Advanced Reactor Stakeholder Public Meeting

September 29, 2021

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 970 740 540#

Time	Agenda	Speaker
10:00 – 10:10 am	Opening Remarks	NRC
10:10 – 10:40 am	Draft White Paper on Licensing Strategies for Micro-Reactors	NRC
10:40 – 11:10 am	Overview of Idaho National Laboratory (INL) Report, INL-EXT-21-61847, “Regulatory Research Planning for Micro-Reactor Development”	INL
11:10 – 11:40 am	Summary of Technical Reports on Transportation and Storage of Advanced Reactor Fuel	NRC
11:40 am – 12:10 pm	Draft White Paper, ARCAP Guidance for Fire Protection (Operations)	NRC
12:10 – 1:00 pm	**Lunch Break**	All
1:00 – 1:45 pm	NRC and Canadian Nuclear Safety Commission (CNSC) Joint Report: Technology Inclusive and Risk-Informed Reviews for Advanced Reactors	NRC and CNSC
1:45 – 3:15 pm	Update on Draft NUREG-2246, “Fuel Qualification for Advanced Reactors"	NRC
3:15 – 3:25 pm	**Break**	All
3:25 – 4:10 pm	Update on White Paper for Scalable Human Factors Engineering and Flexible Staffing for Advanced Reactors	NRC
4:10 – 4:55 pm	Draft Regulatory Guide to Endorse ASME Section XI, Division 2, Reliability and Integrity Management (RIM) Programs for Non-Light Water Reactors	NRC
4:55 – 5:00 pm	Future Meeting Planning and Concluding Remarks	NRC

Advanced Reactor Integrated Schedule of Activities

Advanced Reactor - Summary of Integrated Schedule and Regulatory Activities

Summary of Integrated Schedule and Regulatory Activities (updated 11/02/2020)

Advanced Reactor Program - Summary of Integrated Schedule and Regulatory Activities*

		Legend	
Strategy 1	Knowledge, Skills, and Capability	Concurrence (Division/Interoffice)	• EDO Concurrence Period
Strategy 2	Computer Codes and Review Tools	Federal Register Publication	Commission Review Period**
Strategy 3	Flexible Review Processes	Public Comment Period	▼ ACRS SC/FC (Scheduled or Planned)
Strategy 4	Consensus Codes and Standards	Draft Issuance of Deliverable	External Stakeholder Interactions
Strategy 5	Policy and Key Technical Issues	Final Issuance of Deliverable	↓ Public Meeting (Scheduled or Planned)

Version 11/2/20

Present Day

Strategy	Regulatory Activity	Commission Papers	Guidance	Rulemaking	NEIMA	Complete	2020 Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	2021 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))																													
	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																													
	Code Assessment Reports Volumes 1 (Systems Analysis)					x																								
	Reference plant model for Heat Pipe-Cooled Micro Reactor					x																								
	Reference plant model for Sodium-Cooled Fast Reactor					x																								
	Reference plant model for Fluoride-Salt-Cooled High-Temperature Reactor																													
	Reference plant model for Gas-Cooled Pebble Bed Reactor																													
	Reference plant model for Molten Salt Fueled Reactor																													
	Code Assessment Reports Volumes 2 (Fuel Perf. Anaylsis)					x																								
	FAST code assessment for metallic fuel																													
	FAST code assessment for TRISO fuel																													
	Code Assessment Reports Volumes 3 (Source Term Analysis)					x																								

https://www.nrc.gov/reactors/new-reactors/advanced/details#advSumISRA

Advanced Reactor Program

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Advanced Reactor Stakeholder Meeting

Draft White Paper on Micro-Reactor Licensing Strategies

ADAMS Accession No. ML21235A418

September 29, 2021

Overview

- Purpose and Scope
- Discussion
  - Licensing Strategies
  - Operational Programs
  - External Hazards and Siting
  - Possession and Transportation of Special Nuclear Material
  - Environmental Review
- Next Steps and Questions

Purpose

- Develop optional strategies to streamline licensing of micro-reactors to support growing interest in micro-reactors

Scope

- Licensing strategies under existing regulations
- Potential efficiencies gained at Combined License (COL) stage
- Strategies for streamlining environmental reviews
- Providing recommendations to advanced reactors rulemaking working group on potential rulemaking changes

Licensing Strategies

- Standardization through design certification:
  - No site-specific design features relied on for safety
  - COL Items not used
  - Bounding site parameters at design
  - Operational programs reviewed at design (e.g., topical reports)

Licensing Strategies

- Standardization through COL:
  - Applications are fully standardized
  - Advanced Nuclear Reactor Generic Environmental Impact Statement (ANR GEIS) fully leveraged
  - Design center review approach utilized

Licensing Strategies

- Standardization through manufacturing license:
  - Could be utilized to build micro-reactors in a factory
  - Could reduce need for inspections and verifications at deployment site

Operational Programs

- Use of standardized operational programs to streamline COL review through:
  - NRC approved topical reports and templates
  - NRC review of operational programs within design certification being considered
  - Two Groups of Op. Programs –
    - Group 1 – Material to the findings on the design
      - Part 53 rulemaking working group exploring options to provide finality to technical specifications at design certification stage
    - Group 2 – Not material to adequacy of the design

Operational Programs

- Providing finality in design certification for Groups 1 and 2 programs would constitute policy change and require Commission approval
- May address Group 2 programs in topical reports proposing standardized operational programs, which a COL application could reference

External Hazards and Siting

- Under Part 52 design certification, site parameters compared with actual site characteristics
- Actual characteristics of the site should fall within site parameters in the design certification
- If design certification identifies parameters for which design is bounding for several sites, then COL review can be reduced in this area
- Simplified flow chart developed to facilitate external hazards screening

Flowchart for an acceptable site hazard parameters assessment in the micro-reactor design process

Identify external hazards (EXH) in region of Micro-Rx

Site Investigation to characterize EXH site (EXH-C)

Identify Micro-Rx vendor’s EXH postulated site parameters (EXH-P)

Compare EXH-C with EXH-P

EXH-C within EXH-P?

Y

End assessment; document results

N

Design reanalyzed and EXH-C within modified EXH-P?

Y

End assessment; document results

N

Evaluate mitigating actions

A

End assessment; document results

U

Site not suitable for deployment of this Micro-Rx

Key:
Y = yes
N = no
U = unacceptable performance
A = acceptable performance

External Hazards and Siting

- 10 CFR Part 100.21 & 100.23 requires characterization of several potential hazards
  - Geology, seismology, meteorology, hydrology, etc.
- Recent advances in seismic hazards include:
  - Graded approach depending on seismic design category
  - Development of probabilistic seismic source and ground motion models for Central and Eastern United States (CEUS)
  - Expanded Senior Seismic Hazards Analysis Committee (SSHAC) guidance for Study Levels 1 and 2
  - Updated American Nuclear Society (ANS) Standards 2.27, 2.29 and American Society of Civil Engineers (ASCE) 43-19

Possession and Transportation of Special Nuclear Material

- Fuel reactor in factory and then transport to site of usage
  - Domestic licensing of byproduct material (10 CFR Part 30)
  - Domestic licensing of source material (10 CFR Part 40)
  - Domestic licensing of special nuclear material (10 CFR Part 70)
  - Transportation package certification (10 CFR Part 71) (see slides from August 26, 2021, meeting - ML21237A463)
  - Physical protection of plants and materials (10 CFR Part 73)
  - Material control and accounting of special nuclear material (10 CFR Part 74)

Environmental Review

- Statutory Requirements
  - National Environmental Policy Act
  - Additional Laws - "The NEPA Umbrella"
    - Consultation
    - Compliance
    - Administrative
- Implementing Regulatory Framework
  - 10 CFR Part 51

Environmental Review

- Actions Under Current Framework
  - Guidance
    - Regulatory Guide 4.2 Rev. 3 (ML18071A400)
    - Interim Staff Guidance ISG-029 (ML20252A076)
  - Generic Environmental Impact Statement
- Potential Change to Framework
  - Enhancing Review Flexibility
    - Proposed Update to Part 51 (SECY-21-0001)

Next Steps

- We welcome your questions and feedback on the draft micro-reactor white paper.

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Regulatory & Licensing Challenges in Microreactors

NRC Advanced Reactor Stakeholder Meeting

September 29, 2021

Jason Christensen| Regulatory and Licensing Engineer, Idaho National Lab

Regulatory Research Plan

- In collaboration, INL and ORNL have developed a Regulatory Research Plan (RRP) for the Microreactor Program
- This work links major research activities in microreactor technologies, as sponsored by DOE Office of Nuclear Energy's (DOE-NE), to key regulatory requirements and licensing challenges likely to affect deployments in the domestic commercial energy market
- The RRP examines and prioritizes key research and development opportunities and recommends activities to DOE-NE R&D planners and investigators concerning the expected impact of such work to the microreactor licensing critical path
- Initial report was completed March 5, 2021

MRP Microreactor Program

Industry Survey

- Industry Survey was sent to members of the Advanced Reactor and Microreactor NEI Working Groups
- Stakeholders were asked to rank the following items by criticality:
  - Autonomous and Remote Control/Monitoring
  - Grid Interaction
  - Factory Assembly
  - Transportation
  - Staffing
  - Digital Controls
  - Instrumentation
  - Modeling and Simulation
  - Siting and Environmental Impact
  - Security and Safeguards

MRP Microreactor Program

Industry Survey (Cont’d)

- In addition to the rankings, stakeholders were asked:
  - Is this list complete? If not, what subject areas should be added?
  - Please provide any specific R&D items that DOE can support through funding and laboratory expertise, technology, and assistance, and that would benefit the licensing of microreactor designs.

MRP Microreactor Program

Initial Survey Results

- The initial survey results were grouped into bands based on importance and time criticality
- The areas of highest priority are band 1 and are needed before subsequent bands

Band	Topic Area
1	Autonomous and Remote Control/Monitoring
	Modeling and Simulation
2	Transportation
	Siting and Environmental Impact
	Security and Safeguards
	Factory Assembly
3	Operations, Maintenance, and Security Staffing
	Grid Interaction
	Digital Controls
	Instrumentation

MRP Microreactor Program

Items with Multiple Areas of Regulatory Need

- Transportation:
  - There are multiple stages of transportation throughout the life of a microreactor
    - From factory to use site (fueled but not yet operational)
    - Between use sites (post-operation)
    - From use site to disposition process facility (spent fuel for disposal)
  - Each of these stages of transport are unique and will require the applicant to meet different regulations, such as:
    - Transport Container Design
    - Shielding for each operational stage
    - Shipping Type (air, train, ship, truck)
    - Emergency response

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Items with Multiple Areas of Regulatory Need (cont'd)

- Autonomous and Remote Control/Monitoring
  - Many consider this to be one area
  - This encompasses many different licensing areas
    - Autonomous reactor operation
    - Remote Control/Monitoring
    - Cyber Security
    - Digital Controls
    - Operator Licensing
    - Number of Operators per Reactor
  - Most of these areas are new or would require new licensing approaches

Additional Survey Results

- Other areas of regulatory framework development need identified by stakeholders include:
  - Aircraft Impact Assessment
  - Emergency Planning (currently an NRC rulemaking in progress-see 10CFR 50.160)
  - Control Room Design
  - Siting and Environmental concerns
  - Radiation Protection
  - Waste Management
  - Regulatory Oversight (including oversight of remote and autonomous reactors)
  - Manufacturing Licenses (including fueling and defueling in a factory setting)

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Path Forward

- INL staff will utilize this report to continue expanding on the survey results to better identify specific R&D projects to recommend
- INL and ORNL staff have completed updating the RRP draft with the survey information and have proposed R&D activities that will directly support the Microreactor Program
- Some identified areas do not provide detailed R&D activities because of proprietary information withheld by the developers
- Final Regulatory Research Plan draft is being used to support FY22 DOE Microreactor Regulatory Development Planning

Recommendation for Future Work: Transition from Shipping to Operational Status

- Microreactors are slated to have modular construction techniques that will ship major components or possibly near complete units from the factory to approved siting locations
  - Reduces complexity of on-site assembly and construction
- The transition from shipment regulations (10CFR Part 71) to an operating license under 10CFR Parts 50/52 is not currently addressed by regulations
- 10CFR Part 53 will need to address this missing piece (likely under Subpart E, which contains construction aspects and inspections, tests, analyses, and acceptance criteria)

- **Key Question:** What regulatory process will be used during the transition of a mobile reactor from 10 CFR Part 71 to 10CFR Part 50/52 or Part 53?
- **Deliverable:** Regulatory Strategy Report that outlines the transition that could be submitted to NEI/NRC for comment or endorsement

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FY22 Planned Work: Guidance for Manufacturing Licenses

- Some microreactor developers are planning to fabricate and fuel their reactors at a manufacturing facility
- NRC Manufacturing Licenses are covered under 10CFR Part 52 Subpart F
  - Factory fueling is not covered in Subpart F
- 10CFR Part 53 will contain a subpart covering manufacturing licenses (Subpart E)
  - At this time, no factory fueling regulations are being included

- **Key Question:** What type of regulatory requirements for the factory construction and fueling of a microreactor are required?
- **Deliverable:** A report that provides recommendations for the development of regulations to allow factory fueling and guidance for developers for those recommendations

