 (with specifications for Framework A and Framework B) as Appendix B to Draft RG 1410.
7	Introduction B. Proposed Options	Based on the advanced reactor public meeting that took place on 10-12-2022, NRC also discussed the possibility to use the options under Framework B as long as one adopts safety criteria from Framework A instead of Framework B's principal design criteria. These clarifications allow more flexibility to the industry and should be covered in the DGs otherwise the industry may assume that if they use Framework B or Part 50, they are not allowed to use the methodology from the DG.	Consider addressing how to apply Option 2 and 3 to Framework B.	The revised draft RGs now address both Framework A and Framework B. However, the draft RGs do not discuss AERI.

Appendix B

- Working with NRC Staff, the SwRI team will develop examples of how to implement Option 3 (with specific ties to Framework A and Framework B as necessary).

- Option 3 provides flexibility to an applicant for seismic design considering unique aspects of its plant design, site, and other considerations. We will focus on design and analysis strategies that an applicant can follow to demonstrate compliance with the safety and risk requirements in 10 CFR Part 53 using Option 3.

- As necessary, we will demonstrate key steps in our example strategies with performance and risk analyses similar to the ones already provided in Appendix A.

- More details will be developed in a revision to RIL 2021-04 that is expected to be in the form of a NUREG/CR to support the two RGs.

Example: Comments To be Addressed in the Next Revision of RIL 2021-04/ NUREG

6	B. Proposed Options	Both Option 2 and Option 3 require that the design decisions be "confirmed". For option 2, this is done with SPRA. For Option 3, either by SPRA or other risk-informed approaches. The DG does not actually discuss what this means, but the implication is that this is the site-specific analysis associated with a COL or construction permit application. The RIL doesn't mention this at all. Again, the lack of any guidance brings up concerns because once the design is done you would not want to have to change an SSC from, say SDC-4, to SDC-5. There needs to be some finality to the design requirement that would	See Comment Basis	Discussions in the RIL/NUREG will be expanded to clarify that the use of a F-C curve is not mandatory in Option 3. However, both Framework A and Framework B ultimately require
		not result in a categorization change based on the specific site where a plant would be built. a. Would it be adequate to show that the overall seismic risk is low, or would you also have to show that the F-C for each seismic scenario that includes an SDC-4 SSC is under the curve? b. Would you have to demonstrate that there are no SDC-4 SSCs that are "risk-significant" regardless of the overall or individual scenario risk?		a SPRA type analysis.

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Example: Comments We feel Are Outside the Scope of the RGs

2	Introduction B. Proposed Options D. Implementation	In the advanced reactor public meeting that took place on 10-12-2022, NRC explained that the guide's Options 2 and 3 can also be used under part 50 as long as: • A singular SSE applies and its value is not below per Part 50 Appendix S • Safety criteria similar to those of Part 53 Framework A are clearly defined if the SDC is below a Category 5.	Consider addressing how to apply Options 2 and 3 to Part 50.	This is now outside the scope of the two Draft RGs. NRC staff to decide whether a separate RG could be developed to realize these options.

10 CFR Part 50 and 52

- There are no longer any references to these regulations in the two Draft RGs.

- NRC staff will evaluate the potential to develop additional guidance on how the RIPB approaches using ASCE 43-19 and ASCE 4-16 can be adopted under these regulations

- NRC staff are also planning for an update to RG 1.208 to be consistent with ASCE 43-19 and 10 CFR Part 53.

Example: Comments We feel Are Outside the Scope of the RGs

11	B. Proposed Options	It appears that there should be an option 4 that follows NRC's process for review and approval of facilities such as medical isotope facilities but applied to advanced reactors that have a similar risk level due to their low inventory and dose consequence. Shine medical and Northwest medical clearly detail in their PSAR in Chapter 2 that they rely solely on the seismic methodology of IBC/ASCE 7. Despite these facilities processing radioisotopes under Part 50 licensing, their low-risk thresholds allow the NRC to approve their application despite it utilizing industrial non-nuclear codes. IBC/ASCE 7 are also the codes used for DOE for its nuclear facilities. It is unclear why the same cannot be done for micro-reactors that present a very low risk and can prove it through a source term analysis. It is understood that IBC/ASCE 7 could be used as part of Option 3, however it is not clear if sufficiently low source term results based on an extreme accident would be sufficient.	Consider an option for the use of IBC/ASCE 7 based on medical isotope and DOE nuclear facility precedents.	The scope of the two draft RGs is based on ASCE 43-19. The IBC/ASCE 7 approach to characterizing the hazard and evaluating SSCs differs from that in ASCE 43-19 and current NRC guidance. An evaluation of ASCE 7 for use with commercial nuclear power plants has not been performed by the NRC staff or industry for its applicability to lower-risk facilities. We propose a technical meeting on this topic to address this comment.

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ASCE 7

- ASCE 43, ASCE 4, and associated design codes reflect current Nuclear Industry design and construction practices and produce acceptable design with sufficient margin (actual performance is a function of both design and construction) as demonstrated by recent SPRAs.

- NRC has evaluated ASCE 43 and ASCE 4 in detail and has developed regulatory positions with exceptions, additions, and clarifications.

- ASCE 7 takes a different approach to safety and performance, and NRC staff (and industry to our knowledge) have not yet evaluated how to align this approach with the current NRC regulatory approach for power production commercial nuclear plants.

- The Draft RGs provides an acceptable way to meet the regulations. Therefore, the following statement in RG: "any code other than ASCE 43-19 and ASCE 4-16 for seismic design of SSCs with appropriate justification."

- We propose a technical meeting to discuss what information is needed to evaluate ASCE 7.

Summary

- Revised Draft RG 1410 and Draft RG 1307 have been updated to include both Framework A and Framework B, and all three options are now available for both frameworks.

- The Draft RGs will be revised to address comments received for the Trial DG including adding an Appendix B to illustrate Option 3.

- We have addressed NEI comments and plan to address and incorporate all public comments that fall within the scope of the two Draft RGs.

- We anticipate Appendix B draft in 3 months (end of May 2023).

Future Meeting Planning

- The next periodic stakeholder meeting will be scheduled for April or May 2023.

- If you have suggested topics, please reach out to Steve Lynch at Steven.Lynch@nrc.gov

How Did We Do?

- Click link to NRC public meeting information:

  https://www.nrc.gov/pmns/mtg?do=details&Code=20230075

- Then, click link to NRC public feedback form:

Meeting Feedback

Meeting Feedback Form EXIT

Meeting Dates and Times

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