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Guidance for Manufacturing Licenses

- INL Staff will review current manufacturing license regulations to determine what areas would need to be modified to allow for factory fueling
- Propose changes to regulations to allow for factory fueling, refueling, and defueling a microreactor
- Develop guidance for obtaining a manufacturing license to perform these tasks
- The report(s) generated by this work will be submitted for review and potential endorsement by NEI and/or NRC

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Conclusions

- INL and ORNL staff continue to seek new and updated needs to support the development and deployment of microreactors
- For more information, request a survey, or to provide specific research input, please contact:

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MRP Microreactor Program

MRP Microreactor Program

Approach for Assessing Storage and Transportation of Advanced Reactor Fuels

September 29, 2021

Nick Hansing
Materials Engineer
Division of Fuel Management
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Assessment Process for Advanced Reactor Fuels

- Survey state of knowledge on candidate technology
- Identify information needs
- Determine approach to address needs
- Take action as appropriate on desired approach

Technical Reports

- NRC used the Center for Nuclear Waste Regulatory Analyses (CNWRA) for support in assessment process

- CNWRA provided a series of reports that surveyed the state of knowledge and identified potential challenges/information needs to enhance the efficiency and effectiveness of NRC regulatory reviews

- Focus narrowed to potential information needs of storage and transportation applications for the following advanced reactor fuels (ARF):
  - tristructural isotropic (TRISO) fuel
  - nuclear metal fuel

Subject Reports

- Transportation of Fresh (Unirradiated) ARF types
  - Literature Review: ML20184A151
  - Identification of Potential Challenges: ML20209A541
  - Identification of Potential Information Needs: ML21021A326
- Transportation of Spent (Irradiated) ARF types
  - Literature Review and Identification of Potential Challenges: ML20237F393
- Storage of Spent (Irradiated) ARF types
  - Literature Review: ML20211L885
  - Identification of Potential Challenges: ML20022A217
- Final Summary Report (not yet publicly available)

Key Messages

- Wide variability in potential designs
- Regulatory framework currently provides sufficient requirements and guidance to allow the NRC staff to make a finding of reasonable assurance of adequate protection to public and workers for applications for advanced reactor fuels
- Some topics could use enhancement
- Pre-application meetings are crucial

TRISO Results: Fresh Fuel Transportation

- Very few documents on the topic, but no record of observed or postulated degradation mechanisms
- Versa-Pac VP-55 and VP-110 (CoC 9342)
- Already certified for fresh TRISO fuel transport

- Challenges/information needs? Criticality safety

TRISO Results: Spent Fuel Transportation

- Very limited information - no records of observed degradation or damage of the fuel or any safety issues during transportation
- TN-FSV cask (CoC 9253) could possibly be used to transport spent TRISO fuel
- Challenges/information needs? Criticality safety

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TRISO Results: Storage

- Fort St. Vrain Experience – no observed aging effects or degradation of fuel
- Arbeitsgemeinschaft Versuchsreaktor (AVR) Experience – steel cans leaked in wet storage; casks retained potential releases

- Challenges/information needs? Criticality safety, Corrosion of non-fuel hardware, Mechanical properties of materials

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Metal Fuel Results: Fresh Fuel Transportation

- Very few documents on the topic, but no record of observed or postulated degradation mechanisms
  - UNC-2600 (CoC 5086)
  - ES-3100 (CoC 9315)
  - Versa-Pac Models VP-55 and VP-110 (CoC 9342)

- Challenges/information needs? Criticality safety, Sodium behavior, Mechanical properties of materials

Metal Fuel Results: Spent Fuel Transportation

- No records of fuel degradation/damage or cask safety issues during:
  - FFTF moves using T-3 cask
  - Fermi 1 using stainless steel canisters
  - EBR-II moves using TN-FSV, NAC-LWT and HFEF-5 and HFEF-14

- Challenges/information needs? Criticality safety, Sodium behavior, Mechanical properties of materials, and See next slide

Metal Fuel Results: Spent Transportation Options

- NLI-1/2 cask
  - (CoC No. 9010, spent metal fuel – expired CoC)
- T-3 cask
  - (CoC No. 9132, spent metal fuel – expired CoC)
- NAC-LWT cask
  - (CoC No. 9225, spent metal fuel)
- RH-TRU 72-B cask
  - (CoC No. 9212, remote-handled transuranic wastes – need amendment)
- TRUPACT-II cask
  - (CoC No. 9218, contact-handled transuranic wastes – shielding needed, possibly requiring drop tests, amendment likely needed)

Metal Fuel Results: Storage

- ~2 metric tons of heavy metal EBR-II fuel was in wet storage
- Some water in-leakage resulted from improper sealing
- Department of Energy is moving fuel to dry storage

- Challenges/information needs? Criticality safety, Corrosion of non-fuel hardware, Sodium behavior, Mechanical properties of materials

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Key Messages

- Wide variability in potential designs
- Regulatory framework currently provides sufficient requirements and guidance to allow the NRC staff to make a finding of reasonable assurance of adequate protection to public and workers for applications for advanced reactor fuels
- Some topics could use enhancement
- Pre-application meetings are crucial

Corrosion of non-fuel hardware

- New fuel designs may have new materials and new container environments (TRISO use of graphite, for instance)
- Analysis for chemical, galvanic, or other reactions needed
- One such aspect could be the presence of solid fluoride salt residue from fluoride salt cooled reactors, which generates fluorine under radiolysis

Mechanical properties of materials

- New materials are envisioned for both TRISO and metal fuels
- Legacy data for metal fuel
- New information for new materials and designs – modeling, testing, analyses

Criticality Safety

- Criticality performance for these fuels can be complicated by variations in enrichment and burnup
- Because of lack of critical experiments, applicants should consider either:
  1) Conduct new criticality experiments to expand number available to validate HALEU systems
  2) Use sensitivity/uncertainty analysis techniques (e.g., SCALE/TSUNAMI) to demonstrate that sufficient existing experiments are applicable to HALEU systems
  3) Include enough margin in the criticality analysis for HALEU systems to account for validation uncertainties due to insufficient critical experiments.
- If burnup credit is sought, methods would need to be developed
- Isotopic depletion & criticality (k_{eff}) computer codes need validation

Sodium Behavior

- Limited testing or analysis for sodium-containing metal fuel
- Sodium creep and location shift due to vibration and/or drop
- Reactions with water
- Thermal performance of bonding

Path Forward

- Obtain input from stakeholders on reports, particularly challenges and information needs
  - Letters sent to various stakeholders encouraging pre-application interactions
  - Feedback sought in this meeting as to the desirability of a public meeting/workshop to discuss what would be expected in fuel facility and storage and transportation applications?
- Develop review guidance enhancements (TBD)
- Continue assessment process for additional technologies

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- NRC welcomes pre-application engagement on potential storage and transportation applications.

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Acronyms

- ARF: Advanced Reactor Fuel
- AVR: Arbeitsgemeinschaft Versuchsreaktor
- CNWRA: Center for Nuclear Waste Regulatory Analyses
- EBR-II: Experimental Breeder Reactor-II
- FFTF: Fast Flux Test Facility
- HALEU: high-assay low-enriched uranium
- TRISO: Tristructural isotropic
- LWR: Light water reactor

Stakeholder's Meeting

Advanced Reactor Content of Application Project (ARCAP)

Risk-informed, Performance-Based Fire Protection Program (for Operations)

Interim Staff Guidance (Draft)

Background

- The Advanced Reactor Content of Application Project (ARCAP) has been developing guidance to support the review of non-light water reactors (non-LWRs), modular LWRs and stationary micro-reactors.
- The guidance has been developed as draft Interim Staff Guidance (ISG) documents in the form of draft white papers.
- One of those draft documents is the ISG on “Risk-informed, Performance-Based Fire Protection Program (for Operations)” (ML21253A134).
- The purpose of this ISG is to facilitate the review of advanced reactor applications that use a risk-informed, performance-based approach to develop their fire protection programs.

ARCAP and Technology Inclusive Content of Application Project (TICAP) - Nexus

Outline Safety Analysis Report (SAR) – Based on TICAP Guidance

1. General Plant Information, Site Description, and Overview of the Safety Case
2. Methodologies and Analyses
3. Licensing Basis Event (LBE) Analysis
4. Integrated Evaluations
5. Safety Functions, Design Criteria, and SSC Safety Clasification
6. Safety Related SSC Criteria and Capabilities
7. Non-safety related with special treatment SSC Criteria and Capabilities
8. Plant Programs

Additional SAR Content –Outside the Scope of TICAP

9. Control of Routine Plant Radioactive Effluents, Plant Contamination, and Solid Waste
10. Control of Occupational Doses
11. Organization
12. Initial Startup Programs

Audit/inspection of Applicant Records

- Calculations
- Analyses
- P&IDs
- System Descriptions
- Design Drawings
- Design Specs
- Procurement Specs
- Probabilistic Risk Assessment

Additional Portions of Application

- Technical Specifications
- Technical Requirements Manual
- Quality Assurance Plan (design)
- Fire Protection Program (design)
- Quality Assurance Plan (construction and operations)
- Emergency Plan
- Physical Security Plan
- SNM physical protection program
- SNM material control and accounting plan
- Cyber Security Plan
- Fire Protection Program (operational)
- Radiation Protection Program
- Offsite Dose Calculation Manual
- Inservice inspection/Inservice testing (ISI/IST) Program
- Environmental Report
- Site Redress Plan
- Exemptions, Departures, and Variances
- Facility Safety Program (under consideration for Part 53 applications)

- Safety Analysis Report (SAR) structure based on clean sheet approach

*Additional contents of application outside of SAR are still under discussion. The above list is draft and for illustration purposes only.

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Background (cont.)

- The guidance in this ISG can be applied to any non-LWR, small modular LWR or stationary micro-reactor applying for an operating licensing (OL) and combined license (COL) under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 or 52.

- This document does not provide guidance regarding the licensing requirements for fire protection programs needed to be in place prior to receipt of byproduct, source, or special nuclear material under 10 CFR Parts 30, 40, and 70. A construction permit (CP) applicant may address these fire protection licensing requirements within its CP application (in accordance with 10 CFR 50.31) or separately from the CP application.

- The ISG will be updated to apply to applications under 10 CFR Part 53, when 10 CFR Part 53 is issued.

Requirements

- 10 CFR 50.48(a), which requires that each operating nuclear power plant have a fire protection plan that meets the requirements of either 10 CFR Part 50, Appendix A, Criterion 3 for LWRs or the applicant's proposed principal design criteria that have been deemed acceptable by the NRC. Proposed principal design criteria are required per regulations in 10 CFR 50.34 and 10 CFR 52.79.
- 10 CFR 50.48(c), risk-informed, performance-based fire protection program requirements, which incorporates by reference National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition," with certain exceptions.*

*Although 10 CFR 50.48(c) is not applicable to non-LWRs, stationary micro reactors, and small modular LWRs, elements and concepts in NFPA 805 can be applied to these reactor types with justified exceptions and/or deviations, where appropriate.

Referenced Guidance

- RG 1.189, Revision 4, "Fire Protection for Nuclear Power Plants," which is a comprehensive fire protection guidance document, identifies the scope and depth of fire protection that the NRC staff would consider acceptable for nuclear power plants.
- NFPA 804, "Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants."
  - ➢ Provides useful information when used in conjunction with NRC regulations and guidance – the NRC has not formally endorsed NFPA 804, and some of the information in the NFPA standard may conflict with regulatory requirements.
  - ➢ Provides a deterministic approach to the fire protection program.
- NFPA 805, 2001 Edition.
- NFPA 806, "Performance-Based Standard for Fire Protection for Advanced Nuclear Reactor Electric Generating Plants Change Process."

Referenced Guidance (cont.)

- RG 1.205, Revision 2, “Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants,” which provides NRC guidance on an acceptable approach to meeting 10 CFR 50.48(c).
  - Note that Nuclear Energy Institute (NEI) published implementing guidance for the specific provisions of NFPA 805 and 10 CFR 50.48(c) in NEI 04-02, “Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c).” RG 1.205, Revision 1, endorsed, with clarifications and exceptions, NEI 04-02, Revision 2. RG 1.205, Revision 2~~.~~, updates the previous staff positions in RG 1.205, Revision 1, and endorses NEI 04-02, Revision 3.

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Scope

- The scope of this ISG addresses the review of the application content regarding the fire protection program for operations including application descriptions of:
  - Management policy and program direction and the responsibilities of those individuals responsible for the program/plan's implementation.
  - The integrated combination of procedures and personnel that will implement fire protection program activities.

- This ISG does not address fire protection system design or the fire hazards analysis within the probabilistic risk assessment (PRA).

Objectives

- The fire protection program description should establish the fire protection policy for the protection of safety-significant* SSCs at each plant and the procedures, equipment, and personnel required to implement the program at the plant site.

*10 CFR 50.48 and Appendix A to 10 CFR Part 50 both refer to structures, systems, and components (SSCs) that are “important-to-safety”; in place of this term, this ISG refers to “safety-significant” SSCs to be consistent with terminology used in TICAP and ARCAP reference documents.

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Objectives (cont.)

- The fire protection program description should:
  - Describe the overall fire protection program for the facility;
  - Identify the various positions within the licensee's organization that are responsible for the program;
  - State the authorities that are delegated to each of these positions to implement those responsibilities; and
  - Outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage.

Objectives (cont.)

- The program description should also describe specific features necessary to implement the program such as:
  - Administrative controls and personnel requirements for fire prevention and manual fire suppression activities; and
  - The means to limit fire damage to safety-significant SSCs so their capability to perform safety functions is maintained.

Guidance Topics

- Applicable Industry Codes and Standards
- Organization, Staffing, and Responsibilities
- Fire Protection Staff Training and Qualification
- General Employee Training
- Fire Brigade Training and Qualification
- Chemical Fires (Training and Qualification Program related to the Unique Features of Chemical Fires associated with the Reactor Design)
- Fire Protection Program Documentation and Changes, Configuration Control and Quality Assurance
- Verification and Validation (V&V) of Fire Models

Guidance Topics (cont.)

- Review of physical plant changes and procedure changes for impact on the fire protection program
- Inspection, testing, and maintenance for fire protection systems and features credited by the fire protection hazard analysis
- Monitoring program
- Defined strategies for fighting fires
- Reporting
- Control of combustibles, hazardous materials, and ignition sources
- Housekeeping
- Manual firefighting capabilities

Acceptance Criteria

- The fire protection program description meets applicable Regulatory Positions of RG 1.189 or provides a basis for any deviations.
- The program description provides clear delineations of organization, staffing, and responsibilities, consistent with RG 1.189.
- The fire protection program change process identifies changes that require prior NRC review and approval, consistent with RG 1.189 and RG 1.205.
- The monitoring program includes bases for failure probability assumptions used in the fire PRA, methods used to monitor availability, reliability, and performance of fire protection systems and features, and processes for identifying and implementing corrective actions (NFPA 805 and RG 1.205 are referenced).

Acceptance Criteria (cont.)

- The V&V of fire models is consistent with RG 1.189 and RG 1.205.
- The fire protection program QA requirements are consistent with RG 1.189.
- The fire protection program description adequately addresses the evaluation of compensatory measures for interim use for adequacy and appropriate length of use.
- Training of fire protection staff, fire brigade staff, and general employees regarding the fire protection program is consistent with the applicable sections of RG 1.189 and NFPA 805.
- Note that NFPA 804 and NFPA 806 are not specifically referenced in the Acceptance Criteria, but the ISG acknowledges that the use of these or other referenced standards allows an applicant a way to describe what they have done without having to include a detailed description in their submittal.

QUESTIONS?

Break

Meeting will resume at 1pm EST

Microsoft Teams Meeting

Bridgeline: 301-576-2978

Conference ID: 970 740 540#

NRC-CNSC MOC

Technology Inclusive and Risk-Informed Reviews for Advanced Reactors:

Comparing the US Licensing Modernization Project with the Canadian Regulatory Approach

Chantal Morin, P. Eng
Canadian Nuclear Safety Commission

Eric Oesterle, Senior PM
Nuclear Regulatory Commission

September 29, 2021

OUTLINE

- Background on NRC-CNSC MOC
- Key Messages
- Report Overview and Background
- Comparison observations and conclusions

NRC-CNSC MOC - Background

- On August 15, 2019, the Canadian Nuclear Safety Commission (CNSC) and the Nuclear Regulatory Commission (NRC), signed a Memorandum of Cooperation (MOC) to increase collaboration on the technical reviews of advanced reactor and small modular reactor technologies.

- Under the MOC, the NRC and CNSC plan to issue joint reports that cover topics that are generally applicable to advanced reactor developers and designers, as well as targeted reports addressing specific technical aspects for individual vendors or potential applicants.

NRC-CNSC MOC - Background

- The NRC and CNSC approved several work plans under the MOC which began collaborative efforts in mid-2020
- Work plans included specific topical areas and technical topics associated with specific advanced reactor and small modular reactor vendors as well as work plan focused on process
- Since the initiation of the MOC, the NRC and CNSC have completed a joint technical report associated with the X-energy design and a report comparing the technology-inclusive, risk-informed NRC and CNSC review approaches for advanced reactors

Public website links:

https://www.nrc.gov/reactors/new-reactors/nrc-cnsc-memorandum-of-cooperation.html

https://nuclearsafety.gc.ca/eng/resources/news-room/feature-articles/Sharing-our-expertise-with-the-US-Nuclear-Regulatory-Commission.cfm

Key Messages

- Both the Canadian and U.S. regulatory approaches effectively protect the health and safety of the public and the environment during nuclear power plant operation.

- Both the Canadian and U.S. regulatory approaches include some degree of technology-inclusive, risk-informed, performance-based review.

- Comparing the Canadian and U.S. regulatory approaches is a first step in enhancing understanding of each regulator’s framework to enable greater cooperation and collaboration on advanced reactor designs that are contemplated for deployment in both countries.

- The existing common ground between the Canadian and U.S. regulatory approaches can be leveraged to pursue joint technical reviews of advanced reactor applications in the future.

Historical Overview of LMP

- The recently completed report specifically focused on the **Licensing Modernization Project (LMP)** process and the CNSC review approach for Advanced Reactors.

- LMP was a U.S. Department of Energy (DOE) funded and nuclear industry led effort to develop a **technology-inclusive, risk-informed, and performance-based (TI-RIPB) approach** to developing a safety case.

- Elements of the LMP approach were developed over two decades and the completed process was finalized in NEI 18-04, Rev. 1, dated August 2019.
  - Next Generation New Plant (NGNP)
  - table-top exercises with advanced reactor developers vetted and informed LMP process
  - Several separate DOE reports formed basis for NEI 18-01, Rev. 1

- NEI 18-04 was endorsed in **Regulatory Guide (RG) 1.233** in June 2020.


NEI | TECHNICAL REPORT

Modernization of Technical Requirements
for Licensing of Advanced Non-Light Water Reactors

Risk-Informed Performance-Based Technology
Inclusive Guidance for Non-Light Water Reactor
Licensing Basis Development

Report Revision 1

August 2019


U.S. NUCLEAR REGULATORY COMMISSION
REGULATORY GUIDE 1.233 REVISION 0

GUIDANCE FOR A TECHNOLOGY-INCLUSIVE, RISK-INFORMED, AND PERFORMANCE-BASED METHODOLOGY TO INFORM THE LICENSING BASIS AND CONTENT OF APPLICATIONS FOR LICENSES, CERTIFICATIONS, AND APPROVALS FOR NON-LIGHT-WATER REACTORS

A. INTRODUCTION

NRC public website link: https://www.nrc.gov/reactors/new-reactors/advanced/details.html#flexLearn

REPORT - Overview

- Purpose
- Overview of Regulatory Processes for New Designs
  - Pre-application Interactions
  - Application Interactions
  - Overview of Regulatory Safety Objectives and Dose Limits
  - Technology inclusive, risk-informed, performance-based licensing approaches
- Assessment of Technology-Inclusive, Risk-Informed Approaches
- Suggestions for Future Work
- Conclusions

Technology Inclusive and Risk-Informed Reviews for Advanced Reactors:
Comparing the US Licensing Modernization Project with the Canadian Regulatory Approach

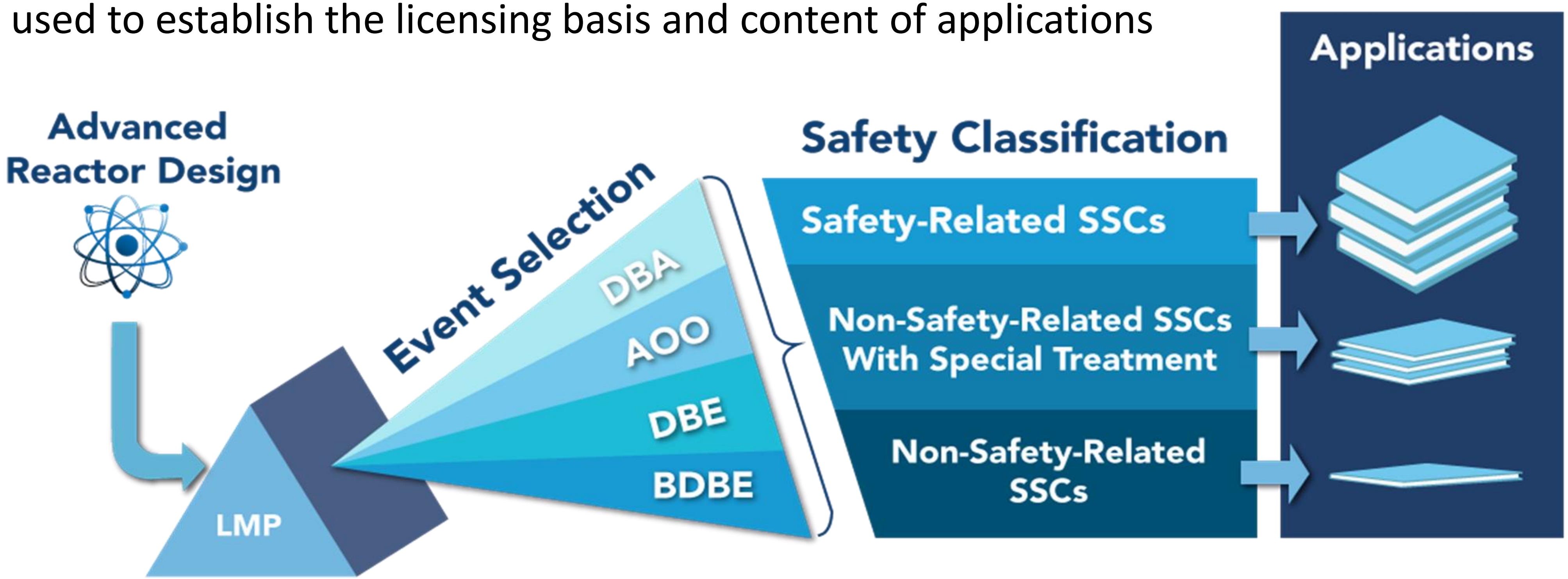
Public website links:

https://www.nrc.gov/reactors/new-reactors/nrc-cnsc-memorandum-of-cooperation/joint-reports-of-the-nrc-and-cnsc.html

https://nuclearsafety.gc.ca/eng/resources/news-room/feature-articles/Sharing-our-expertise-with-the-US-Nuclear-Regulatory-Commission.cfm

Background on NRC LMP Approach

- NRC's LMP approach is a **technology-inclusive, risk-informed, and performance-based (TI-RIPB) approach** for non-LWR advanced reactors
- NRC's LMP approach is frequency-consequence oriented approach using PRA that can be used to establish the licensing basis and content of applications

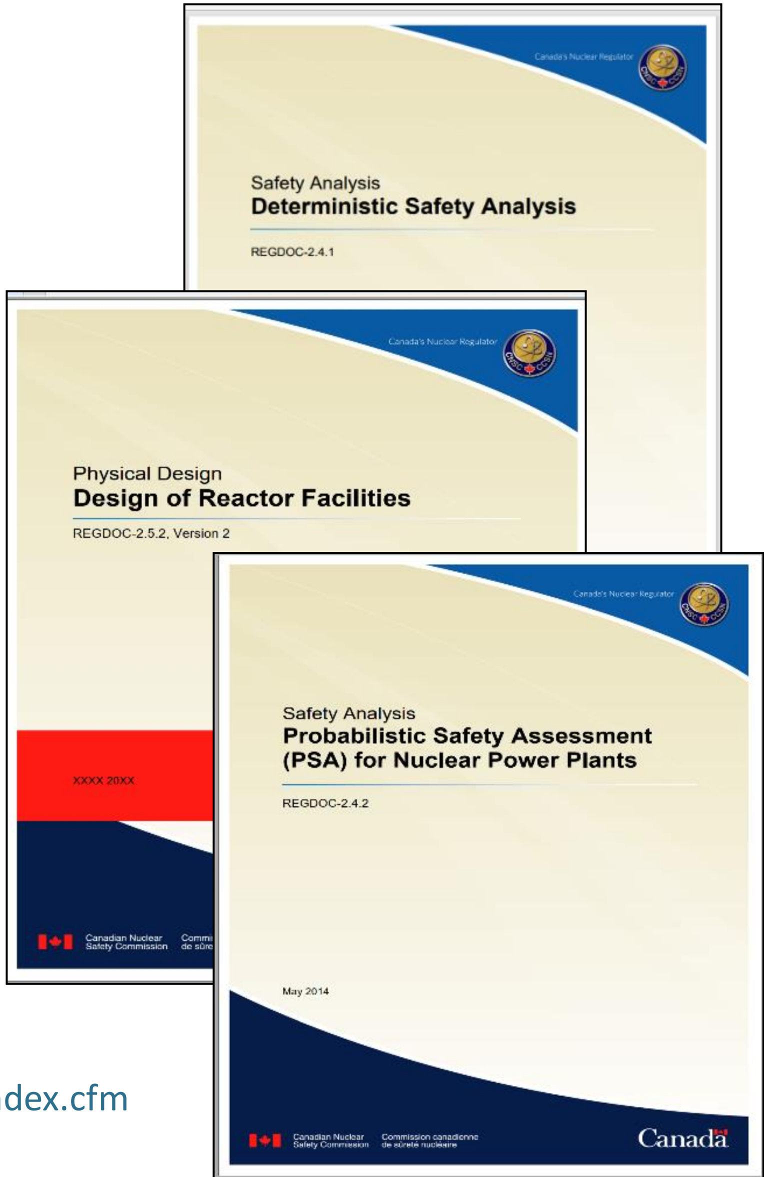


NRC public website link: https://www.nrc.gov/reactors/new-reactors/advanced/details-1.html

Background on CNSC Approach

- CNSC has a long history of risk-informed approach
  - aligned with IAEA
  - generally not very prescriptive

- Regulatory Documents describe expectations for:
  - Postulated Initiating Events determination
  - Safety Analysis: both deterministic and probabilistic
  - Safety System Classification
  - Defence in Depth, etc.



CNSC public website link: http://www.nuclearsafety.gc.ca/eng/acts-and-regulations/regulatory-documents/index.cfm

Report – General Observations

High level similarities: Technology-inclusive, risk-informed approaches

- Safety analysis process includes both probabilistic and deterministic methods
  - PRA* is foundational to the NRC LMP approach; PRA is also an important part of Canadian approach
  - Multi-unit event sequences are addressed
- Safety goals and objectives
  - Demonstrate that frequently occurring plant events have minor consequences
  - Demonstrate that events with severe potential consequences have a very low frequency of occurrence
- Licensing basis events/postulated initiating events
  - Abnormal Operation Occurrences (AOOs)
  - Design Basis Events (DBEs)/Design Basis Accidents (DBAs)
  - Beyond Design Basis Events (BDBEs)/Beyond Design Basis Accidents (BDBAs)

*NOTE: Probabilistic Risk Assessment (PRA) is the US terminology for Probabilistic Safety Assessment (PSA) – interchangeable

Report – General Observations

High level similarities (continued):

- Fundamental safety functions (3C's: Control, Cool, Confine)

US NRC	CNSC
Reactivity and Power Control	• Control of reactivity • Monitoring of safety-critical parameters to guide operator actions
Heat Removal	• Removal of heat from the fuel
Radionuclide Retention	• Confinement of radioactive material • Control of operational discharges and hazardous substances, as well as limitation of accidental releases
	• Shielding against radiation

- Defense-in-Depth (DID) adequacy
  - approach generally consistent with IAEA concept of layers of defense
  - layers associated with design, programs, procedures, risk assessments
  - no overreliance on a single layer of defense

Report – General Observations

High level differences:

- PRA and Risk Metrics
  - CNSC uses Level 2 PRA results;
  - NRC LMP approach uses Level 3
  - NRC LMP approach includes some cumulative risk metrics
- Threshold levels and definitions for LBEs
  - DBE vs. DBA
  - BDBE vs. BDBA
- Dose limits associated with DBEs/DBAs
  - NRC – 25 rem (max)
  - CNSC – 2 rem

Report – General Observations

High level differences:

- Single failure criterion
  - Embedded within NRC LMP DID assessment
  - CNSC maintains use of SFC and includes some limited relaxation flexibility

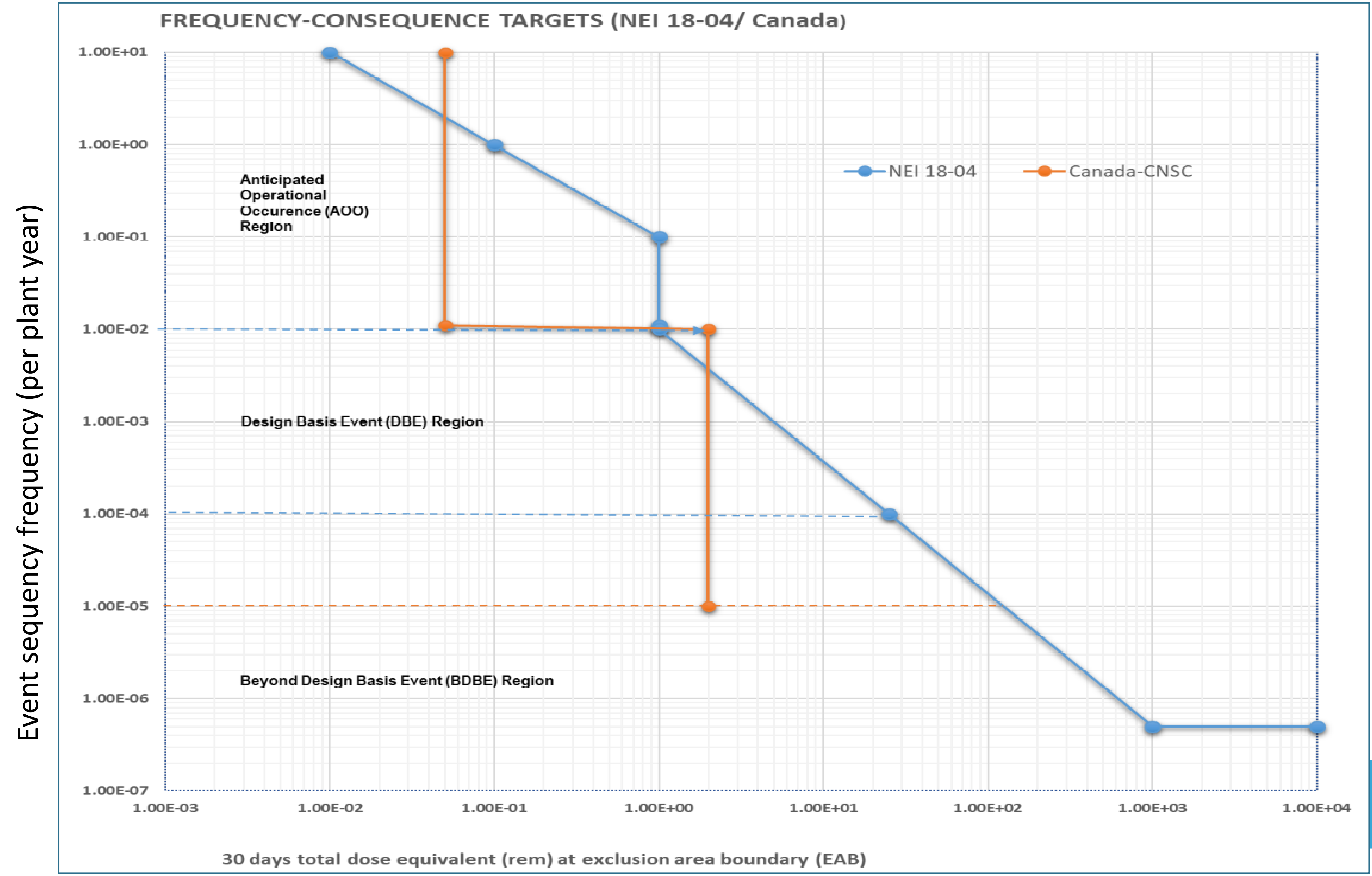
High level observation:

- Safety classification of SSCs
  - PSA metrics or DID importance
  - CNSC accepts licensee proposals on safety classification

NRC LMP	CNSC (flexible to applicant proposals)
Safety Related	Important to Safety
Not Safety Related with Special Treatment	Important to Safety
Not Safety Related	Not important to Safety

Report – General Observations

Dose acceptance criteria and event classification



Event sequency frequency (per plant year)

Report – General Observations

Dose acceptance criteria and event classification

Event Category	NRC LMP		CNSC	
	Frequency Ranges (per year)	Dose Limits (Rem)	Frequency Ranges (per year)	Dose Limits (Rem) (Sv)
AOO	$>10^{-2}$	Frequency $1\text{-}10^{-1}$/y: **0.1 to 1** Frequency $10^{-1}\text{-}10^{-2}$/y: **1**	$>10^{-2}$	**5mRem** 0.5 mSv
DBA/DBE	**$10^{-2} – 10^{-4}$**	DBE: **1 – 25** DBA: 10 CFR 50.34 limit **25** Rem	**$10^{-2} – 10^{-5}$**	**2 Rem** **20 mSv**
BDBA/BDBE	**$10^{-4} – 5\text{x}10^{-7}$**	**25 – 750**	**$<10^{-5}$** **(no lower limit defined)**	**No dose limit defined, Safety Goals applied**

Report – General Observations

- The NRC LMP approach is more similar to the CNSC's approach than the traditional NRC licensing approach for LWRs.

- Although consistent at a high level, NRC's LMP approach uses PRA more extensively while CNSC's approach considers other factors in decision making (operating experience, regulatory authority input, etc.).

Report – General Conclusions

- Excellent exercise for both regulators to become aware of each others approaches for new reactor safety reviews
- Benefits the facilitation of future cooperative efforts on joint technical reviews
- Better general understanding at the CNSC of this new approach which could facilitate the conduct of some Vendor Design Reviews
- Ongoing work plans under the CNSC-NRC Memorandum of Cooperation (MOC) involve joint reviews of limited technical topics for specific advanced reactor designs that will further develop our capabilities to perform joint technical reviews

THANK YOU

Questions?

ACRONYMS

AOO: Anticipated Operational Occurrence
BDBE: Beyond Design Basis Events (BDBA also)
DBA: Design Basis Accident (only safety related system are credited)
DBE: Design Basis Events (all systems are credited)
DEC: Design Extension Conditions (subset of BDBA)
DiD: Defence in Depth
F-C: Frequency-Consequence
LBE: Licensing Basis Events
LMP: Licensing Modernization Project
TI-RIPB: Technology-Inclusive Risk Informed Performance Based
PIE: Postulated Initiating Event
PSA/PRA: Probabilistic Safety Assessment/ Probabilistic Risk Assessment
QHO: Quantitative Health Objectives
SSC: Structures, Systems, Components

Draft NUREG-2246, Fuel Qualification for Advanced Reactors

Discussion of Comments Received during Public Comment Period (June 30-August 30, 2021)

September 29, 2021

Background

- Early Stakeholder Engagement
  - Framework presented at May 7, 2020, periodic advanced reactor stakeholder meeting
  - Draft white paper released in support of October 1, 2020, periodic advanced reactor stakeholder meeting
- International coordination with the Nuclear Energy Agency (NEA) – Working Group on the Safety of Advanced Reactors (WGSAR)
  - Regulatory Perspectives on Nuclear Fuel Qualification for Advanced Reactors
- Draft NUREG-2246 is an iteration of draft white paper, with adjustments to address feedback received at stakeholder meetings and NEA-WGSAR input

Advanced Reactor Program

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Purpose and Considerations

- NUREG-2246 provides a framework for evaluating a nuclear fuel design and attempts to enable a transparent, efficient, and thorough safety review
- Informed by staff experience gained from licensing solid fuel reactor designs (particularly LWR designs), advanced reactor fuel testing performed to-date, and the accelerated fuel qualification (AFQ) considerations
- Focused on areas where irradiated fuel tests have been required

Purpose and Considerations

- An attempt was made to develop generically applicable criteria
  - Some criteria may not be applicable to all fuel types or reactor designs
  - Criteria may not be sufficient in some cases
- Additional activity
  - Separate guidance being developed for molten salt reactors
  - NUREG-2246 is being exercised for a generic assessment of metal fuel (Idaho National Lab) and TRISO fuel (Pacific Northwest National Lab)

NUREG-2246, Comment Submittal Received

- Draft NUREG-2246, Fuel Qualification for Advanced Reactors
  - FRN published on June 30, 2021
- Three comment submittals received
  - (1) Public, ML21243A353
  - (2) Nuclear Energy Institute (NEI), ML21243A356
  - (3) Public, ML21246A124
- NEI's comment submittal requested a public meeting to further discuss the fuel qualification guidance

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Status of Comment Review

Submitter	Comments	Review Status	Changes to NUREG-2246
1-Public	1	Complete	None
2-NEI	30	Complete	17 comments resulted in edits to NUREG-2246, Revisions are still being developed
3-Public	8	5 of 8 Comments Reviewed	Ongoing

Outline of Highlighted Comments

Comment	Topic
NEI-1, NEI-24, Public 3-1	Non-LWR examples
NEI-5	Role of research literature
NEI-7	Regulatory basis for fuel qualification
NEI-9	Scope of degradation mechanisms
NEI-10	GDC/ARDCs
NEI-14	Fuel safety functions
NEI-16	First core applications

Comment	Topic
NEI-17	TRISO manufacturing
NEI-18	Safe shutdown
NEI-23	Coolable geometry
NEI-26	Control element insertion
NEI-29	Test envelope and Performance envelope
NEI-30	Quality assurance

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NEI Comment #1 and #24 and Public Comment 3-1, General

There are numerous advanced reactor designs, with many different fuel designs, and each fuel design has its own unique challenges and advantages. This draft NUREG has a lot of guidance heavily based on existing LWR fuel designs, which may not be applicable to all advanced reactor designs. Please make the guidance more applicable to non-traditional fuel designs.

Please add examples for non-traditional LWR fuel designs.

<u>Proposed Response Summary</u>:

- Staff acknowledges that NUREG-2246 is informed by lessons-learned from experience with LWR fuel but disagrees with the characterization that it is "based on existing LWR fuel designs."
- Section 1.3, "Scope," states that assessment criteria draws from results from advanced reactor fuel testing performed to-date, and accelerated fuel qualification (AFQ) considerations
- On-going work to exercise NUREG-2246 framework through generic assessments of metal fuel (Idaho National Lab) and Tristructural Isotropic (TRISO) fuel (Pacific Northwest National Lab)

NEI Comment #5, Page 1-1

The NUREG primarily references the Crawford report when establishing the objective of fuel qualification and its corresponding basis. A technical paper (the Crawford report) does not establish a regulatory basis and is not appropriate to establish the regulatory need for fuel qualification. Furthermore, the noted economic operation in the Crawford report is not something that NRC should be regulating on.

Please replace the reference of a technical paper with a regulatory one (when identified as a means to do so) and remove reference to economics as a regulatory goal.

Proposed Response Summary:

- Staff disagrees with the characterization that a technical paper served as a primary basis for NUREG-2246, but acknowledges that the research by Crawford, Porter, Hayes, Meyer, and Petti informed their judgement on an applicable definition of “fuel qualification,” for use in NUREG-2246.
- Definition in Section 3 was selected because its an accurate characterization of fuel qualification based on the NRC regulations, staff’s experience from licensing solid fuel reactor designs, results from advanced reactor fuel testing performed to-date, and AFQ considerations.

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NEI Comment #7, Page 1-1

The text states "this framework relies on regulatory requirements that are applicable to applications for design certifications, combined licenses, manufacturing licenses, or standard design approvals." However, fuel qualification itself is not required for any of the mentioned licensing approvals. Fuel qualification is therefore only necessary to meet the requirements in Section 2.1 to the extent that the fuel is specifically relied upon as a safety feature.

Please clarify the connection of fuel qualification to the noted licensing approvals, or modify the text to more clearly denote more context to how potentially applicable requirements would inform what is necessary for fuel qualification.

<u>Proposed Response Summary</u>:

- The term “fuel qualification” is not explicitly defined or used in NRC regulations.
- There are regulatory requirements generically applicable to power reactor applications that are generally associated with nuclear fuel behavior under conditions of normal operation, including the effect of anticipated operational occurrences, and accident conditions.
- Regulatory requirements and connections to regulation is provided in Section 2.1, “Regulatory Basis”

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NEI Comment #9, Page 1-2 and Section 3.1.1

Fuel life-limiting failure and degradation mechanisms are not just due to irradiation during reactor operation. Other degradation mechanisms, chemical attacks, hydrogen pickup, high temperature, and time at temperature during AOOs or Design Basis Accidents also impact fuel performance.

Please remove "due to irradiation" to expand the applicability of the statement to include other failure mechanisms.

Proposed Response Summary:

- Propose change from “due to irradiation” to “due to irradiation and irradiation assisted phenomena.”
- Agree that there are degradation mechanisms beyond irradiation
  - Primary obstacle to qualifying nuclear fuel has generally been demonstrating fuel performance at the desired exposure
  - most fuel degradation phenomena are impacted by irradiation

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NEI Comment #10, Section 2.1

The regulatory basis denoted is 50.43(e) and the design criteria (GDC and ARDC). However, the GDC and ARDC are not requirements for, and as guidance are not required to be met by, non-LWRs. Thus, non-LWRs may choose to develop PDCs through another method. The guidance should clarify that fuel qualification is only necessary if it is determined to be one of the PDCs for the design, based upon the fuel being relied upon as a safety feature.

Please clarify the text to indicate how fuel qualification could be used to demonstrate compliance to 50.43 but is not necessary if fuel is not relied upon as a safety feature. Additional context on how potentially applicable requirements would inform what is necessary for fuel qualification would be helpful.

<u>Proposed Response Summary</u>:

- Agree that GDCs and ARDCs are not necessarily requirements for non-LWRs but are instead considered guidance for non-LWR advanced reactor applicants in developing proposed PDCs.
- The cited GDC/ARDC are associated with safety functions generally involving nuclear fuel that are not otherwise captured in NRC regulations (e.g., fuel safety limits, maintaining coolable geometry) and are expected to be addressed as part of fuel qualification.
- Considering revisions to Section 2.1 to accurately reflect requirements associated with PDCs

NEI Comment #14, Section 2.2.3

The following text is misleading because fuel may have safety functions as noted in the text, but it is not required to. It is possible to not credit the fuel and instead credit other mechanisms outside of the fuel matrix. Fuel qualification is therefore only necessary to meet the requirements in Section 2.1 to the extent that the fuel is specifically relied upon as a safety feature.

"Fuel qualification partially addresses the fundamental safety functions of control of reactivity, cooling of radioactive material, and confinement of radioactive material…"

Please clarify the role of fuel qualification and its necessity only if being relied upon and/or credited in the safety analysis as some designs may not. Additional context on how potentially applicable requirements would inform what is necessary for fuel qualification would be helpful.

<u>Proposed Response Summary</u>:

- Staff recognizes that the role of fuel in the protection against the release of radioactivity can vary depending upon the reactor design.
- Considering revising text to clarify that the role of fuel in the safety functions can be addressed in a graded manner in accordance with the degree to which the fuel is credited.

NEI Comment #16, Section 2.4 and Section 3.4.2

The text here on lead test specimen programs is applicable only if we had existing/operating advanced reactors. It does not discuss alternatives for fuel to be qualified for first core applications where lead test specimens are not possible.

Please add information on fuel qualification for first core applications.

Proposed Response Summary:

- Considering adding a new Section 2.5, “First Core Applications” (and renumbering the current Section 2.5 to 2.6)

NEI Comment #17, Section 3.1.3

The TRISO SER allows for manufacturing independence as long as the final product has properties that fall within the specification range. Please note TRISO as an example of "insensitivity to manufacturing processes."

Please add text to denote that the TRISO is an example of fuel that has an insensitivity to manufacturing processes and instead measurable criteria can be used to justify predicted performance.

Proposed Response Summary:

- Revise Sections 3.1 and 3.1.3 to reference the EPRI TRISO Topical Report and associated safety evaluation report, reflecting that key end-state parameters for TRISO particles have been identified that provide assurance of fuel performance during normal operation

NEI Comment #18, Figures 3-3 and 3-8, Section 3.2.3

Criterion G2.3 includes a statement for the "ability to achieve and maintain safe shutdown can be assured." The NUREG further defines safe shutdown as "a state in which the reactor is subcritical, decay heat is being removed, and radionuclide inventory is contained." However, safe shutdown and safe state are not interchangeable. Not all reactors must be subcritical to be safe and therefore it is not necessary for all fuel types to be subcritical. Industry recommends aligning the NUREG with other documentation that use the phrase "a safe stable end-state" instead.

Please change criterion G2.3 to "Ability to achieve and maintain a safe, stable, end state" and revise the text throughout to be consistent with this revised criterion.

Proposed Response Summary:

- Relatively recent NRC policy papers have clarified that maintaining subcriticality with only safety-related structures, systems, and components may not be required (SECY-18-0099)
- Staff expects that nuclear fuel be designed such that forces on the nuclear fuel, resulting from internal or external events, will not preclude the eventual achievement of a subcritical state

NEI Comment #23, Page 3-8

No specific criterion is set as to what the term "coolable geometry" means for non-traditional LWR fuel designs. For example, some advanced reactor fuel designs like the MSFR and the MCFR are planning on using liquid salt with fissile product as both their coolant and their fuel, which would require clarification/flexibility on the definition of "coolable geometry."

Please clarify what a coolable geometry criterion would be for fuel designs that do not have a containment (e.g., LWR fuel cladding) and how this will ensure ability to attain safe, stable, end state.

Proposed Response Summary:

- Section 3.2.3.1.1, “G2.3.1(a) – Identification of Phenomena” provides examples of the types of phenomena that could cause a loss of coolable geometry.
- Staff recognizes that these criteria cannot be specified generically for all fuel designs
- Section 1.3, “Scope,” clarifies that some criteria may not apply to liquid fuel forms (e.g., MSR fuel), and these fuel forms may require additional or alternative criteria.

NEI Comment #26, Page 3-10

For designs that do not have neutron control element insertion (e.g., control drums), having criteria to ensure control element insertion paths would not be necessary.

Please revise text to indicate criteria should be specified only for designs with neutron control elements whose insertion is credited in accident response models.

Proposed Response Summary:

- Revise Section 3 to reference Section 1.3, “Scope”
  - The assessment criteria draws on regulatory experience gained from licensing solid fuel reactor designs (particularly LWR designs), results from advanced reactor fuel testing performed to-date, and AFQ considerations.
  - An attempt has been made to develop generically applicable criteria.
  - Staff recognizes that some criteria may not be applicable to all fuel types or reactor designs.

NEI Comment #29, Page 3-16

The text does not address methods for justifying when fuel can be used beyond its performance envelope when lead test specimens are not available.

Please add information on what adequate justification is needed to expand the performance envelope using experimental data without the use of lead test specimens.

Proposed Response Summary:

- Reference Section 3.3.2.2.4, "EM G2.2.4-Restricted Domain,"
  - *Application of an evaluation model outside of the supporting test envelope (see Section 3.4.2) may be justified based on physical arguments (e.g., that the evaluation model provides a simplified or bounding treatment of physical phenomena). Justification for extrapolation of a model outside of the test envelope is strengthened by the use of physics-based models, such as those discussed in Section 2.3, which are informed by fundamental information about fuel evolution and behavior, as opposed to empirically derived models (Terrani, et al., 2020).*

NEI Comment #30, Page 3-17

ASME NQA-1 is not the only way to qualify fuel data. For data collected at national laboratories, the application of this standard may not be possible, despite the national laboratories using alternative and acceptable quality assurance methods. NRC should accept data from the technical experts at national laboratories if the data was collected under the lab's QA program as noted in ML20054A297, where NRC staff determined that Argonne National Lab's quality assurance program plan is based on the method provided in ASME's NQA-1-2008/2009 and satisfies the quality assurance requirements of Appendix B to 10 CFR Part 50.

Please either remove or modify the text to allow for data to also be qualified under the commercial grade dedication (CGD) process rather than stating data must be made compliant.

<u>Proposed Response Summary</u>:

- Revise Section 3.4.3.1 to clarify that approaches other than those provide in ASME's NQA-1, including CGD, may be acceptable means for justifying that data is collected under an appropriate QA program

Next Steps

- Continue to develop formal responses to public comment submittals
- Additional public meeting(s), if needed
- Advisory Committee on Reactor Safeguards (ACRS) Full Committee Meeting
  - Tentatively scheduled for November 2021
- Issue final NUREG-2246
  - Expected to be published in January/February 2022
- Stakeholder engagement on on-going work to exercise NUREG-2246 framework—metal fuel and TRISO fuel

Advanced Reactor Program

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Break

Meeting will resume in 10 minutes

Microsoft Teams Meeting

Bridgeline: 301-576-2978

Conference ID: 970 740 540#

Advanced Reactor Program

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Scalable Human Factors Engineering Review Framework for Advanced Reactors

NRR – Division of Advanced Reactors and Non-power Production and Utilization Facilities

NRR – Division of Reactor Oversight

September 29, 2021

Agenda

- Background
- Advanced Reactor Implications for Human-System Integration
- Scalable HFE Framework
- Next Steps
- Questions/Comments

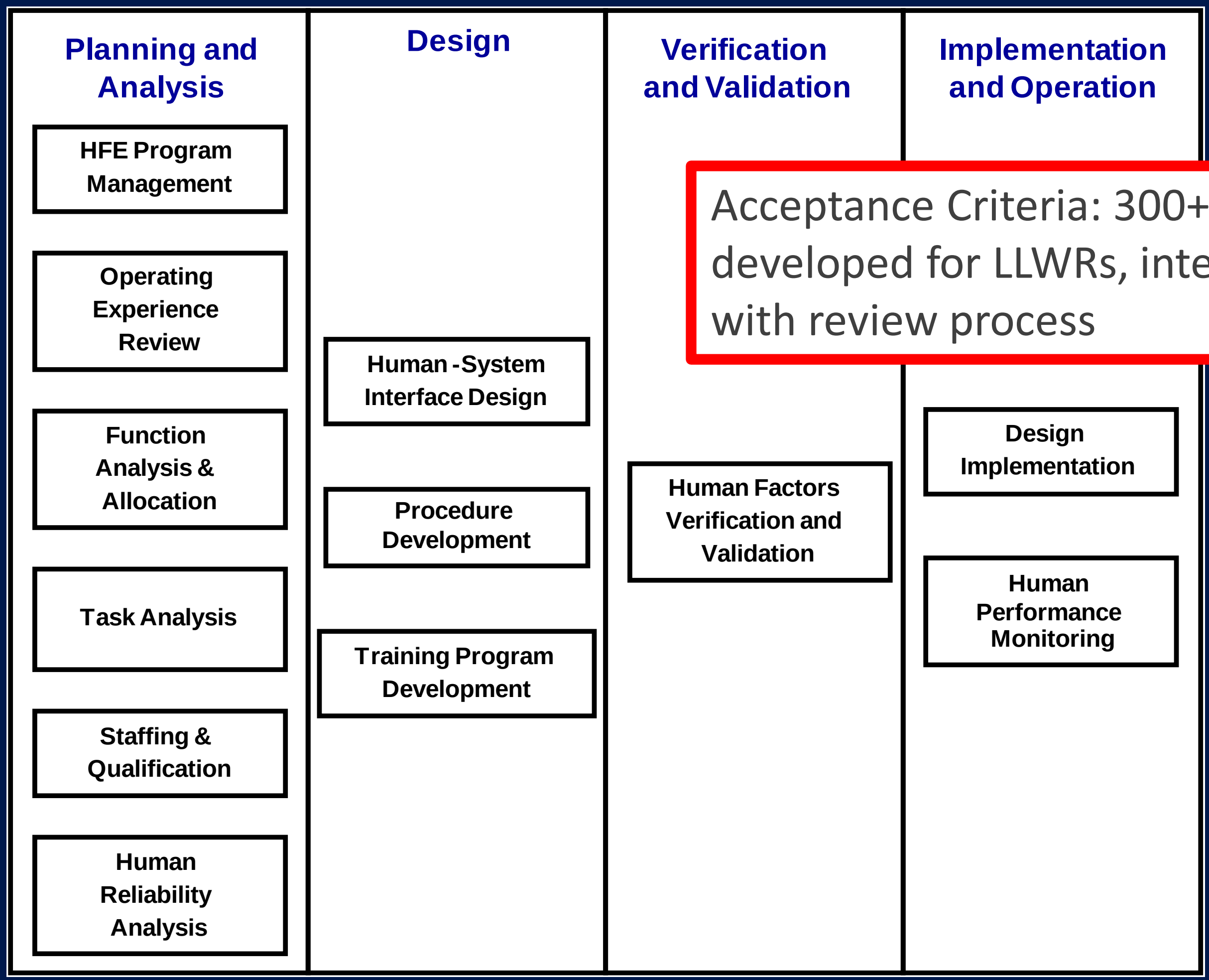
Current Framework

10 CFR 50.34(f)(2)(iii)

"Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. (I.D.1)"

Current Framework

NUREG-0711, Human Factors Engineering Program Review Model



Acceptance Criteria: 300+, developed for LLWRs, integrated with review process

Background

White Paper: Risk-Informed and Performance-Based Human-System Considerations for Advanced Reactors*

- Released March 25, 2021 (ML21069A003)
- Presented to advanced reactor stakeholders on April 8, 2021
- Presented as Key Guidance to ACRS S/C on May 20, 2021

> **This draft staff white paper has been prepared and is being released to support ongoing public discussions.**
>
> **This paper has not been subject to NRC management and legal reviews and approvals, and its contents are subject to change and should not be interpreted as official agency positions.**

Risk-Informed and Performance-Based
Human-System Considerations
for Advanced Reactors

Prepared by the
U.S. Nuclear Regulatory Commission,
Office of Nuclear Reactor Regulation,
Division of Reactor Oversight,
in conjunction with the
Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 2021

Background

White Paper: Risk-Informed and Performance-Based Human-System Considerations for Advanced Reactors*

Key messages:

- If an advanced reactor design presents very low radiological risk then the current regulatory framework for operation of large LWRs may be unnecessary for reasonable assurance of safety.
- A new regulatory framework for advanced reactors (10 CFR 53) should be capable of addressing novel operational concepts for a wide variety of advanced reactor technologies.
- A risk-informed, performance-based, and technology-inclusive regulatory framework for advanced reactors must appropriately consider the role of humans and **human-system integration**

Background

White Paper: Risk-Informed and Performance-Based Human-System Considerations for Advanced Reactors*

Topics included:

- diverse and novel operational characteristics, including automation of operations
- staffing and qualifications of operations personnel
- evolution in control room concepts

Implications for human-system integration

Passive Safety Features, Inherent Safety Characteristics & Automation

- Will be used to varying degrees and in different combinations among advanced reactor designs
- Achieve safety through different means
- Have different implications for the role of personnel in the assurance of plant safety

Implications for human-system integration

Automation of Plant Operations

- Automation is implemented in levels that span from manual to autonomous operation.
- Even in an autonomous design, there may still exist a need for humans to implement manual operations under certain circumstances, such as for defense-in-depth.
- Automation generally enhances operational performance, however other operational effects must be considered as well (e.g., operators losing manual control proficiency).

Implications for human-system integration

Staffing and Qualifications

- Applicants may propose alternative staffing models.
  - Number of personnel
  - Crew composition
  - Span of control / responsibility
- Operator licensing for advanced reactors may include
  - allowances for varying licensing examination scope on a facility-specific basis
  - modified simulator requirements.

Implications for human-system integration

The Evolving Concept of the “Control Room”

- Some advanced reactor facilities may wish to not utilize traditional control rooms
- Operations functions may become decentralized
- New missions for reactor facilities may be bring the emergence of functions that have no precedent within traditional control rooms

Implications for human-system integration

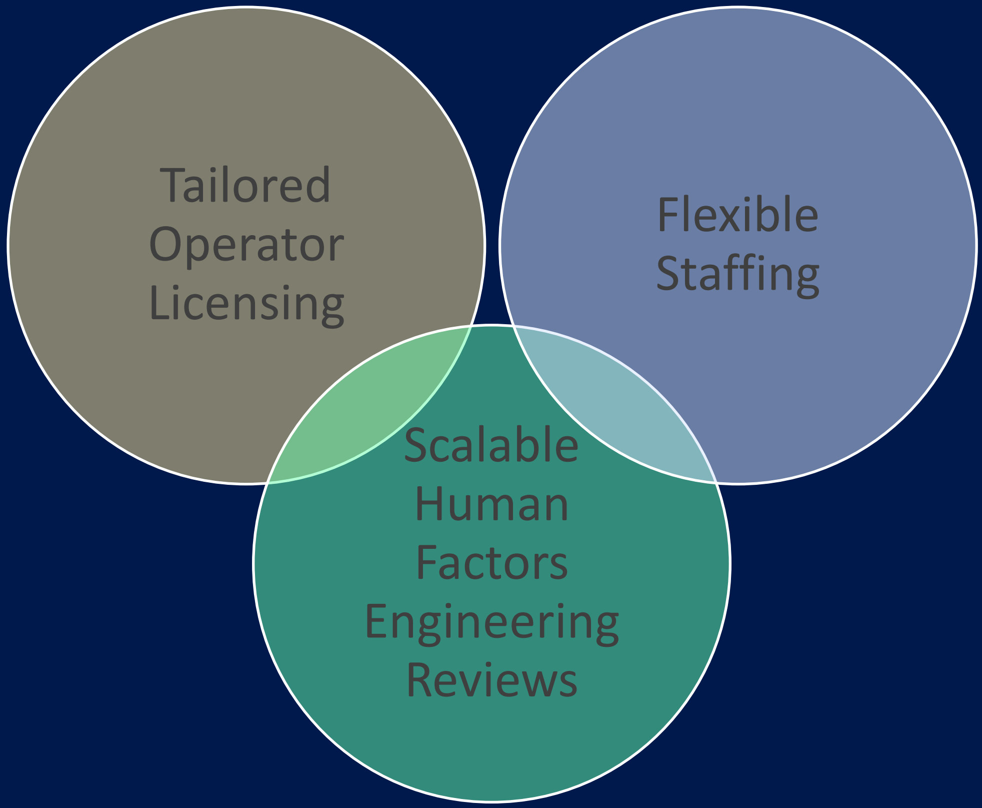
Summary

Advanced reactors are likely to vary in terms of the

- Role
- Number
- Qualifications
- Crew composition
- Work settings

of personnel responsible for plant operational safety

Part 53 working groups addressing human-system considerations



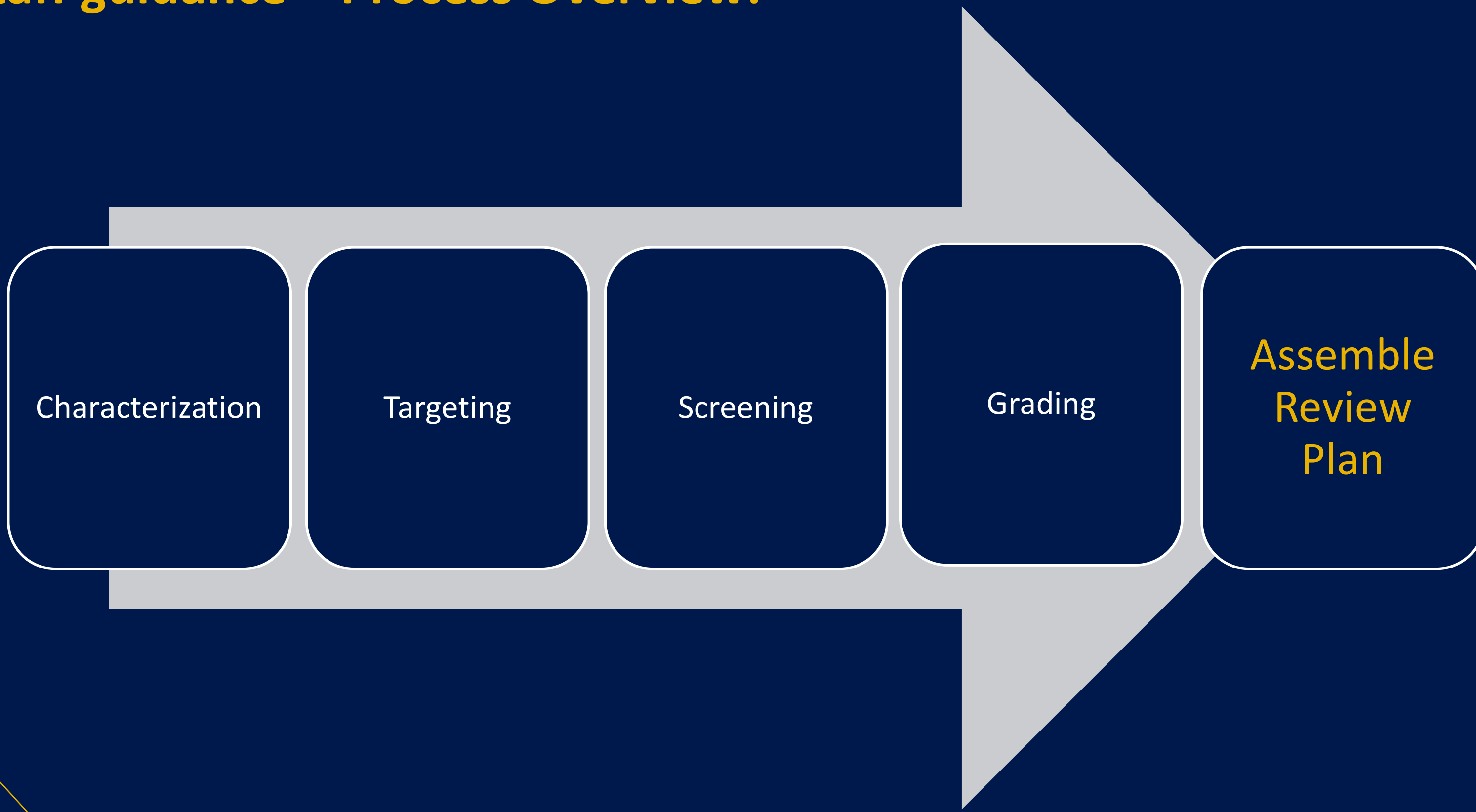
Scalable Human Factors Engineering Reviews

Technical Basis Development:

- Brookhaven National Laboratory contracted to develop a method for scaling the scope and depth of HFE reviews for advanced reactors (ML21266A192)
  - Review of advanced reactor design concepts
  - Identification of human performance considerations
  - Review of NRC's existing review plans / methods for non-LLWR nuclear technologies
- Method enables staff to readily adjust the focus and level of HFE review based upon risk/safety insights and the unique characteristics of the facility design/operation
- Staff developing guidance for scalable HFE review using BNL technical basis report

Scalable Human Factors Engineering Reviews

Staff guidance – Process Overview:



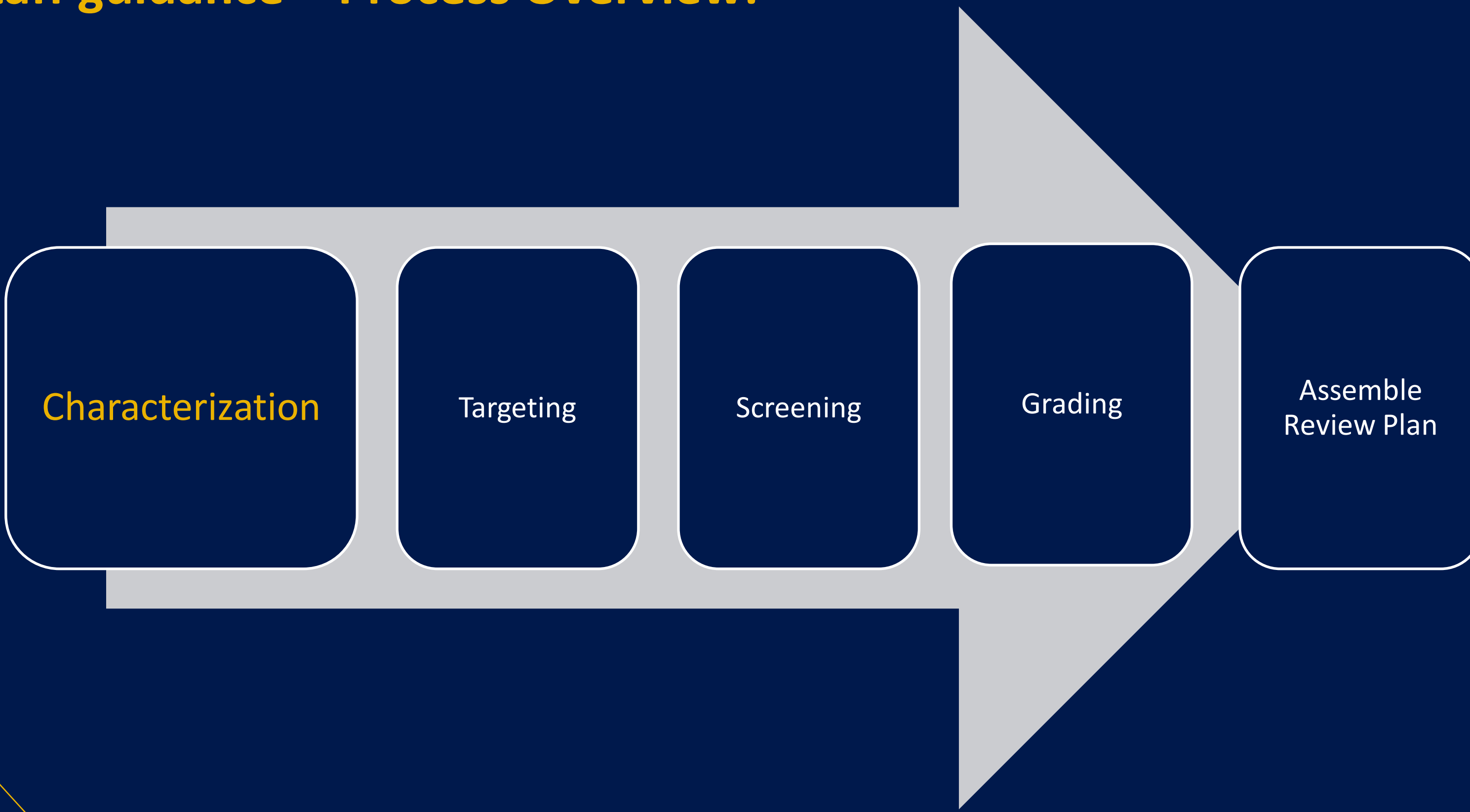
Scalable Human Factors Engineering Reviews

Staff Guidance :

- Objective: Provide a standard process for the development of an application-specific, risk-informed HFE review plan that is focused on those aspects of a design and its operation most important to the assurance of facility operational safety
- Relationship to other HFE guidance:
  - Does not replace existing NRC guidance (e.g., NUREGs 0711 and 0700)
  - Facilitates selective use of existing guidance and standards (e.g., IEEE, ANS) as justified by state-of-the-art

Scalable Human Factors Engineering Reviews

Staff guidance – Process Overview:



Scalable Human Factors Engineering Reviews

Characterization :

- Objective: Develop an understanding of a facility design and its operation from the perspective of their implications for human performance and human factors engineering
- Process: Staff review of applicant submittals, beginning pre-application when possible, to identify safety and risk insights and those features/characteristics of the design and its operation that will be important to informing the scope and depth of the HFE review
- Inputs:
  - Facility concept of operations
  - General safety and risk insights for the facility
  - Important human actions (from risk and deterministic analyses)
  - Human factors engineering activities conducted/planned
  - Compliance with CFR requirements / planned exemption requests

Scalable Human Factors Engineering Reviews

Characterization :

Concept of operations: A concept of operations (ConOps) defines the goals and expectations for the new system from the perspective of users and other stakeholders and defines the high-level considerations to address as the detailed design evolves. An HFE-focused ConOps addresses the following six dimensions:

- Plant Goals (or Missions)
- Agents' Roles and Responsibilities
- Staffing, Qualifications, and Training
- Management of Normal Operations
- Management of Off-normal Conditions and Emergencies
- Management of Maintenance and Modifications

Scalable Human Factors Engineering Reviews

Staff guidance – Process Overview:

Scalable Human Factors Engineering Reviews

Targeting:

- Objective: Identify aspects of the applicant's facility design and operations that warrant staff HFE review
- Process: Review the results of the characterization considering
  - risk importance
  - safety significance
  - uncertainty (e.g., potential unknowns introduced by novelty, lack of detail)
- Inputs: Characteristics of the facility design and operation

Scalable Human Factors Engineering Reviews

Targeting:

- Example characteristics:
  - New missions
  - New staffing positions
  - Safety function monitoring
  - Risk-important human actions
  - New hazards
  - High levels of automation

Scalable Human Factors Engineering Reviews

Staff guidance – Process Overview:

Scalable Human Factors Engineering Reviews

Screening:

- Objective: Identify the applicant's HFE activities that warrant staff review
- Process: Review the results of the characterization and targeting considering
    - HFE activities completed / planned by the licensee
    - Characteristics identified for targeting

  Assess HFE activities needed to establish technical basis for the use / design of targeted characteristics
- Inputs:
    - Characteristics of the design and operation targeted for review
    - HFE activities completed / planned by the licensee

Scalable Human Factors Engineering Reviews

Screening:

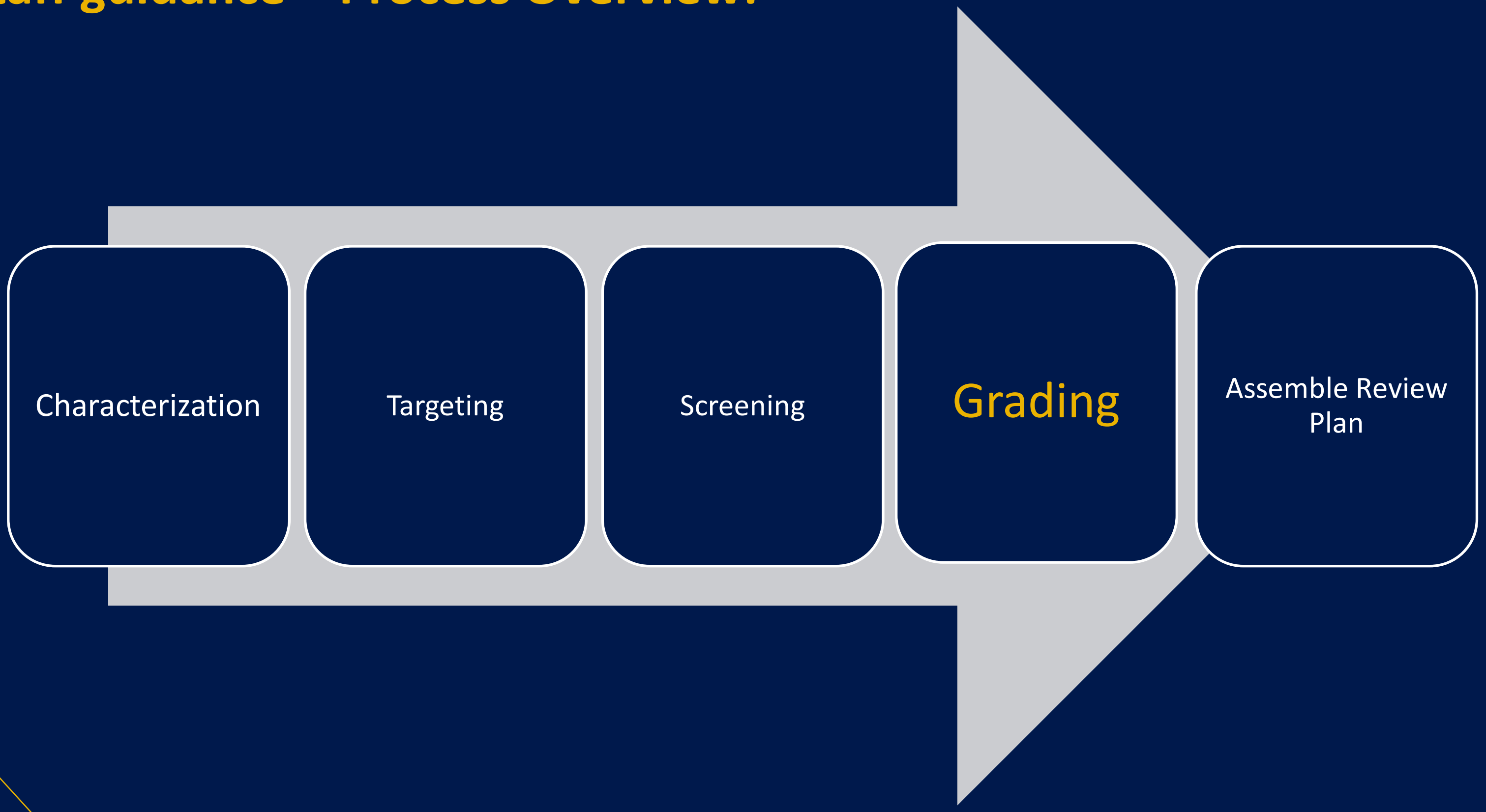
- Example:

Example	Targeted Characteristic	HFE Activity Selected for Review*
1	Novel technology	Operating experience review
2	New staffing model	Staffing plan validation
3	New job function	Task analysis
4	New mission	Integrated system validation

* Selected activities are for explanatory purposes only

Scalable Human Factors Engineering Reviews

Staff guidance – Process Overview:



Scalable Human Factors Engineering Reviews

Grading:

- Objective: Identify the standards and guidelines to be applied to the review of the applicant's submittals
- Process: Review the applicant's submittals to identify the HFE standards that the applicant has applied to their design, methods, and processes and assess the adequacy of the identified standard for their intended purpose. Establish the standards and guidelines to be applied in the staff HFE review.
- Inputs:
  - Characteristics of the design and operation targeted for staff review
  - Applicant's HFE activities selected by for staff review
  - HFE standards and guidelines cited by the applicant

Scalable Human Factors Engineering Reviews

Screening:

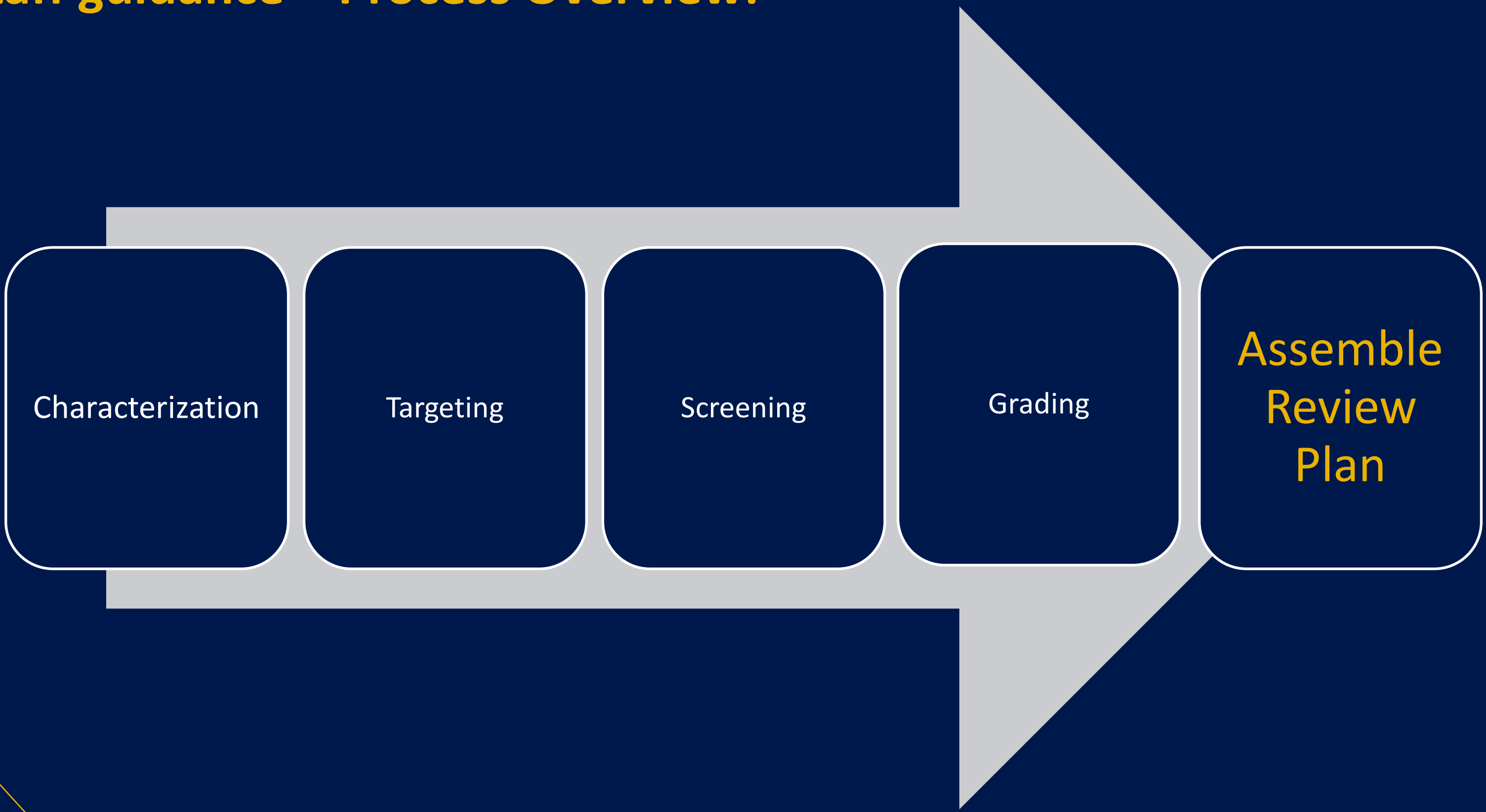
- Example:

Example	Characteristic / HFE Activity	Standard / Guideline Cited by Applicant	Standard / Guideline Selected by staff*
1	Alarm system	NUREG-0700 based style manual	NUREG-0700
2	Task Analysis	None	NUREG-0711, Chapter 5
3	Staffing	NUREG-1791	NUREG-1791 as amended for advanced reactors
4	Computer-based procedures	IEEE-1023	IEEE-1786

* Selected standards and guidelines are for explanatory purposes only

Scalable Human Factors Engineering Reviews

Staff guidance – Process Overview:



Scalable Human Factors Engineering Reviews

Assemble the review plan:

- Objective: Develop an application-specific plan for an HFE technical review of the applicant’s submittal.
- Process: Staff will review results of the targeting, screening and grading activities in conjunction with the applicant’s schedule for subsequent HFE activities and submittals to develop an integrated plan that optimizes the technical review, use of resources, and schedule for completion.
- Inputs:
  - Characteristics of the design and operation targeted for staff review
  - Applicant’s HFE activities selected by for staff review
  - HFE standards and guidelines to be applied to the review
  - Applicant’s schedule for the conduct of HFE activities and related submittals

Scalable Human Factors Engineering Reviews

SUMMARY:

- Encourages and facilitates early engagement between applicants and the staff
- Tailors the HFE technical review to the specific application using risk and safety insights
- Establishes a standard, technology inclusive process for scaling a review that does not presume a given form or scope of applicant HFE activities / program
- Enables the application of standards, guidelines, and criteria according to their fit for the facility, technology, method, or process under review

Next Steps

Guidance Development:

Activity	Target Date
Issue summary report for technical basis development	October 15
Complete draft guidance for scaling HFE technical review plans	November 30
Issue draft guidance for stakeholder review and feedback	December 15

Questions/Comments

Endorsement of ASME Section XI, Division 2, Reliability and Integrity Management (RIM) Programs for Non-Light Water Reactors

September 29, 2021

Steve Philpott, Project Manager
Tim Lupold, Senior Mechanical Engineer
Division of Advanced Reactors and Non-Power Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Objectives for Today's Presentation

- Alert Stakeholders that DG-1383 is publicly available for comment
- Overview the staff's review of ASME Code, Section XI, Division 2
- Familiarize stakeholders with ASME Code, Section XI, Division 2 (RIM)
- Present the unique regulatory aspects of preservice and inservice inspection (PSI / ISI) programs for advanced reactors
- Identify the conditions the staff has recommended in the draft guide on the use of ASME Code, Section XI, Division 2
- Communicate plans for public comment period and final regulatory guide

Review Team

The RIM Working Group

- Bruce Lin – Materials Engineer – RES/DE
- Ian Tseng – Mechanical Engineer – NRR/DEX
- Hanh Phan – Senior Reliability and Risk Analysis Analyst – NRR/DANU
- Steve Philpott – Project Manager – NRR/DANU
- Tim Lupold – Senior Mechanical Engineer – NRR/DANU
- Robert Roche-Rivera – Program Manager – RES/DE
- Steve Downey – Senior Reactor Inspector – R-II
- Isaac Anchondo-Lopez – Reactor Inspector – R-IV

And other SMEs as needed

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

RIM Review Project Schedule

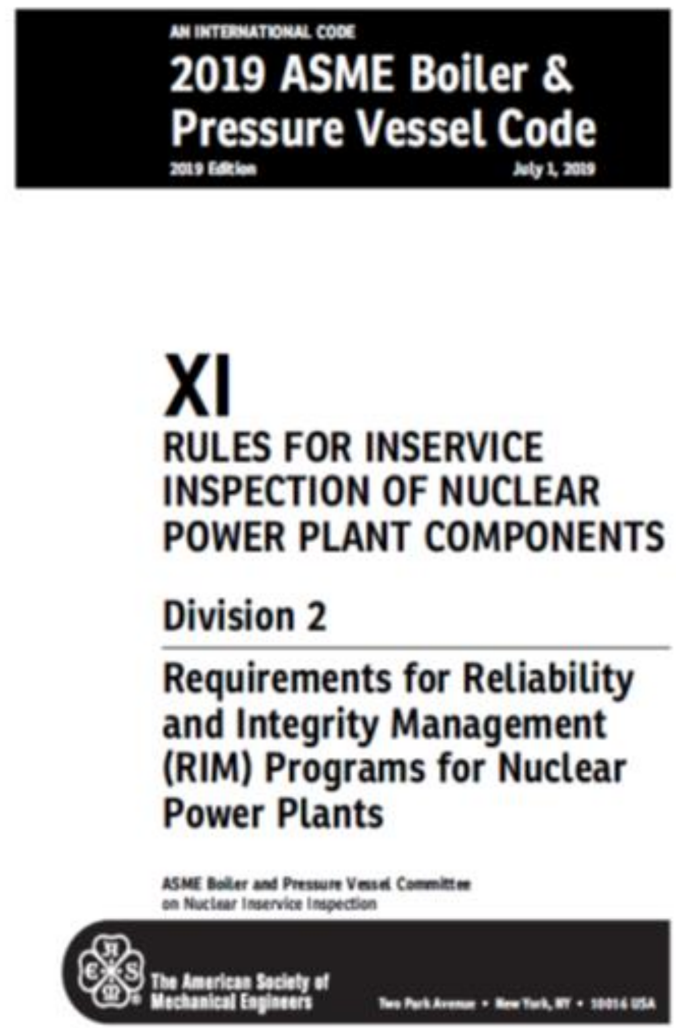
- Responded to ASME letter and initiated detailed staff review: Aug 2020
- Reviewed code and developed initial staff positions: Aug – Dec 2020
- Developed staff positions and draft regulatory guide (RG): Jan – Sep 2021
- Published draft RG: Sep 2021

Regulatory Guide Project Schedule

- Federal Register announcement of availability for public comment
- 45-day public comment period
- Public comment resolution
- Issue as Final Regulatory Guide shortly after next ASME Code Edition Rulemaking
  - Goal is by June 30, 2022

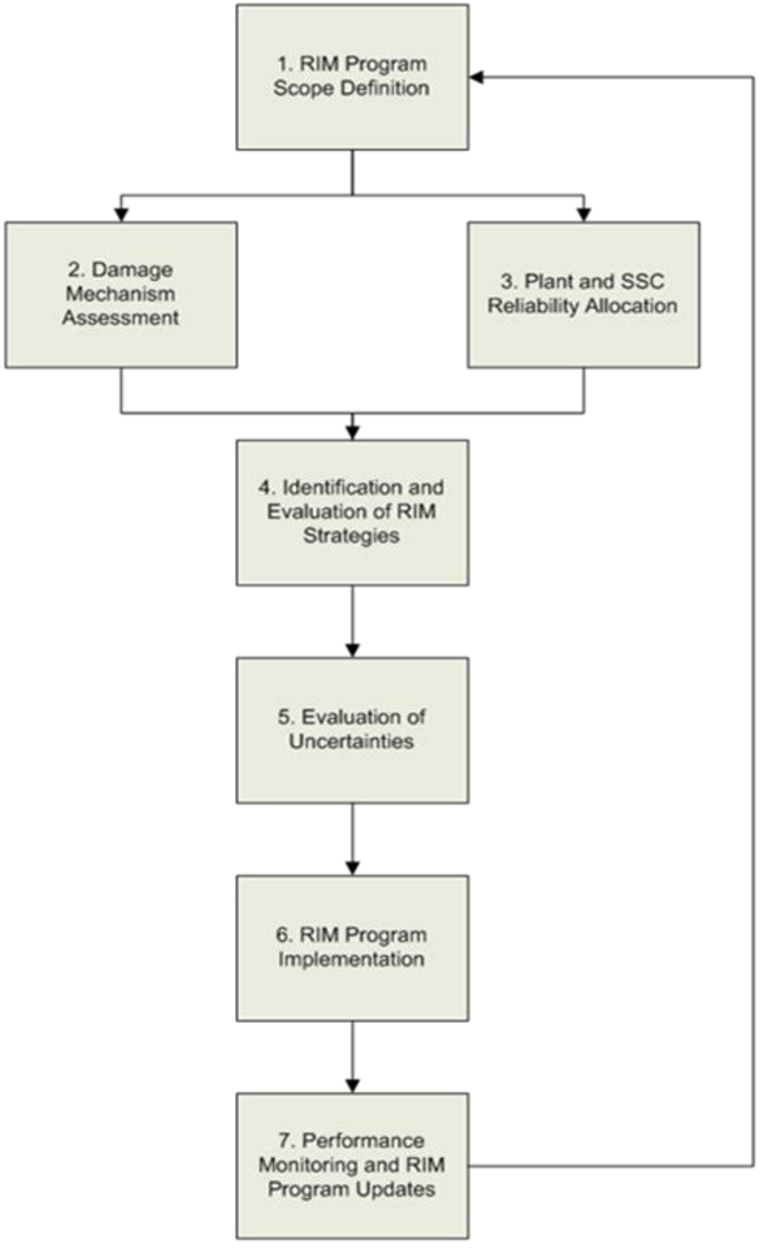
ASME Section XI, Division 2 (RIM) - Introduction

- RIM is a program to ensure the reliability and integrity of components.
- Based on achieving an acceptable level of reliability. Establishes reliability targets from the PRA.
- RIM expert panels ensure that components can be managed through RIM strategies to ensure that reliability targets will be met.
- Allows flexibility for Owners to implement alternative strategies from Section XI, Division 1 requirements.
- “Technology neutral“ – applicable to all reactor designs
- Technology-specific appendices



ASME Section XI, Division 2 (RIM) - Introduction

- Step 1: Determine Scope of SSCs for RIM Program
- Step 2: Evaluate SSC Damage Mechanisms
- Step 3: Determine Plant and SSC Level Reliability and Capability Requirements
- Step 4: Identify and Evaluate RIM Strategies to Achieve Reliability Targets
- Step 5: Evaluate Uncertainties in Reliability Performance
- Step 6: Implement RIM Program
- Step 7: Monitor SSC Reliability Performance and Update RIM Program



RG 1.246 (DG-1383) Structure

Section A

- Purpose
- Applicability (non-LWRs)
- Regulations & Related Guidance

Section B

- Background
- Bases for NRC Staff Positions

Section C

- Staff Regulatory Guidance (Conditions)

Regulatory Aspects

Use of license condition

- Proposed regulatory tool

Difference between LWRs and non-LWRs

- LWRs: 50.55a governs
- Non-LWRs: no clear regulation
  - Part 50 Appendix A is guidance for developing PDC for non-LWRs
    - RG 1.232 ARDC
  - 10 CFR 50.34(b)(6)(iv) and 52.79(a)(29)(i)
    - plans for conducting normal operations, including maintenance, surveillance, and periodic testing of SSCs

DG-1383 Conditions

Use in conjunction with a license condition

- Information to be submitted with the application

Use with 2019 Edition of Section XI-Division 1

Document how aspects of Section XI-Division 2 are considered

Changes / Information to be sent to the NRC

- For review and approval
- For information
- Other information to be available for audit

ANDE-1 not approved for personnel Qualification

DG-1383 Conditions – Cont.

Editions of NQA-1 and RA-S-1.4

- RG 1.28 and Trial Use RG 1.247, respectively

Justify acceptability of the PRA in RIM program

Cannot override construction code NDE without approval

Pre-Service Inspections not addressed in Code / Encoded examination requirement limited to volumetric exams

Information within Code that is in course of preparation is to be developed and submitted for review and approval as appropriate by the applicant

DG-1383 Conditions – Cont.

- Appendix V to be considered for low pressure applications
- Records retention to be IAW QA program requirements
- Stress relaxation to be considered as a degradation mechanism
- Liquid leak test clarifications and hold time limits
- Minor errata type corrections

Next Steps

- DG-1383 public comment period until mid-November (45 days)
- Resolution of comments
- Final RG issued: targeting mid-2022 (ASME Code edition final rule)
- NRC Reviewer and Inspection Guidance
  - Develop plan and vet through management

Questions or Comments?

Future Meeting Planning

- The next periodic stakeholder meeting is scheduled for November 10, 2021.
- If you have suggested topics, please reach out to Prosanta.Chowdhury@nrc.gov.

Advanced Reactor Program

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